6.4 Steam Generator Tube Rupture Transient

6.4.1 Thermal-Hydraulic Analysis for Offsite Radiological Consequences

In support of the Indian Point Unit 2 (IP2) Stretch Power Uprate (SPU) Program, a steam generator tube rupture (SGTR) thermal-hydraulic analysis for calculation of the radiological consequences has been performed. The analysis was performed using the Nuclear Steam Supply System (NSSS) design parameters for a power uprate to a nominal core power of 3216 MWt.

The major hazard associated with an 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. The primary thermal-hydraulic parameters that affect the calculation of doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator, the amount of primary to secondary break flow that flashes to steam and the amount of steam released from the ruptured steam generator to the atmosphere. The radiological consequences analysis will be discussed in subsection 6.11.9 of this report.

6.4.1.1 Input Parameters and Assumptions

The accident analyzed is the double-ended rupture of a single steam generator tube. It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the Reactor Coolant System (RCS), and that reactor trip and safety injection (SI) are automatically initiated on low-pressurizer pressure. Loss-of-offsite power (LOOP) is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the steam generator atmospheric relief valves (ARVs) and/or safety valves. After plant trip and SI actuation, it is assumed that the RCS pressure stabilizes and the break flow equilibrates at the point where incoming SI flow is balanced by outgoing break flow as shown in Figure 6.4-1. The equilibrium primary-to-secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR, at which time it is assumed that the operators have completed the actions necessary to terminate the break flow and the steam releases from the ruptured steam generator.

The analysis does not require that the operators demonstrate the ability to terminate break flow within 30 minutes from the start of the event. It is recognized that the operators may not be able to terminate break flow within 30 minutes for all postulated SGTR events. As discussed below, the LOFTTR2 analysis supports operator actions to terminate break flow at 60 minutes. The purpose of the calculation is to provide conservatively high mass-transfer rates for use in the radiological consequences analysis. This is achieved by assuming a constant break flow at the

equilibrium flow rate, with a constant flashing fraction that does not credit the plant cooldown, for a relatively long time period. Thirty minutes was selected for this purpose. This modeling is consistent with the SGTR analysis presented in Section 14.2.4 of the current *Updated Final Safety Analysis Report* (UFSAR) (Reference 1).

In addition to the above-discussed licensing basis analysis, a supplemental plant response to the event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, and condenser availability. The analysis methodology includes the simulation of the operator actions for recovery from a SGTR based on the IP2 Emergency Operating Procedures (EOPs), which are based on the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs). Conservative operator action times were assumed for analysis purposes and are not intended to serve as a basis for actual operator action times in procedures or training.

The LOFTTR2 analyses were performed for the time period from the SGTR initiation until the primary and secondary pressures were equalized (break flow termination at 60 minutes). 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 primary-to-secondary break flow and 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. The mass release information was compared to the licensing basis analysis to verify that the licensing basis analysis modeling break flow for only 30 minutes is limiting with respect to offsite and control room doses.

After 30 minutes, it is assumed in the licensing basis analysis that steam is released only from the intact steam generators to dissipate the core decay heat and to subsequently cool the plant down to the Residual Heat Removal System (RHRS) operating conditions. It is assumed that the RHRS is capable of removing core decay heat within 30 hours after initiation of the SGTR, and that steam releases are terminated at that time. A primary and secondary side mass and energy balance is used to calculate the steam release for the intact steam generators from 0 to 2 hours, from 2 to 8 hours, and from 8 to 30 hours.

The following analysis assumptions and input parameters were used.

- Analysis methodology is consistent with current UFSAR analysis (Reference 1).
- LOOP is assumed to occur concurrent with the reactor trip.
- The core power is 3216 MWt.

- The RCS average temperature range is 549.0 to 572.0°F.

- The steam generator tube plugging (SGTP) range is 0 to 10 percent.

- The main feedwater temperature range is 390 to 436.2°F

- The low-pressurizer pressure SI actuation setpoint is 1847.7 psia.

- The lowest steam generator safety valve reseat pressure is 885.2 psia. This includes an 18-percent main steam safety valve (MSSV) blowdown, which covers the -3-percent safety valve setpoint tolerance.

- The maximum high-head safety injection (HHSI) flow rates from all 3 HHSI pumps are shown below:

RCS Pressure (psia)	HHSI Flow Rate (lbm/sec)
914.7	163.4
1014.7	144.0
1114.7	130.7
1214.7	114.4
1314.7	96.4
1414.7	71.8
1514.7	40.5
1614.7	0.0

- The time the RHR is capable of removing all decay heat (termination of steam releases) is less than 30 hours after event initiation.

- The break-flow flashing fraction is calculated based on the initial hot leg temperature (605.8°F) for the pre-reactor trip break-flow flashing fraction. Following reactor trip the break-flow flashing fraction is based upon a hot leg temperature equal to the saturation temperature of the RCS pressure where the break-flow rate equals safety injection flow rate (T_{sat}(1500 psia) = 596.2°F).

- The duration of the break-flow to the ruptured steam generator and steam releases from the ruptured steam generator is assumed to be terminated at 30 minutes.

- The minimum total auxiliary feedwater (AFW) flow rate supplied to the plant is 380 gpm.

6.4.1.2 Description of Analyses and Evaluations

The SGTR analysis supports an average temperature (T_{avg}) window range of 549.0°F up to 572.0°F. Plant secondary side conditions (for example, steam pressure, flow, and temperature) are based on high and low tube plugging (0-percent up to 10-percent average/peak) to bound all possible conditions. Four separate cases have been analyzed as follows:

1. T_{avg} = 549.0°F and SGTP = 0 percent
2. T_{avg} = 549.0°F and SGTP = 10-percent average/peak
3. T_{avg} = 572.0°F and SGTP = 0 percent
4. T_{avg} = 572.0°F and SGTP = 10-percent average/peak

In total, 4 cases were considered in the SGTR thermal-hydraulic analysis to bound the operating conditions for the uprate. Note that these 4 cases are individually analyzed to determine the limiting steam release and limiting break flow between 0 and 30 minutes (break-flow termination) for the radiological consequences calculation.

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured steam generator. Since a transient break-flow calculation is not performed for IP2, a detailed time-dependent flashing fraction that incorporates the expected changes in primary side temperatures cannot be calculated. Instead, a conservative calculation of the flashing fraction is performed using the limiting conditions from the break-flow calculation cases. Two time intervals are considered, as in the break-flow calculations: pre- and post-reactor trip (SI initiation occurs concurrently with reactor trip). Since the RCS and steam generator conditions are different before and after the trip, different flashing fractions would be expected.

The flashing fraction is based on the difference between the primary side fluid enthalpy and the saturation enthalpy on the secondary side. Therefore, the highest flashing will be predicted for the case with the highest primary side temperatures. For the flashing-fraction calculations, it is conservatively assumed that all of the break flow is at the hot leg temperature (T_{hot}) (the break is assumed to be on the hot-leg side of the steam generator). Similarly, a lower secondary side pressure maximizes the difference in the primary and secondary enthalpies, resulting in more flashing. The highest possible pre-trip flashing fraction, based on the range of operating conditions covered by this analysis, is for a case with a T_{hot} of 605.8°F, an initial RCS pressure of 2250 psia, and an initial secondary pressure of 590 psia. All cases consider the same post-trip RCS pressure of 1500 psia and post-trip steam generator pressure of 885.2 psia. The post-trip flashing fraction is based on a hot leg temperature at saturation conditions with the RCS at the equilibrium pressure of 1500 psia.

A single calculation is performed to determine long-term steam releases from the intact steam generators for the time interval from the start of the event (0 hours) to 2 hours, 2 hours to 8 hours, and from 8 hours to RHR conditions at 30 hours. The 0- to 2-hour calculations use the 0- to 30-minute intact steam generators' steam release results from the case that resulted in the highest intact steam generators' steam flow rates.

A simple mass and energy balance is assumed in the calculation of the break flow and steam releases. The energy balance is based on the following assumed conditions at 30 minutes:

- The RCS fluid is at the equilibrium pressure and no-load temperature.
- The pressurizer fluid and steam generator secondary fluid for both the ruptured and intact steam generators is at saturation conditions at the no-load temperature.
- The fuel and clad, primary system metal, pressurizer metal, and steam generator secondary metal are at no-load temperature. Since the RCS fluid is not at a consistent energy state with the ruptured steam generator and the remainder of the primary and secondary systems, energy must be dissipated to reduce the RCS fluid from equilibrium pressure and no-load temperature to saturation at no-load temperature.

It is assumed that the plant is then maintained stable at the no-load temperature until 2 hours, and that steam will be released from only the intact steam generators to dissipate the energy from the reduction in the RCS fluid energy state and the core decay heat from 30 minutes to 2 hours.

After 2 hours, it is assumed that plant cooldown to RHR cut-in conditions is initiated by releasing steam from only the intact steam generators. It is assumed that cooldown to RHR cut-in conditions is completed within 8 hours after the SGTR since the cooldown should be accomplished within this time period. However, at 8 hours the RHRS may not be capable of removing all the residual decay heat. Therefore, between 8 and 30 hours steam is released from the intact steam generators to remove the residual decay heat. After the RHR is capable of removing all decay heat, it is assumed that further cooldown is performed using the RHRS, and that the steam release from the intact steam generators is terminated. The energy to be dissipated from 2 to 8 hours and 8 to 30 hours is calculated from an energy balance for the primary and secondary systems between no-load conditions at 2 hours, and the RHR entry conditions at 8 hours, plus the core decay heat load from 2 to 8 hours and 8 to 30 hours. The amount of steam released from the intact steam generators is calculated from a mass and energy balance for the intact steam generators.

6.4.1.3 Acceptance Criteria

There are no criteria associated with the thermal-hydraulic calculations. The results of the calculations are used in the determination of the offsite and control room dose radiological consequences. Acceptance criteria for offsite and control room doses are discussed in subsection 6.11.9 of this report.

6.4.1.4 Results

The tube rupture break flow and ruptured steam generator atmospheric steam releases from 0 to 30 minutes for the 4 different SGTR cases (discussed in subsection 6.4.1.2 of this report) are summarized in Table 6.4-1. Based on the results of these 4 SGTR cases, bounding values for break flow and steam releases are provided in Table 6.4-2, along with the long-term steam releases, and steam generator water mass data to be used in radiological consequences analysis. For an SGTR event, the amount of radioactivity released to the atmosphere is highly dependent on the amount of steam released through the safety valves associated with the ruptured steam generator. Therefore, the worst radiological consequences result from the SGTR case with the greatest amount of steam released. Likewise, a greater break flow results in greater radiological contamination of the secondary side that, in turn, results in a greater amount of activity released along with the steam. Maximum break flow and steam release, therefore, represent bounding values that are conservative for an offsite and control room dose evaluation. Additional margin has been added to the primary-to-secondary break flow and steam releases to allow for future design changes.

The results of the radiological consequences analysis of an SGTR are discussed in subsection 6.11.9 of this document.

6.4.1.5 Conclusions

The SGTR thermal-hydraulic analysis to be used in the radiological consequences calculation has been completed in support of the IP2 SPU Program. Subsection 6.11.9 of this report presents the offsite and control room dose consequences based in the thermal-hydraulic data in Table 6.4-2.

6.4.2 References

1. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report*, Docket No. 50-247.

Table 6.4-1 Case-Specific SGTR Thermal-Hydraulic Results[1]	
Tube Rupture Break Flow for 0 - 30 min.	
T_{avg} = 549.0°F, 0% SGTP	115,794 lbm
T_{avg} = 549.0°F, 10% SGTP	115,920 lbm
T_{avg} = 572.0°F, 0% SGTP	114,285 lbm
T_{avg} = 572.0°F, 10% SGTP	114,630 lbm
Steam Release from Ruptured Steam Generator for Reactor Trip - 30 min.[2]	
T_{avg} = 549.0°F, 0% SGTP	57,965 lbm
T_{avg} = 549.0°F, 10% SGTP	56,964 lbm
T_{avg} = 572.0°F, 0% SGTP	70,231 lbm
T_{avg} = 572.0°F, 10% SGTP	68,207 lbm

Notes:

1. No margin added.
2. Prior to reactor trip the steam flow rate is unaffected by the SGTR.

Table 6.4-2 Bounding SGTR Thermal-Hydraulic Results for Radiological Dose Analysis	
Reactor Trip, SI Actuation, and LOOP	289.8 seconds
Pre-Trip (less than 289.8 sec)	
Tube Rupture Break Flow	29,000 lbm
Percentage of Break Flow which Flashes	21.0%
Steam Release Rate to Condenser	1075.55 lbm/sec for each steam generator
Post-Trip (after 289.8 sec)	
Tube Rupture Break Flow	99,000 lbm
Percentage of Break Flow which Flashes	13.0%
Steam Release from Ruptured Steam Generator up to 30 minutes	77,300 lbm
Steam Released from Intact Steam Generators up to 2 Hours	542,000 lbm
Steam Release from Intact Steam Generator from 2 - 8 Hours	1,090,000 lbm
Steam Release from Intact Steam Generator from 8 - 30 Hours	1,760,000 lbm
Steam Generator Maximum Mass	92,000 lbm/steam generator
Steam Generator Minimum Mass	67,000 lbm/steam generator

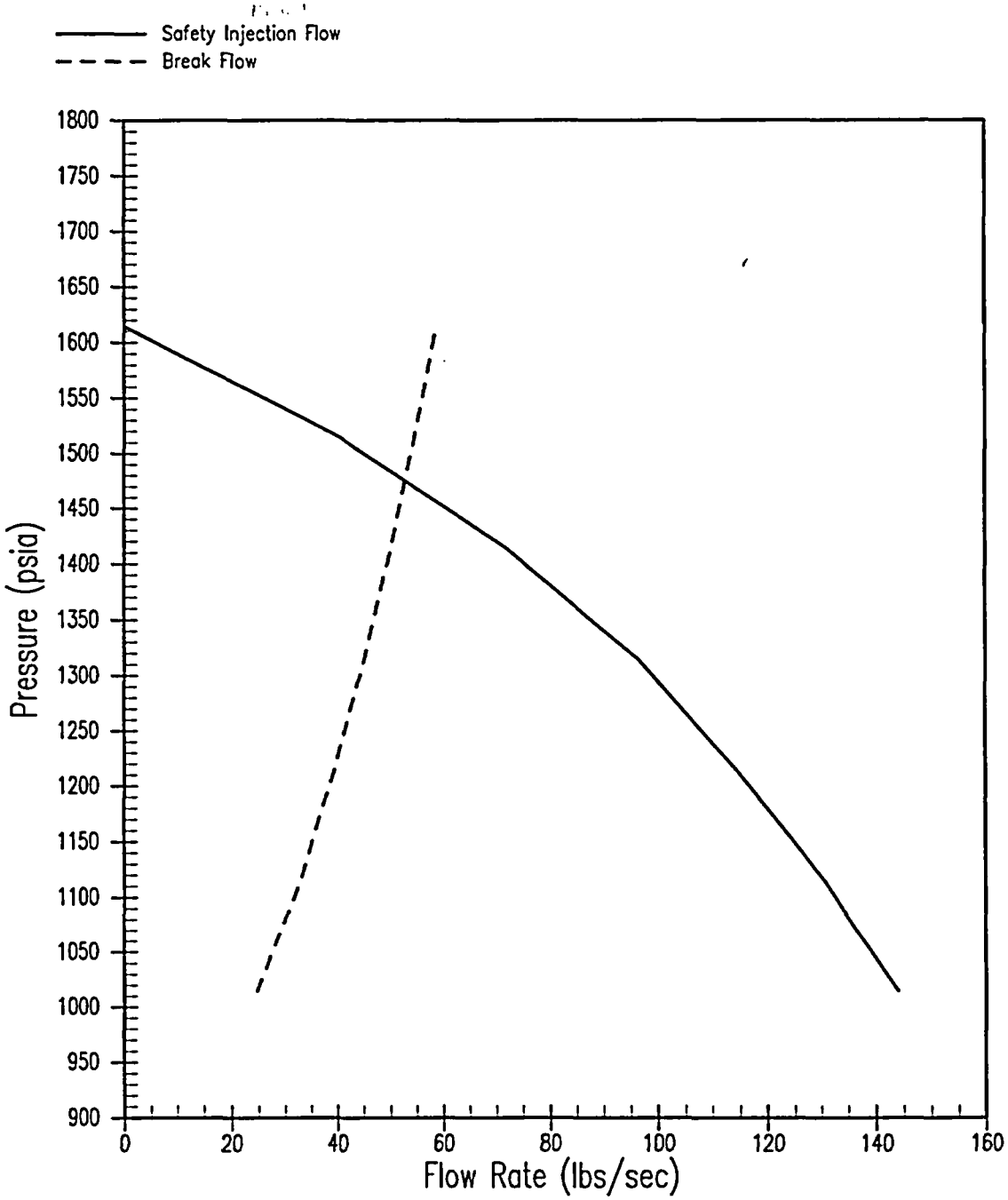


Figure 6.4-1

SI Flow and Break Flow versus RCS Pressure

6.5 LOCA Containment Integrity

The uncontrolled release of pressurized high temperature reactor coolant, termed a loss-of-coolant accident (LOCA), will result in release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. Both the long-term and short-term effects on containment resulting from a postulated LOCA were considered for the Stretch Power Uprate (SPU) Program at Indian Point (IP2).

To demonstrate the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA (LBLOCA), the long-term LOCA mass and energy (M&E) releases were analyzed to approximately 10^7 seconds and used 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 to limit the temperature excursion to less than the Environmental Qualification (EQ) acceptance limits. For this 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 IP2, and has also been used and approved on many plant-specific dockets. Subsection 6.5.1 of this report discusses the long-term LOCA M&E releases generated for this program. The results of this analysis were used in the containment integrity analysis (see subsection 6.5.3).

The short-term LOCA-related M&E 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, loop compartments, and the pressurizer compartment. For the steam generator compartment and the pressurizer vault, the fact that IP2 is approved for leak-before-break (LBB) methodology was used to qualitatively demonstrate that any changes associated with the SPU are offset by the LBB benefit of using the smaller Reactor Coolant System (RCS) nozzle breaks, thus demonstrating that the current licensing bases for these subcompartments remain bounding. Thus, any changes associated with the SPU program will be offset by the LBB benefit and the current Final Safety Analysis Report releases documented in the *IP2 Updated FSAR* (UFSAR) will remain bounding. Subsection 6.5.2 discusses the short-term evaluation conducted for this program.

6.5.1 Long-Term LOCA M&E Releases

The revised M&E release rates described in this section were used as input for the containment pressure calculations discussed in subsection 6.5.3. The M&E releases were revised using the Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version, described in Reference 1. The long-term LOCA M&E releases are provided for the hypothetical double-ended pump suction (DEPS) rupture and double-ended hot leg (DEHL) rupture break cases for IP2 at the SPU conditions.

6.5.1.1 Input Parameters and Assumptions

The M&E release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs were used and instrumentation uncertainties were included. For example, the RCS operating temperatures were chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance of +7.5°F was then added. Nominal parameters were used in certain instances. For example, the RCS pressure in this analysis was based on a nominal value of 2250 psia plus an uncertainty allowance (+60 psi). All input parameters were consistent with accepted analysis methodology.

Some of the most critical items were 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 below. Tables 6.5-1 through 6.5-3 present key data assumed in the analysis.

The core rated power of 3216 MWt was adjusted for calorimetric error (+2.0 percent of power) for use in the analysis. This returns the analysis basis to the 2 percent uncertainty that was used prior to the 1.4-percent Measurement Uncertainty Recapture (MUR) Uprate Program. As previously noted, RCS operating temperatures bounding the highest average coolant temperature range were used in the analysis. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures, which are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband was reflected in the initial RCS temperatures. As previously discussed, the initial RCS pressure in this analysis was based on a nominal value of 2250 psia plus an allowance that accounts for the measurement uncertainty on pressurizer pressure. The selection of 2310 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 to 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 (that is, the core-stored energy). The core-stored energy is increased by 15 percent for conservatism. Thus, the analysis very conservatively accounts for the stored energy in the core.

The RCS volume is increased by 3 percent, which is composed of a 1.6-percent allowance for thermal expansion and a 1.4-percent allowance for uncertainty.

A uniform steam generator tube plugging (SGTP) level of 0 percent was modeled. This assumption maximized the reactor coolant volume and fluid release by including the RCS fluid in all steam generator tubes. During the post-blowdown period, the steam generators are active heat sources since significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0-percent tube plugging assumption maximized heat transfer area and, therefore, the transfer of secondary heat across the steam generator tubes. Additionally, this assumption reduced the reactor coolant loop (RCL) resistance, which reduced the ΔP upstream of the break for the pump suction breaks and increased break flow. Thus, the analysis very conservatively modeled the effects related to SGTP.

The M&E release calculation modeled configurations and failure assumptions to conservatively bound alignments for safety injection (SI) flows. The cases include: a minimum safeguards case (2 high-head safety injection [HHSI], and 1 low-head safety injection [LHSI] pumps); and a maximum safeguards case (3 HHSI, and 2 LHSI pumps).

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

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

- Allowance for RCS temperature uncertainty (+7.5°F)

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

- Core-rated power of 3216 MWt

- Allowance for calorimetric error (+2.0 percent of power)

- Conservative heat transfer coefficient (that is, steam generator primary-to-secondary heat transfer and RCS metal heat transfer)

- Allowance in core-stored energy for the effect of fuel densification

- A margin in core-stored energy that is a statistical value dependent upon fuel type, power level, and burnup

- An allowance for RCS initial pressure uncertainty (+60 psi)

- A maximum containment backpressure equal to design pressure (61.7 psia)

- Minimum RCS loop flow (80,700 gpm/loop)

- Main feedwater addition following a signal to close the flow control valve

- SGTP leveling (0-percent uniform)

  — Maximizes reactor coolant volume and fluid release

  — Maximizes heat transfer area across the steam generator tubes

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

Based on these conditions and assumptions, a bounding analysis of IP2 was made for the release of M&E from the RCS for a postulated LOCA at the SPU core power of 3216 MWt.

6.5.1.2 Description of Analyses

The evaluation model (EM) used for the long-term LOCA M&E release calculations is the March 1979 model described in WCAP-10325-P-A (Reference 1). This EM has been reviewed and approved generically by the NRC. The approval letter is included with Reference 1. This

model has previously been applied to IP2, and also has been used and approved on the plant-specific dockets for other Westinghouse pressurized water reactors (PWRs).

This report section presents the long-term LOCA M&E releases generated in support of the IP2 SPU Program. These M&E releases were used in the containment integrity analysis discussed in subsection 6.5.3.

6.5.1.3 LOCA M&E Release Phases

The containment system receives M&E 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 4 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 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) - 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.

6.5.1.4 Computer Codes

The M&E release evaluation model in WCAP-10325-P-A (Reference 1) comprises 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 IP2.

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 in which the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and water supplied by the Emergency Core Cooling System (ECCS) refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model (see subsection 6.5.1.8.2 of this report).

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

6.5.1.5 Break Size and Location

Generic studies have been performed to determine the limiting postulated break size for LOCA M&E releases. The double-ended guillotine break has been determined to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the RCS loop can be postulated for pipe rupture for any release purposes:

- Hot leg (between vessel and steam generator)
- Cold leg (between pump and vessel)
- Pump suction (between steam generator and pump)

The break locations analyzed for the SPU 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 for each break location.

The DEHL rupture has been shown in previous studies 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 transferred from the steam generator secondary is minimal because the majority of the fluid that exits the core vents directly to containment, bypassing the steam generators. As a result, the reflood M&E releases were reduced significantly as compared to

either the pump suction or cold leg break locations for which 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 (that is, 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 were calculated and presented in this section of the report.

The cold leg break location has also been determined in previous studies 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 addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

6.5.1.6 Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the M&E release rates for each break analyzed. An inherent assumption in the generation of the M&E release is that offsite power is lost. This results in the actuation of the emergency diesel generators (EDGs), which are required to power the Safety Injection System (SIS). This is not an issue for the blowdown period, which is limited by the DEHL break.

Two cases have been 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 EDG. 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 affect the amount of ECCS flow. The analysis of these two cases provides confidence that the effect of credible single failures is bounded.

6.5.1.7 Acceptance Criteria for Analyses

A large-break LOCA (LBLOCA) is classified as an American Nuclear Society (ANS) Condition IV event—an infrequent fault. To satisfy the Nuclear Regulatory Commission (NRC) acceptance criteria presented in the *Standard Review Plan* (SRP), Section 6.2.1.3, the relevant requirements are as follows:

- 10CFR50, Appendix A (Reference 2)
- 10CFR50, Appendix K, paragraph I.A (Reference 3)

To meet these requirements, the following must be addressed:

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

6.5.1.8 Mass and Energy Release Data

6.5.1.8.1 Blowdown M&E Release Data

The SATAN-VI code is used for computing the blowdown transient. The code uses 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 thermo-dynamic 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 WCAP-10325-P-A (Reference 1).

Table 6.5-4 presents the calculated M&E 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; break path 2 refers to the M&E exiting from the steam generator side of the break. Table 6.5-5 presents the mass balance for the DEHL break. Table 6.5-6 presents the energy balance for the DEHL break.

Table 6.5-7 presents the calculated M&E releases for the blowdown phase of the DEPS break with minimum ECCS flows. Table 6.5-13 presents the calculated M&E releases for the blowdown phase of the DEPS break with 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.

6.5.1.8.2 Reflood M&E 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 SI and accumulators, reactor coolant pump (RCP) 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 thermo-dynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates 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; that is, 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 use and application of the M&E release evaluation model (Reference 1) in recent analyses, for example, D. C. Cook Docket (Reference 4). Even though the WCAP-10325-P-A (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 4). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (that is, thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of 2 distinct physical processes. The first is a two-phase interaction with steam condensation 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 must be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data have been reviewed for validation of the containment integrity reflood steam/water mixing model. These data were generated in 1/3-scale tests (Reference 5) and are the largest scale data available and, thus, most clearly simulate the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

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 WCAP-10325-P-A (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 post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture break. For this break, there are two flow paths available in the RCS by which M&E can be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the RCP. 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, which is condensed, that is credited in this analysis. Based upon the postulated break location and the actual physical presence of the ECCS injection nozzle, this assumption is justified. A description of the test and the test results are contained in References 1 and 5.

Tables 6.5-8 and 6.5-14 present the calculated M&E releases for the reflood phase of the pump suction double-ended rupture, minimum safeguards, and maximum safeguards cases, respectively.

The transient response of the principal parameters during reflood are given in Tables 6.5-9 and 6.5-15 for the DEPS cases.

6.5.1.8.3 Post-Reflood M&E Release Data

The FROTH code (Reference 6) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates from the steam generator metal and secondary side water to the 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, a significant amount of reverse heat transfer 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. In 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 steam generator depressurization (see subsection 6.5.1.8.5 of this document for additional information).

The methodology for the use of this model is described in WCAP-10325-P-A (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.

Tables 6.5-10 and 6.5-16 present the two-phase post-reflood M&E release data for the pump suction double-ended cases, minimum and maximum ECCS assumptions.

6.5.1.8.4 Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society (ANS) approved ANS Standard 5.1 (Reference 7) for the determination of decay heat. This standard was used in the M&E release. Table 6.5-22 lists the decay heat curve used in the M&E release analysis, post-blowdown, for the IP2 SPU Program.

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

- Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.
- Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
- Fission rate is constant over the operating history of maximum power level.
- The factor accounting for neutron capture in fission products has been taken from Equation 11 up to 10,000 seconds and from Table 10, both of ANSI/ANS-5.1 (Reference 7), beyond 10,000 seconds.
- The fuel has been assumed to be at full power for 10^8 seconds.

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

- The total recoverable energy associated with 1 fission has been assumed to be 200 MeV/fission.

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

Based upon the NRC staff review as indicated in the *Safety Evaluation Report* (SER) of the March 1979 EM (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.

6.5.1.8.5 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 calculation for steam generator cooldown by removing steam generator secondary energy at different rates (that is, 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.

In the FROTH calculations, steam generator heat removal rates were 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 the saturation temperature at the containment design pressure (61.7 psia). The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The steam generator 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. 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 the current methodology, 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, that is, 14.7 psia and 212°F.

6.5.1.8.6 Sources of M&E

The sources of mass considered in the LOCA M&E release analysis are given in Tables 6.5-5, 6.5-11, and 6.5-17. 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 6.5-6, 6.5-12, and 6.5-18. The energy sources include:

- RCS water
- Accumulator water (all 4 inject)
- Pumped SI water
- Auxiliary feedwater
- Decay heat
- Core-stored energy
- RCS metal (includes steam generator tubes)
- Steam generator metal (includes transition cone, shell, wrapper, and other internals)
- Steam generator secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary, feedwater pump coastdown after the signal to close the flow control valve)

Energy reference points are the following:

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

The M&E 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 Zirconium-water reaction heat was considered because the clad temperature is assumed not to rise high enough for the Zirconium-water reaction heat to be of any significance.

The sequence of events for the LOCA transients are shown in Tables 6.5-19 through 6.5-21.

6.5.1.8.7 Conclusions

The consideration of the various energy sources in the long-term M&E release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in SRP Section 6.2.1.3 have been satisfied. The results of this analysis are used in the containment integrity analysis, as shown in subsection 6.5.3.

6.5.2 Short-Term LOCA Mass and Energy Releases

6.5.2.1 Purpose

An evaluation was conducted to determine the effect of the IP2 SPU on the short-term LOCA-related M&E releases that support subcompartment analyses discussed in Chapter 14.3.5.4 of the IP2 UFSAR (Reference 8). The following text was taken from UFSAR, Chapter 14.3.5.4, which demonstrates that IP2 has been licensed for the application of LBB technology.

> *References 65 and 66 demonstrate that RCS primary loop pipe breaks need not be considered in the structural design basis of the Indian Point 2 Plant. Therefore, implementation of Leak-Before-Break (LBB) Technology has eliminated the large RCS breaks from dynamic consideration. For the LOCA event, the break locations and the break sizes are significantly less severe than the previously mentioned RCS double-ended breaks. The previously calculated subcompartment pressure of 6.4 psi is discussed in 14.3.5.4.1. The subcompartment pressure loadings have been evaluated and it has been determined that the loadings, including LBB and operation at 3083.4 MWt, are less than 6.4 psi. The peak differential pressure across the primary shield wall is bounded by the design pressure of 1000 psi, as discussed in Section 14.3.5.4.1. The effects of the differential operating parameters at 3083.4 MWt do not result in a challenge to the subcompartment designs.*

APPLICABLE FSAR REFERENCES

65. *WCAP-10977, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Indian Point Unit 2," Original - November 1985, Rev. 1 - March 1986, Rev. 2 - December 1986.*

66. *WCAP-10977 Supplement 1, "Additional Information in Support of the Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Indian Point Unit 2," January 1989.*

20. *Letter from Donald S. Brinkman, USNRC to Mr. Stephen B. Bram Nuclear Power Consolidated Edison Company, dated February 23, 1989, "Safety Evaluation of Elimination of Dynamic Effects of Postulated Primary Loop Pipe Ruptures from Design Basis for Indian Point Unit 2 (TAC No* 68318) Docket No. 50-247.*

6.5.2.2 Discussion and Evaluation

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) that 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, thus, the Zaloudek correlation, which models this condition, is currently used in the short-term LOCA M&E release analyses with the SATAN computer program.

This calculation was used to conservatively evaluate the effect of the changes in RCS temperature conditions due to the SPU Program on the short-term releases. This was accomplished by maximizing the reservoir pressure and minimizing the RCS inlet and outlet temperatures for the current analysis of record (AOR), and, by minimizing the RCS inlet and outlet temperatures for the SPU data. Since this maximizes the change in short-term LOCA M&E releases, data representative of the lowest inlet and outlet temperatures with uncertainty subtracted were used for the SPU evaluation of short-term M&E releases.

For this evaluation, an RCS pressure of 2310 psia, a vessel/core inlet temperature of 506.8°F, and a hot leg temperature of 576.2°F were used.

Current Licensing Basis Analyses

IP2 is approved for LBB (Reference 9) for the primary loop and LBB eliminates the dynamic effects of these pipe ruptures from the design basis. This means that the current RCL breaks no longer have to be considered for subcompartment short-term effects. Since these breaks have been eliminated, the next largest branch nozzles must be considered for design verification. The LBB cases that have been analyzed for IP2 are a DEHL break with a 0.5-break area multiplier and a double-ended cold leg (DECL) break with a 0.32-break area multiplier. The 0.5-DEHL break represents a double-ended pressurizer surge line break and the 0.32-DECL represents an accumulator surge line break. These LBB breaks calculated significantly lower pressurizations, (that is, 3.708 psi [DEHL] and 1.5076 psi [DECL]) than calculated for the full double-ended severance of the main RCS piping. The evaluation determined that the increase in subcompartment pressurization due to the lower SPU RCS temperatures resulted in a 0.0353-psi increase in the DEHL break, and a 0.0596-psi increase in the DECL break. These increases, when added to the peak pressurizations listed above, do not result in exceeding the 6.4-psi subcompartment pressurization design limit.

6.5.2.3 Results and Conclusion

The short-term LOCA-related M&E releases discussed in Chapter 14.3 of the UFSAR have been reviewed to assess the effects associated with the SPU Program conditions for IP2. Since IP2 is approved for LBB, the decrease in M&E releases associated with the smaller RCS branch line breaks, as compared to the larger RCS pipe breaks, more than offsets the effects associated with the IP2 SPU Program conditions. Therefore, the current licensing basis subcompartment analyses that consider breaks in the primary loop RCS piping remain bounding.

6.5.3 Long-Term LOCA Containment Response

6.5.3.1 Accident Description

The IP2 containment systems are designed such that for all LOCA break sizes, up to and including the double-ended severance of a reactor coolant pipe, the containment peak pressure remains below the design pressure. This section discusses the containment response subsequent to a hypothetical LOCA. The containment response analysis uses the long-term M&E release data from subsection 6.5.1 of this document.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment. The effect of LOCA M&E releases on the containment pressure is addressed to assure that the containment

pressure remains below its design pressure at the SPU conditions. In support of equipment design and licensing criteria (for example, qualified operating life), long-term containment pressure and temperature transients for post-accident environmental conditions are generated to conservatively bound the potential post-LOCA containment conditions.

6.5.3.2 Input Parameters and Assumptions

An analysis of containment response to the rupture of the RCS must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified for the analysis as shown in Table 6.5-23.

Values for the initial temperature of the service water (SW) and refueling water storage tank (RWST) have been specified, along with containment spray (CS) pump flowrate and reactor containment fan cooler (RCFC) heat removal performance. These values (shown in Tables 6.5-23 and 6.5-24) are chosen conservatively. Long-term sump recirculation is addressed via Residual Heat Removal System (RHRS) heat exchanger performance. The primary function of the RHRS is to remove heat from the core by using the ECCS. Table 6.5-23 provides the RHRS parameters assumed in the analysis.

A series of cases were performed for the LOCA containment response. Subsection 6.5.1 documented the M&E releases for the minimum and maximum safeguards cases for a DEPS break and the releases from the blowdown of a DEHL break.

For the maximum safeguards DEPS case, the failure of a containment spray pump was assumed as the single failure, which leaves available as active heat removal systems: containment spray pump and 4 RCFCs. Table 6.5-25 provides the performance data for 1 spray pump in operation. Emergency safeguards equipment data are given in Table 6.5-23.

The minimum safeguards case was based upon a diesel train failure, DG23, (which leaves available as active heat removal systems: 1 containment spray pump and 4 RCFCs. However, only 3 RCFCs were credited for the DEPs break with minimum ECCS flows. Due to the duration of the DEHL transient (that is, blowdown only), no containment safeguards equipment is modeled.

The calculations for the DEPS minimum safeguards case were performed for 1.0 million seconds (approximately 11 days) and the maximum safeguards case 1.0 million seconds (approximately 11 days). The DEHL cases were terminated soon after the end of the blowdown. The sequence of events for each of these cases is shown in Tables 6.5-26 through 6.5-28.

The following are the major assumptions made in the analysis.

- The M&E released to the containment are described in subsection 6.5.1 of this document for LOCA.

- Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.

- Air is taken as an ideal gas, while compressed water and steam tables are used for water and steam thermo-dynamic properties.

- For the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

- The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the containment fan coolers.

6.5.3.3 Description of COCO Model

Calculation of containment pressure and temperature is accomplished by use of the digital computer code COCO (Reference 10). COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for particular containment design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The COCO code has been used and determined to be acceptable to calculate containment pressure transients for many dry containment plants, most recently including Vogtle Units 1 and 2, Turkey Point Unit 3, Salem Units 1 and 2, Diablo Canyon Units 1 and 2, IP2, and IP3. Transient phenomena within the RCS affect containment conditions by means of convective M&E transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into a water- (pool-) phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of M&E as applied to each system, together with appropriate boundary conditions. As thermo-dynamic equations of state and conditions may vary during the transient, the equations have been derived for possible cases of superheated or

saturated steam, and subcooled or saturated water. Switching between states is handled automatically by the code.

Passive Heat Removal

The significant heat removal source during the early portion of the transient is the containment structural heat sinks. Provision is made in the containment pressure response analysis for heat transfer through, and heat storage in, both interior and exterior walls. Each wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for heat conduction into and out of the node and temperature rise of the node. Table 6.5-29 is the summary of the containment structural heat sinks used in the analysis. The thermal properties of each heat sink material are shown in Table 6.5-30.

The heat transfer coefficient to the containment structure for the early part of the event is calculated based primarily on the work of Tagami (Reference 11). From this work, it was determined that the value of the heat transfer coefficient can be assumed to increase parabolically to a peak value. In COCO, the value then decreases exponentially to a stagnant heat transfer coefficient that is a function of steam-to-air-weight ratio. The heat transfer coefficient (h) for stagnant conditions is based upon Tagami's steady state results.

Tagami presents a plot of the maximum value of the heat transfer coefficient, (h), as function of "coolant energy transfer speed," defined as follows:

$$h = \frac{\text{total coolant energy transferred into containment}}{(\text{containment volume})(\text{time interval to peak pressure})}$$

From this, the maximum heat transfer coefficient of steel is calculated:

$$h_{max} = 75\left(\frac{E}{t_p V}\right)^{0.60} \qquad \text{(Equation 1)}$$

where:

h_{max} = maximum value of h (Btu/hr ft^2 °F)

t_p = time from start of accident to end of blowdown for LOCA and steam line isolation for secondary breaks (sec)

V = containment net free volume (ft^3)

E = total coolant energy discharge from time zero to t_p (Btu)

75 = material coefficient for steel

(Note: Paint is addressed by the thermal conductivity of the material (paint) on the heat sink structure, not by an adjustment on the heat transfer coefficient.) The basis for the equations is a Westinghouse curve fit to the Tagami data.

The parabolic increase to the peak value is calculated by COCO according to the following equation:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)^{0.5}, 0 \le t \le t_p \qquad \text{(Equation 2)}$$

where:

h_s = heat transfer coefficient between steel and air/steam mixture (Btu/hr ft^2 °F)
t = time from start of event (sec)

For concrete, the heat transfer coefficient is taken as 40 percent of the value calculated for steel during the blowdown phase.

The exponential decrease of the heat transfer coefficient to the stagnant heat transfer coefficient is given by:

$$h_s = h_{stag} + \left(h_{max} - h_{stag}\right)e^{-0.05(t-t_p)} \qquad t > t_p \qquad \text{(Equation 3)}$$

where:

h_{stag} = $2 + 50X, 0 < X < 1.4$
h_{stag} = h for stagnant conditions (Btu/hr ft^2 °F)
X = steam-to-air weight ratio in containment

Active Heat Removal

For a large break, the engineered safety features (ESFs) are quickly brought into operation. Because of the brief period of time required to depressurize the reactor coolant system or the main steam system, the containment safeguards are not a major influence on the blowdown peak pressure; however, they reduce the containment pressure after the blowdown and maintain a low, long-term pressure and a low, long-term temperature.

RWST, Injection

During the injection phase of post-accident operation, the ECCS pumps water from the RWST into the reactor vessel. Since this water enters the vessel at RWST temperature, which is less than the temperature of the water in the vessel, it is modeled as absorbing heat from the core until the saturation temperature is reached. SI and containment spray (CS) can be operated for a limited time, depending on the RWST capacity.

RHR, Sump Recirculation

After the supply of refueling water is exhausted, the recirculation system is operated to provide long term cooling of the core. In this operation, water is drawn from the sump, cooled in a RHR exchanger, then pumped back into the reactor vessel to remove core residual heat and energy stored in the vessel metal. The heat is removed from the RHR heat exchanger by the component cooling water (CCW). The RHR heat exchangers and CCW heat exchangers are coupled in a closed-loop system, for which the ultimate heat sink (UHS) is the SW cooling to the CCW heat exchangers.

Containment Spray

CS is an active removal mechanism, which is used for rapid pressure reduction and for containment iodine removal. During the injection phase of operation, the CS pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase, spray is provided by diverting some of the low head SI to the spray rings. However, no credit was taken for recirculation spray in calculating the peak containment pressure.

When a spray droplet enters the hot, saturated, steam-air containment environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the droplet. This mass flow will carry energy to the droplet. Simultaneously, the temperature difference between the atmosphere and the droplet will cause the droplet temperature and vapor pressure to rise. The vapor pressure of the droplet will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the droplet will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling droplet are as follows:

$$\frac{d}{dt}(Mu) = mh_g + q \qquad \text{(Equation 4)}$$

where,

M	=	droplet mass
u	=	internal energy
m	=	diffusion rate
hg	=	steam enthalpy
q	=	heat flow rate
t	=	time

Note that

$$\frac{d}{dt}(M\ ) = m \qquad \text{(Equation 5)}$$

where,

q	=	hcA * (Ts - T)
m	=	kgA * (Ps - Pv)
A	=	drop surface area
h_c	=	coefficient of heat transfer
k_g	=	coefficient of mass transfer
T	=	droplet temperature
T_s	=	steam temperature
P_s	=	steam partial pressure
P_v	=	droplet vapor pressure

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu'.

Both Nu and Nu' may be calculated from the equations of Ranz and Marshall (Reference 12).

$$Nu = 2 + 0.6\,(Re)^{1/2}\,(Pr)^{1/3} \qquad \text{(Equation 6)}$$

where,

Nu = Nusselt number for heat transfer
Pr = Prandtl number
Re = Reynolds number

$$Nu' = 2 + 0.6\,(Re)^{1/2}\,(Sc)^{1/3} \qquad \text{(Equation 7)}$$

where,

Nu' = Nusselt number for mass transfer
Sc = Schmidt number

Thus, Equations 4 and 5 can be integrated numerically to find the internal energy and mass of the droplet as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean droplet produced by the spray nozzles rises to a value within 99 percent of the bulk containment temperature in less than 2 seconds. Detailed calculations of the heatup of spray droplets in post-accident containment atmospheres by Parsly (Reference 13) show that droplets of the size encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical PWR containment. These results confirm the assumption that the containment spray will be 100 percent effective in removing heat from the atmosphere.

RCFC

The RCFCs are another means of heat removal. Each RCFC has a fan that draws in the containment atmosphere from the upper volume of the containment via a return air riser. The RCFCs are cooled by the SW. The steam/air mixture is routed through the enclosed RCFC unit, past essential SW cooling coils. The RCFC then discharges the air through ducting containing a check damper. The discharged air is directed at the lower containment volume. See Table 6.5-24 for the assumed RCFC heat removal capability for the containment response analyses.

6.5.3.4 Acceptance Criteria

A LOCA is an ANS-Condition-IV event—an infrequent fault. The relevant requirements for the containment response for design-basis containment integrity containment response to a design-basis LOCA for containment integrity are as follows:

- General Design Criteria (GDC) 10 and GDC 49 from the UFSAR (Reference 8), Chapter 5.1. To satisfy the requirements of GDCs 10 and 49, the peak calculated containment pressure should be less than the containment design pressure of 47 psig.

- UFSAR (Reference 8), Chapter 9.1, GDC 52.

- UFSAR (Reference 8), Chapter 14.3 requirement that the calculated pressure at 24 hours should be less than 50 percent of the peak calculated value.

6.5.3.5 Analysis Results

The containment pressure, steam temperature, and water (sump) temperature profiles for the DEPS LOCA cases are shown in Figures 6.5-1 through 6.5-4. The results of the DEHL break with minimum ECCS flows are shown in Figures 6.5-5 through 6.5-6. Tables 6.5-32 through 6.5-34 provide detailed results for the analyses.

6.5.3.5.1 DEPS Break with Minimum Safeguards

This analysis assumes a LOOP in coincidence with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single-failure assumption is the failure of a diesel to start, resulting in 1 train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards equipment being available. Furthermore, LOOP delays the actuation times of the safeguards equipment due to the time required for diesel startup after receiving the SI signal.

The postulated RCS break results in a rapid release of M&E to the containment with a resulting rapid rise in the containment pressure and temperature. This rapid rise in containment pressure results in the generation of a containment Hi-1 signal at 1.75 seconds, and a containment Hi-3 signal at 11.32 seconds. The containment pressure continues to rise rapidly in response to the release of M&E, reaching the peak blowdown pressure of 38.533 psig at 24 seconds, and then decreasing slightly as the end of blowdown occurs at 26.4 seconds (pressure of 38.36 psig.) The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a slow process of filling the RCS downcomer in preparation for reflood has begun. During the refill period, the RCFCs start at 61.75 seconds. Since the M&E release during this period is

low and the RCFCs are removing heat, the pressure decreases slightly to 36.82 psig at approximately 46 seconds, the time at which the intact loop accumulators have emptied. The pressure then starts to slowly rise in response to the loss of steam condensation in the RCS loops and the introduction of the accumulator nitrogen gas to the containment.

Containment spray (CS) initiation occurs at 71.48 seconds. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core, which offsets the downcomer head. This reduction in flooding rate and the continued action of the RCFCs and CS leads to a slowly decreasing pressure as the end of reflood is reached at 239.706 seconds. At this time in the transient simulation, by design of the WCAP-10325-P-A (Reference 1) model, energy removal is initiated from the steam generator secondary side at a very increased rate, resulting in a rise in containment pressure from 239.706 seconds out to 1262.4 seconds when sufficient energy has been removed from the steam generators to bring the intact-loops steam generator secondary pressure down to 20 psi below the containment design pressure of 47 psig. The steam generator secondary energy release results in an ultimate containment pressure of 45.71 psig at 1264.1 seconds. After this peak is reached, the M&E release is reduced since the large energy removal from the steam generators has been accomplished.

Containment pressure slowly decreases until the cold leg recirculation time is reached at 1500.46 seconds. After the RHRS is realigned for cold leg recirculation, an increase in the SI temperature (due to delivery from the hot sump and a reduction in steam condensation) results in an increase in containment pressure. Containment spray is terminated at 2354 seconds at the RWST LO-LO level setpoint. By 3600 seconds, the steam generator secondary energy has been reduced to a low value and the containment pressure begins a steady decline. This trend continues until the end of the transient at 1.0 E+07 seconds (~116 days).

6.5.3.5.2 DEPS Break with Maximum Safeguards

The DEPS break with maximum safeguards has a transient history similar to the minimum safeguards case discussed in subsection 6.5.3.5.1 of this report. Table 6.5-27 provides the key sequence of events and Table 6.5-33 shows that a peak pressure of 39.67 psig was calculated at 319 seconds.

6.5.3.5.3 DEHL Break with Minimum Safeguards

This analysis assumes a LOOP in coincidence with a double-ended rupture of the RCS piping between the reactor vessel outlet nozzle and the steam generator inlet (a break in the RCS hot leg). The associated single failure assumption is the failure of a diesel to start, resulting in one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards equipment being available. Furthermore, LOOP delays the

actuation times of the safeguards equipment due to the time required for diesel startup after receipt of the SI signal.

The postulated RCS break results in a rapid release of M&E to the containment with a resulting rapid rise in both the containment pressure and temperature. This rapid rise in containment pressure results in the generation of a containment Hi-1 signal at 1.86 seconds and a containment Hi-3 signal at 9.87 seconds. The containment pressure continues to rise rapidly in response to the release of M&E, reaching the peak blowdown pressure of 40.62 psig at 23.5 seconds and then decreasing slightly as the end of blowdown occurs at 29.2 seconds. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and the process of filling the RCS downcomer in preparation for reflood has begun. Since the reflood for a hot leg break is very fast due to the low resistance to steam venting posed by the broken hot leg, Westinghouse terminates hot leg break M&E release transients at the end of blowdown. The basis for this is further developed in References 1 and 6.

6.5.3.5.4 DEHL Break with Maximum Safeguards

The DEHL break with maximum safeguards was not analyzed since neither the ECCS pumps nor containment safeguards start prior to the end of blowdown. Thus, the maximum ECCS case would be identical to the minimum ECCS case discussed in subsection 6.5.3.5.3 of this report.

6.5.3.6 Conclusions

LOCA containment response analyses have been performed as part of the IP2 SPU Program. The analyses included long-term pressure and temperature profiles for the DEPS minimum and maximum ECCS flow cases. As illustrated in the Table 6.5-31, the analyzed design cases resulted in a peak containment pressure that was less than the containment design pressure of 47 psig. The long-term pressures are well below 50 percent of the peak value within 24 hours. Based on these results, the applicable LOCA criteria for IP2 have been met. Thus, all typical design accident and IP2 containment design criteria have been met at SPU conditions.

6.5.4 References

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

2. 10CFR50, Appendix A, *General Design Criteria for Nuclear Power Plants.*

3. 10CFR50, Appendix K, *ECCS Evaluation Models.*

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

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

6. WCAP-8264-PA (Proprietary), WCAP-8312-A (Nonproprietary), *Topical Report Westinghouse Mass and Energy Release Data For Containment Design*, Rev. 1, Shepard, et al., August 1975.

7. ANSI/ANS-5.1-1979, *American National Standard for Decay Heat Power in Light Water Reactors*, The American Nuclear Society Standards Institute, Inc., LaGrange Park, Illinois, August 1979.

8. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report*, Docket No. 50-247.

9. Letter from D. S. Brinkman (NRC) to S. B. Bram (Nuclear Power Consolidated Edison Company), *Safety Evaluation of Elimination of Dynamic Effects of Postulated Primary Loop Pipe Ruptures from Design Basis for Indian Point Unit 2*, (TAC No. 68318) Docket No. 50-247, February 23, 1989.

10. WCAP-8327 (Proprietary) and WCAP-8326 (Nonproprietary), *Containment Pressure Analysis Code (COCO)*, F. M. Bordelon and E. T. Murphy, July 1974.

11. *Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965*, No. 1, Takashi Tagami,.

12. D. W. Ranz and W. R. Marshall, Jr., "Evaporation for Drops," *Chemical Engineering Progress*, 48, pp.141-146, March 1952.

13. ORNL-TM-2412 Part VI, *Design Consideration of Reactor Containment Spray System. Part VI, The Heating of Spray Drops in Air-Steam Atmospheres*, L. F. Parsly, January 1970.

14. WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Nonproprietary), *Westinghouse LOCA Mass and Energy Release Model for Containment Design*, March 1979.

Table 6.5-1 System Parameters Initial Conditions for IP2 SPU	
	Value
Parameters	SPU
Core Thermal Power (MWt) (without uncertainties)	3216
RCS Total Flowrate (lbm/sec)	34,250
Vessel Outlet Temperature (°F) (with uncertainty)	613.3
Core Inlet Temperature (°F) (with uncertainty)	545.7
Vessel Average Temperature (°F)	579.5
Initial Steam Generator Steam Pressure (psia)	788
SGTP (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	104,300.1
Assumed Maximum Containment Backpressure (psia)	61.7
Accumulator Water volume (ft^3) per accumulator (including line volume) N_2 cover gas pressure (psia) Temperature (°F)	 770 700 130
SI Delay, Total (sec) (from beginning of event) (minimum ECCS case) (maximum ECCS case)	49.1 45

Note:

Core thermal power, RCS total flow rate, RCS coolant temperatures, and steam generator

Table 6.5-2 SI Flow Minimum Safeguards Case for IP2 SPU	
RCS Pressure (psia)	Total Flow (gpm)
Injection Mode (reflood phase)	
14.7	3250.0
34.7	3097.8
54.7	2932.7
61.7	2871.2
74.7	2753.6
94.7	2558.3
114.7	2330.3
214.7.	872.1
Injection Mode (post-reflood phase)	
61.7	2871.2
Cold Leg Recirculation Mode	
61.7	1864.0
Hot Leg Recirculation Mode	
61.7	822.0

Table 6.5-3 Safety Injection Flow Maximum Safeguards Case for IP2 SPU	
RCS Pressure (psia)	Total Flow (gpm)
Injection Mode (reflood phase)	
14.7	6320.50
34.7	5996.18
54.7	5652.86
74.7	5283.84
94.7	4862.22
114.7	4389.80
174.7	1865.02
214.7	1651.00
Injection Mode (post-reflood phase)	
61.7	5523.7
Cold Leg Recirculation Mode	
61.7	6320.5
Hot Leg Recirculation Mode	
61.7	6320.5

Table 6.5-4

DEHL Break
Blowdown M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
.00000	.0	.0	.0	.0
.00107	45431.7	28592.9	45428.6	28589.4
.00213	44959.4	28295.7	44682.5	28114.2
.102	45793.5	29109.1	25686.4	16129.1
.201	33239.6	21496.0	22659.2	14136.4
.301	32619.6	21083.9	20264.4	12470.7
.401	31885.0	20590.9	19026.0	11515.6
.502	31493.9	20329.6	18206.6	10840.9
.601	31480.4	20318.8	17625.0	10345.0
.701	31467.6	20326.8	17235.4	9988.6
.801	31177.0	20177.2	16891.0	9682.4
.902	30811.3	19992.6	16655.5	9455.9
1.00	30442.3	19816.4	16457.2	9266.0
1.10	30178.5	19718.7	16334.0	9128.3
1.20	29985.2	19679.7	16264.2	9030.1
1.30	29765.5	19624.5	16277.2	8984.8
1.40	29462.3	19513.9	16329.9	8966.8
1.50	29085.4	19347.4	16416.8	8972.9
1.60	28711.1	19176.2	16524.2	8994.9
1.70	28394.9	19041.5	16647.5	9030.0
1.80	28092.0	18914.9	16773.3	9071.1
1.90	27710.5	18731.3	16894.8	9114.1
2.00	27250.2	18485.7	17006.0	9155.3
2.10	26796.9	18239.2	17106.1	9193.9
2.20	26405.9	18037.0	17195.7	9229.7
2.30	26026.9	17844.1	17274.6	9262.1
2.40	25597.1	17606.7	17340.4	9289.5
2.50	25129.9	17331.2	17393.6	9311.9
2.60	24686.1	17068.7	17437.0	9330.3
2.70	24290.3	16839.1	17470.2	9344.3
2.80	23893.3	16603.8	17493.4	9353.7
2.90	23492.6	16354.8	17505.6	9358.0
3.00	23104.3	16107.1	17508.7	9357.8
3.10	22737.0	15867.2	17504.4	9354.2

Table 6.5-4 (Cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
3.20	22389.3	15634.4	17493.1	9347.1
3.30	22077.7	15422.8	17476.1	9337.2
3.40	21783.4	15217.4	17454.0	9324.9
3.50	21490.5	15003.4	17426.7	9310.0
3.60	21223.8	14801.9	17394.4	9292.6
3.70	20991.8	14622.4	17358.7	9273.7
3.80	20770.1	14444.9	17319.3	9253.0
3.90	20560.5	14270.0	17275.4	9230.1
4.00	20383.2	14116.6	17227.8	9205.6
4.20	20059.6	13822.3	17120.7	9150.8
4.40	19813.2	13575.6	16995.4	9087.4
4.60	19604.3	13349.2	16852.0	9015.4
4.80	19455.3	13162.8	16689.9	8934.5
5.00	19357.0	13009.0	16509.6	8845.2
5.20	19365.9	12932.0	16313.3	8748.6
5.40	19406.3	12882.9	16099.0	8643.5
5.60	19468.9	12860.5	15867.0	8530.1
5.80	19556.2	12870.9	15615.7	8407.4
6.00	19707.4	12910.1	15356.3	8281.5
6.20	11028.8	8300.7	15067.3	8140.3
6.40	14455.2	10617.0	14672.0	7941.4
6.60	14508.9	10566.7	14237.6	7722.0
6.80	14629.0	10544.4	13845.1	7525.9
7.00	14807.5	10579.1	13465.1	7335.8
7.20	14990.4	10601.2	13051.2	7125.5
7.40	15147.5	10624.3	12651.4	6922.1
7.60	15324.6	10652.1	12283.8	6735.6
7.80	15406.3	10588.1	11920.3	6549.8
8.00	15644.1	10638.4	11545.0	6356.1
8.20	15556.0	10536.5	11181.0	6167.9
8.40	15808.7	10588.4	10841.7	5992.4
8.60	16038.3	10635.9	10516.7	5824.0
8.80	16243.1	10677.0	10199.7	5659.1
9.00	16434.6	10716.8	9899.8	5502.5
9.20	16619.4	10756.8	9613.9	5352.9
9.40	16808.2	10802.1	9342.8	5210.8
9.60	17018.1	10862.2	9081.5	5073.5

Table 6.5-4 (Cont.)

Double-Ended Hot Leg Break
Blowdown Mass and Energy Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
9.80	17302.6	10968.3	8833.8	4943.2
10.0	17347.4	10951.5	8587.4	4813.4
10.2	17122.8	10772.8	8345.5	4686.2
10.4	15365.1	9795.4	8104.7	4559.7
10.602	14402.9	9265.0	7870.3	4437.2
10.604	14399.0	9262.5	7866.6	4435.2
10.607	14396.3	9260.9	7864.2	4433.9
10.609	14394.2	9259.5	7862.2	4432.9
10.8	14314.8	9194.1	7642.1	4318.4
11.0	14303.2	9168.8	7427.1	4207.3
11.2	14286.7	9144.1	7231.0	4106.7
11.4	14268.5	9117.2	7039.7	4008.3
11.6	14235.3	9078.9	6850.8	3911.2
11.8	14133.9	9001.0	6665.1	3816.0
12.0	13860.2	8831.1	6483.7	3723.3
12.2	13361.0	8543.6	6304.3	3632.2
12.4	12862.1	8261.1	6126.6	3542.6
12.6	12547.5	8079.7	5955.6	3457.0
12.8	12329.5	7951.6	5788.3	3373.8
13.0	12138.1	7840.4	5628.4	3294.8
13.2	11932.6	7723.9	5474.2	3219.1
13.4	11697.7	7593.9	5327.5	3147.3
13.6	11418.6	7442.6	5182.9	3076.7
13.8	11117.0	7281.9	5044.5	3009.5
14.0	10808.3	7119.7	4908.8	2944.0
14.2	10513.3	6967.5	4777.7	2881.1
14.4	10232.1	6825.1	4650.0	2820.0
14.6	9944.6	6681.6	4522.3	2759.0
14.8	9642.7	6534.6	4391.9	2696.5
15.0	9312.3	6377.9	4248.5	2627.8
15.2	8962.9	6217.0	4092.0	2554.0
15.4	8607.5	6059.7	3918.8	2473.7
15.6	8230.6	5898.9	3734.6	2389.4
15.8	7823.9	5728.2	3547.8	2303.4
16.0	7408.1	5558.9	3361.8	2216.0
16.2	7003.5	5399.7	3189.1	2131.5
16.4	6605.4	5247.9	3034.3	2052.1
16.6	6205.0	5096.2	2898.8	1978.6
16.8	5809.6	4945.4	2784.2	1913.1

Table 6.5-4 (Cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
17.0	5418.6	4794.4	2687.3	1855.1
17.2	5037.9	4645.9	2604.3	1803.6
17.4	4654.9	4468.2	2532.0	1757.9
17.6	4317.7	4214.3	2465.5	1715.5
17.8	4092.9	4020.9	2404.2	1676.5
18.0	3912.3	3859.9	2346.2	1640.3
18.2	3471.5	3595.3	2289.7	1606.7
18.4	3010.9	3305.0	2234.0	1575.2
18.6	2713.6	3107.4	2178.1	1544.9
18.8	2476.1	2875.7	2123.0	1515.5
19.0	2277.1	2699.9	2073.9	1487.3
19.2	2102.0	2522.1	2025.1	1455.8
19.4	1987.8	2406.6	1972.2	1422.9
19.6	1858.1	2262.5	1904.6	1390.6
19.8	1732.1	2117.2	1821.2	1366.2
20.0	1620.5	1990.4	1718.2	1345.1
20.2	1508.6	1860.9	1595.9	1316.2
20.4	1405.8	1743.3	1474.1	1276.4
20.6	1309.4	1633.1	1369.7	1240.5
20.8	1222.1	1531.5	1267.3	1214.7
21.0	1146.3	1445.3	1143.8	1196.4
21.2	1074.3	1361.7	1014.4	1163.9
21.4	1011.3	1284.7	856.4	1034.6
21.6	959.8	1221.2	744.5	912.0
21.8	916.3	1166.8	642.3	790.3
22.0	877.2	1117.9	575.6	710.2
22.2	825.8	1053.1	520.8	643.3
22.4	774.8	989.3	442.9	548.5
22.6	726.6	929.0	403.6	500.8
22.8	654.6	838.1	374.4	464.9
23.0	611.8	785.3	360.8	448.2
23.2	580.1	746.3	343.4	427.4
23.4	545.2	702.3	325.1	404.9
23.6	504.4	650.4	317.1	395.2
23.8	509.9	654.9	295.6	368.6
24.0	522.0	673.2	269.9	337.2
24.2	514.7	663.5	275.4	344.2
24.4	509.9	657.6	278.5	348.3
24.6	504.6	650.7	274.1	342.8

Table 6.5-4 (Cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
24.8	493.8	636.8	256.7	321.2
25.0	484.3	624.5	254.1	318.3
25.2	474.5	611.6	241.9	303.2
25.4	470.6	606.4	238.5	299.0
25.6	462.9	596.3	227.1	285.0
25.8	450.9	580.8	189.6	238.2
26.0	423.4	545.3	167.1	210.4
26.2	388.3	500.6	178.8	225.3
26.4	381.0	487.6	153.7	193.7
26.6	407.2	521.4	136.6	172.6
26.8	395.8	499.9	156.5	197.7
27.0	430.6	530.6	174.5	220.1
27.2	408.0	513.6	186.8	235.5
27.4	450.7	557.4	175.5	221.5
27.6	432.8	543.9	200.1	252.3
27.8	436.3	542.5	181.9	229.3
28.0	513.0	630.5	183.0	230.9
28.2	537.6	661.0	216.2	272.5
28.4	552.1	677.6	224.9	283.2
28.6	492.9	615.3	203.0	256.0
28.8	316.7	407.5	201.8	254.4
29.0	89.2	116.7	60.2	76.2
29.2	.0	.0	.0	.0
→				

Notes:

1. M&E exiting from the reactor vessel side of the break
2. M&E exiting from the steam generator side of the break

Table 6.5-5 DEHL Break Mass Balance for IP2 SPU				
Time (seconds)		0.00	29.20	29.20+δ
		Mass (thousand lbm)		
Initial	In RCS and ACC	731.97	731.97	731.97
Added Mass	Pumped injection	.00	.00	.00
	Total added	.00	.00	.00
Total Available		731.97	731.97	731.97
Distribution	Reactor coolant	524.25	96.33	123.07
	Accumulator	207.72	129.91	103.17
	Total contents	731.97	226.24	226.24
Effluent	Break flow	.00	505.71	505.71
	ECCS spill	.00	.00	.00
	Total effluent	.00	505.71	505.71
Total Accountable		731.97	731.95	731.95

Table 6.5-6 DEHL Break Energy Balance for IP2 SPU				
Time (seconds)		0.00	29.20	29.20+δ
		Energy (million Btu)		
Initial Energy	In RCS, ACC, S GEN	784.57	784.57	784.57
Added Energy	Pumped injection	.00	.00	.00
	Decay heat	.00	8.37	8.37
	Heat from secondary	.00	-.23	-.23
	Total added	.00	8.14	8.14
Total Available		784.57	792.72	792.72
Distribution	Reactor coolant	305.58	22.91	25.57
	Accumulator	20.67	12.93	10.27
	Core stored	27.00	10.30	10.30
	Primary metal	166.68	155.81	155.81
	Secondary metal	40.99	40.76	40.76
	Steam generator	223.66	222.24	222.24
	Total contents	784.57	464.94	464.94
Effluent	Break flow	.00	327.28	327.28
	ECCS spill	.00	.00	.00
	Total effluent	.00	327.28	327.28
Total Accountable		784.57	792.23	792.23

Table 6.5-7

DEPS Break
(minimum safeguards case)
Blowdown M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
.00000	.0	.0	.0	.0
.00102	84849.3	45618.7	38460.5	20632.5
.00204	40171.1	21550.8	39870.6	21387.9
.00312	40159.2	21545.3	39615.5	21250.0
.101	39922.0	21501.5	19727.2	10572.6
.201	40974.9	22258.1	22318.8	11974.5
.302	44538.4	24465.2	23349.8	12534.5
.402	44682.9	24869.9	23496.8	12619.0
.502	43717.2	24675.3	23121.9	12423.6
.601	44127.3	25230.0	22645.7	12173.7
.701	43629.7	25226.5	22265.5	11975.3
.801	42265.6	24676.5	22065.9	11873.3
.902	40921.3	24115.0	21964.7	11823.2
1.00	39778.0	23661.6	21916.4	11800.6
1.10	38622.2	23215.4	21868.9	11777.7
1.20	37319.2	22690.8	21834.5	11761.1
1.30	35944.7	22103.1	21808.2	11748.2
1.40	34695.2	21548.2	21802.4	11746.0
1.50	33741.2	21128.7	21829.4	11761.1
1.60	32976.7	20806.2	21824.2	11758.5
1.70	32268.2	20512.2	21705.1	11693.9
1.80	31535.6	20200.6	21555.6	11612.9
1.90	30716.4	19828.7	21402.4	11530.0
2.00	29976.9	19501.2	21256.3	11451.0
2.10	29196.0	19140.5	21112.8	11373.8
2.20	28344.1	18723.1	20957.2	11290.1
2.30	27335.3	18191.3	20784.2	11197.1
2.40	26135.3	17523.8	20600.6	11098.5
2.50	24599.3	16613.7	20405.2	10993.7
2.60	22535.4	15316.1	20199.6	10883.6
2.70	21036.9	14396.0	19989.1	10771.0
2.80	20639.2	14205.3	19798.0	10668.9
2.90	19987.2	13800.9	19602.5	10564.7
3.00	19522.8	13516.0	19407.4	10460.8
3.10	19376.6	13446.0	19204.5	10352.8
3.20	19201.3	13342.5	18988.4	10237.7
3.30	18810.9	13089.4	18759.2	10115.5
3.40	18480.2	12877.9	18531.2	9994.0

Table 6.5-7 (Cont.) DEPS Break (minimum safeguards case) Blowdown M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
3.50	18122.3	12639.8	18312.4	9877.7
3.60	17692.9	12347.4	18094.2	9761.8
3.70	17211.2	12017.4	17873.1	9644.3
3.80	16724.6	11684.5	17658.8	9530.6
3.90	16264.9	11369.3	17458.8	9424.7
4.00	15820.9	11062.9	17267.5	9323.6
4.20	14978.6	10480.2	16898.7	9128.8
4.40	14290.6	10002.9	16558.4	8949.6
4.60	13707.5	9594.2	16243.5	8784.0
4.80	13236.1	9259.0	15953.9	8632.0
5.00	12820.2	8956.5	15678.5	8487.5
5.20	12481.3	8705.1	15423.5	8353.9
5.40	12234.8	8509.6	15184.4	8228.7
5.60	12013.2	8329.5	14948.5	8105.1
5.80	11845.2	8184.2	14736.2	7994.2
6.00	11685.0	8040.9	14523.9	7883.0
6.20	11581.6	7932.9	14331.5	7782.7
6.40	11553.1	7869.2	14292.3	7767.5
6.60	12293.5	8327.7	14710.1	8000.1
6.80	11952.9	8084.8	14733.3	8016.3
7.00	10706.7	7749.8	14578.9	7935.9
7.20	9189.8	7146.8	14522.0	7909.0
7.40	8892.9	6975.6	14368.8	7828.6
7.60	8891.4	6931.8	14262.7	7774.8
7.75	8873.2	6896.3	14156.0	7719.6
7.80	8859.4	6880.4	14108.8	7694.8
8.00	8797.6	6810.4	13887.9	7577.7
8.20	8769.7	6749.4	13661.1	7457.0
8.40	8796.7	6713.6	13513.7	7379.2
8.60	8840.2	6682.9	13477.8	7360.1
8.80	8856.5	6634.0	13353.8	7289.4
9.00	8859.7	6593.2	13187.4	7194.4
9.20	8823.9	6536.6	13058.2	7120.2
9.40	8768.8	6472.5	12930.8	7047.5
9.60	8705.1	6398.9	12780.7	6962.8
9.80	8632.0	6315.4	12629.7	6878.0

Table 6.5-7 (Cont.)

DEPS Break
(minimum safeguards case)
Blowdown M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
10.0	8548.7	6227.9	12492.6	6801.4
10.2	8442.2	6132.4	12355.1	6724.5
10.401	8328.6	6042.5	12216.2	6646.9
10.402	8328.1	6042.1	12215.6	6646.6
10.403	8327.4	6041.6	12214.9	6646.2
10.6	8200.0	5951.8	12080.4	6571.1
10.8	8065.0	5865.6	11944.4	6495.3
11.0	7923.0	5780.9	11806.4	6418.6
11.2	7779.9	5700.5	11668.3	6342.2
11.4	7631.7	5620.8	11530.0	6265.9
11.6	7480.7	5541.8	11389.2	6188.4
11.8	7327.0	5463.6	11250.5	6112.2
12.0	7176.5	5390.8	11111.5	6036.0
12.2	7026.3	5316.4	10968.4	5957.5
12.4	6883.8	5243.8	10826.9	5880.1
12.6	6747.1	5172.2	10685.1	5802.7
12.8	6615.8	5101.2	10544.3	5726.0
13.0	6488.9	5030.5	10403.6	5649.5
13.2	6366.6	4960.4	10263.5	5573.5
13.4	6249.0	4891.3	10125.0	5498.4
13.6	6135.2	4822.6	9987.3	5423.8
13.8	6026.2	4755.4	9853.5	5351.4
14.0	5919.5	4688.6	9716.7	5277.2
14.2	5816.8	4623.7	9595.4	5206.8
14.4	5713.0	4557.7	9502.1	5137.0
14.6	5602.6	4486.4	9411.0	5054.8
14.8	5476.2	4402.2	9352.2	4978.2
15.0	5328.9	4296.1	9293.2	4893.3
15.2	5176.5	4174.4	9247.0	4809.1
15.4	5038.0	4049.0	9247.1	4747.4
15.6	4932.8	3939.5	9212.9	4671.6
15.8	4851.8	3846.7	9144.7	4586.4
16.0	4778.1	3768.8	9017.4	4479.5
16.2	4704.7	3700.7	8935.1	4400.9
16.4	4630.2	3639.4	8875.9	4338.7
16.6	4554.4	3583.5	8757.9	4251.9
16.8	4479.1	3533.9	8641.2	4168.4
17.0	4403.7	3489.6	8576.6	4112.2
17.2	4326.9	3449.9	8497.4	4051.1
17.4	4248.8	3415.1	8359.4	3964.2
17.6	4169.2	3386.2	8225.0	3880.8

Table 6.5-7 (Cont.) DEPS Break (minimum safeguards case) Blowdown M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
17.8	4088.8	3361.4	8059.9	3783.8
18.0	4003.3	3338.9	8074.0	3771.4
18.2	3911.2	3320.1	7968.5	3702.9
18.4	3816.0	3306.6	7735.9	3574.5
18.6	3711.0	3295.4	7557.1	3468.0
18.8	3599.3	3288.9	7489.2	3409.1
19.0	3476.9	3281.6	7426.3	3351.2
19.2	3339.5	3274.3	7291.6	3261.6
19.4	3112.8	3201.9	7025.2	3115.2
19.6	2872.0	3112.9	6504.2	2859.2
19.8	2656.5	3017.4	5888.0	2563.0
20.0	2468.0	2906.1	5468.7	2349.1
20.2	2308.0	2789.5	5535.0	2328.4
20.4	2123.5	2598.3	6023.0	2465.5
20.6	1956.7	2408.2	6647.5	2649.6
20.8	1812.0	2238.1	6252.1	2446.3
21.0	1685.8	2087.7	5729.5	2214.5
21.2	1569.8	1948.2	5441.1	2075.4
21.4	1456.6	1811.1	5220.4	1957.8
21.6	1354.4	1686.8	5013.5	1843.9
21.8	1255.2	1565.8	4785.1	1723.3
22.0	1167.3	1458.1	4545.6	1601.6
22.2	1080.0	1351.1	4308.3	1484.2
22.4	1007.1	1261.7	4080.5	1374.5
22.6	925.4	1160.7	3859.9	1271.9
22.8	867.4	1089.4	3646.6	1176.7
23.0	825.3	1037.2	3444.0	1089.3
23.2	789.4	992.7	3240.0	1005.3
23.4	746.7	939.5	3027.4	922.4
23.6	698.4	879.3	2818.3	844.1
23.8	652.2	821.7	2632.9	775.9
24.0	605.7	763.5	2411.8	700.2
24.2	558.3	704.1	2167.0	620.5
24.4	510.1	643.7	1892.7	535.3
24.6	461.8	583.1	1579.6	442.0
24.8	413.7	522.6	1221.0	338.7
25.0	364.5	460.6	822.5	226.8
25.2	313.9	396.9	427.8	117.6
25.4	261.3	330.6	101.7	28.0
25.6	207.1	262.2	.0	.0
25.8	156.1	197.8	.0	.0

Table 6.5-7 (Cont.) DEPS Break (minimum safeguards case) Blowdown M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
26.0	102.9	130.6	.0	.0
26.2	40.0	50.9	.0	.0
26.4	.0	.0	.0	.0

Notes:

1. M&E exiting from the reactor vessel side of the break
2. M&E existing from the steam generator side of the break

Table 6.5-8 DEPS Break (minimum safeguards case) Reflood M&E Releases for IP2 SPU				
Time	Break Path No.1[1]		Break Path No.2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
26.9	.0	.0	.0	.0
27.1	.0	.0	.0	.0
27.2	.0	.0	.0	.0
27.3	.0	.0	.0	.0
27.4	.0	.0	.0	.0
27.42	.0	.0	.0	.0
27.5	31.9	37.5	.0	.0
27.6	13.9	16.4	.0	.0
27.7	12.9	15.2	.0	.0
27.8	18.5	21.8	.0	.0
27.9	23.4	27.6	.0	.0
28.0	29.8	35.1	.0	.0
28.1	35.0	41.2	.0	.0
28.2	39.7	46.7	.0	.0
28.3	44.2	52.1	.0	.0
28.4	47.1	55.5	.0	.0
28.5	51.2	60.3	.0	.0
28.7	54.5	64.2	.0	.0
28.8	58.2	68.6	.0	.0
28.84	60.4	71.1	.0	.0
28.9	61.2	72.1	.0	.0
29.0	64.6	76.1	.0	.0
29.1	67.4	79.4	.0	.0
29.2	70.6	83.1	.0	.0
29.3	73.2	86.2	.0	.0
29.4	75.6	89.1	.0	.0
30.4	97.2	114.5	.0	.0
31.4	115.0	135.6	.0	.0
32.4	130.5	153.9	.0	.0
33.4	144.3	170.2	.0	.0
33.7	148.5	175.2	.0	.0
34.4	279.2	329.8	2803.8	441.0
35.5	382.8	453.1	4084.0	671.4
36.5	380.7	450.7	4057.5	673.4
37.5	374.3	443.1	3986.8	665.2
38.5	367.8	435.4	3914.7	656.5
38.7	366.6	433.8	3900.4	654.8

Table 6.5-8 (Cont.)

DEPS Break
(minimum safeguards case)
Reflood M&E Releases for IP2 SPU

Time	Break Path No.1[1]		Break Path No.2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
39.5	361.5	427.8	3843.7	647.9
40.5	355.4	420.5	3774.2	639.4
41.5	349.5	413.5	3706.6	631.0
42.5	343.8	406.7	3640.9	622.9
43.5	338.3	400.2	3577.1	614.9
44.5	333.0	393.9	3515.2	607.2
44.7	332.0	392.6	3503.0	605.7
45.5	327.9	387.8	3455.1	599.8
46.5	412.6	487.9	233.8	276.9
47.5	566.9	673.5	318.8	379.0
48.5	551.8	655.3	310.6	369.1
49.5	470.2	557.7	326.3	288.5
50.3	446.6	529.3	314.3	272.8
50.5	444.3	526.6	313.1	271.3
51.5	434.2	514.5	307.9	264.6
52.5	424.5	502.9	302.9	258.3
53.5	415.1	491.7	298.1	252.1
54.5	405.9	480.7	293.3	246.1
55.5	397.0	470.1	288.7	240.3
56.5	388.2	459.7	284.2	234.6
56.6	387.4	458.6	283.8	234.0
57.5	379.7	449.5	279.8	229.0
58.5	371.4	439.5	275.6	223.6
59.5	363.2	429.8	271.4	218.4
60.5	355.2	420.3	267.3	213.2
61.5	347.4	411.0	263.3	208.2
62.5	339.8	401.9	259.4	203.3
63.5	332.3	393.0	255.5	198.5
64.5	324.9	384.3	251.8	193.8
65.5	317.8	375.7	248.2	189.2
66.5	310.8	367.4	244.6	184.8
67.5	303.9	359.3	241.1	180.4
68.5	297.2	351.3	237.7	176.2
69.5	290.7	343.5	234.4	172.0
70.5	284.3	336.0	231.1	168.0
71.5	278.1	328.6	228.0	164.1
72.4	272.6	322.1	225.2	160.7
72.5	272.0	321.4	224.9	160.3
73.5	266.1	314.3	221.9	156.6
74.5	260.3	307.5	219.0	153.0
75.5	254.7	300.9	216.2	149.5

Table 6.5-8 (Cont.) DEPS Break (minimum safeguards case) Reflood M&E Releases for IP2 SPU				
Time	Break Path No.1[1]		Break Path No.2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
76.5	249.3	294.4	213.5	146.2
77.5	244.0	288.2	210.8	142.9
78.5	238.9	282.0	208.2	139.7
79.5	233.9	276.1	205.7	136.6
80.5	229.0	270.3	203.3	133.6
81.5	224.2	264.7	200.9	130.7
82.5	219.6	259.2	198.6	127.9
83.5	215.1	253.9	196.4	125.2
84.5	210.8	248.8	194.3	122.5
85.5	206.6	243.8	192.2	120.0
86.5	202.5	239.0	190.2	117.5
87.5	198.6	234.3	188.2	115.1
89.5	191.0	225.4	184.5	110.6
91.5	184.0	217.0	181.0	106.4
93.5	177.4	209.3	177.8	102.4
94.6	174.0	205.2	176.2	100.4
95.5	171.3	202.0	174.9	98.8
97.5	165.6	195.3	172.1	95.5
99.5	160.3	189.1	169.6	92.4
101.5	155.5	183.3	167.2	89.5
103.5	151.0	178.0	165.1	86.9
105.5	146.8	173.1	163.1	84.5
107.5	143.0	168.6	161.3	82.3
109.5	139.5	164.5	159.7	80.3
111.5	136.4	160.8	158.2	78.5
113.5	133.5	157.4	156.8	76.9
115.5	130.8	154.3	155.6	75.4
117.5	128.5	151.4	154.5	74.0
119.5	126.3	148.9	153.5	72.8
121.5	124.4	146.6	152.6	71.7
123.5	122.6	144.6	151.8	70.8
125.0	121.5	143.2	151.2	70.1
125.5	121.1	142.8	151.1	69.9
127.5	119.7	141.1	150.4	69.1
129.5	118.5	139.7	149.8	68.4
131.5	117.4	138.4	149.3	67.8
133.5	116.4	137.2	148.9	67.3
135.5	115.6	136.3	148.5	66.8
137.5	114.9	135.4	148.2	66.4
139.5	114.2	134.6	147.9	66.0
141.5	113.7	134.0	147.6	65.7
143.5	113.2	133.4	147.4	65.4

Table 6.5-8 (Cont.) DEPS Break (minimum safeguards case) Reflood M&E Releases for IP2 SPU				
Time	Break Path No.1[1]		Break Path No.2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
145.5	112.8	133.0	147.2	65.2
147.5	112.5	132.6	147.0	65.0
149.5	112.3	132.4	146.9	64.9
151.5	112.1	132.2	146.8	64.8
153.5	112.0	132.0	146.8	64.7
155.5	111.9	131.9	146.7	64.6
157.5	111.8	131.8	146.7	64.6
159.5	111.8	131.8	146.6	64.5
161.1	111.8	131.7	146.6	64.5
161.5	111.8	131.8	146.6	64.5
163.5	111.8	131.8	146.6	64.5
165.5	111.8	131.8	146.6	64.5
167.5	111.9	131.9	146.7	64.6
169.5	112.0	132.0	146.7	64.6
171.5	112.1	132.2	146.7	64.6
173.5	112.2	132.3	146.8	64.7
175.5	112.4	132.5	146.8	64.8
177.5	112.5	132.6	146.9	64.8
179.5	112.7	132.8	146.9	64.9
181.5	112.9	133.0	147.0	65.0
183.5	113.0	133.3	147.1	65.1
185.5	113.2	133.5	147.1	65.1
187.5	113.4	133.7	147.2	65.2
189.5	113.6	133.9	147.3	65.3
191.5	113.8	134.2	147.4	65.4
193.5	114.0	134.4	147.4	65.5
195.5	114.3	134.7	147.5	65.6
197.5	114.5	134.9	147.6	65.7
199.5	114.7	135.2	147.7	65.8
201.5	114.9	135.4	147.8	65.9
203.5	115.1	135.7	147.9	66.0
205.5	115.3	135.9	147.9	66.1
207.5	115.5	136.1	148.0	66.2
209.5	115.7	136.4	148.1	66.3
211.5	115.9	136.6	148.2	66.4
213.5	116.1	136.9	148.3	66.5
215.5	116.3	137.1	148.4	66.6
217.5	116.5	137.4	148.4	66.7
219.5	116.8	137.6	148.5	66.8
221.5	117.0	137.9	148.6	66.9
223.5	117.2	138.1	148.7	67.0
225.5	117.4	138.4	148.8	67.1

Table 6.5-8 (Cont.) DEPS Break (minimum safeguards case) Reflood M&E Releases for IP2 SPU				
Time	Break Path No.1[1]		Break Path No.2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
227.5	117.6	138.7	148.9	67.2
229.5	117.9	138.9	149.0	67.4
231.5	118.1	139.2	149.1	67.5
233.5	118.3	139.5	149.1	67.6
235.5	118.5	139.7	149.2	67.7
237.5	118.8	140.0	149.3	67.8
239.5	119.0	140.3	149.4	67.9
239.7	119.0	140.3	149.4	67.9

Notes:

1. M&E exiting from the reactor vessel side of the break
2. M&E existing from the steam generator side of the break

Table 6.5-9

DEPS Break
(minimum safeguards case)
Principle Parameters During Reflood for IP2 SPU

Time Seconds	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction (-----)	Core Height (feet)	Downcomer Height (feet)	Flow Fraction (-----)	Injection Total (pounds mass per second)	Injection Accumulator (pounds mass per second)	Injection Spill (pounds mass per second)	Enthalpy Btu/lbm
26.4	190.0	.000	.000	.00	.00	.250	.0	.0	.0	.00
27.1	188.6	20.878	.000	.50	1.17	.000	6713.2	6713.2	.0	99.50
27.4	187.2	24.643	.000	1.09	1.23	.000	6647.4	6647.4	.0	99.50
27.8	186.9	2.520	.126	1.34	2.09	.225	6528.7	6528.7	.0	99.50
28.1	187.0	2.568	.184	1.39	2.78	.288	6457.6	6457.6	.0	99.50
28.8	187.3	2.426	.300	1.50	4.37	.322	6325.9	6325.9	.0	99.50
29.4	187.5	2.365	.373	1.58	5.66	.333	6214.4	6214.4	.0	99.50
33.7	189.4	2.656	.616	2.00	14.86	.353	5534.7	5534.7	.0	99.50
35.5	190.2	4.041	.665	2.18	16.12	.552	4887.5	4887.5	.0	99.50
37.5	191.2	3.817	.693	2.39	16.12	.548	4666.9	4666.9	.0	99.50
38.7	191.8	3.708	.703	2.51	16.12	.545	4554.1	4554.1	.0	99.50
44.7	195.4	3.354	.726	3.00	16.12	.529	4072.9	4072.9	.0	99.50
45.5	195.9	3.319	.727	3.06	16.12	.527	4017.0	4017.0	.0	99.50
46.5	196.5	3.905	.732	3.14	16.05	.638	.0	.0	.0	.00
47.5	197.3	4.688	.732	3.24	15.59	.640	.0	.0	.0	.00
50.3	199.4	3.866	.735	3.51	14.51	.608	358.5	.0	.0	78.02
56.6	204.5	3.371	.738	4.00	13.08	.603	367.3	.0	.0	78.02
64.5	211.7	2.865	.737	4.54	11.70	.596	375.5	.0	.0	78.02
72.4	219.2	2.447	.735	5.00	10.70	.587	381.4	.0	.0	78.02
83.5	229.1	1.994	.731	5.54	9.81	.571	386.6	.0	.0	78.02
94.6	236.8	1.674	.727	6.00	9.37	.554	389.8	.0	.0	78.02
109.5	244.9	1.407	.723	6.52	9.23	.532	392.0	.0	.0	78.02

Table 6.5-9 (Cont.)

DEPS Break
(minimum safeguards case)
Principle Parameters During Reflood for IP2 SPU

Time Seconds	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction (-----)	Core Height (Feet)	Downcomer Height (Feet)	Flow Fraction (-----)	Injection Total (pounds mass per second)	Injection Accumulator (pounds mass per second)	Injection Spill (pounds mass per second)	Enthalpy Btu/lbm
125.0	251.7	1.265	.721	7.00	9.42	.516	393.0	.0	.0	78.02
143.5	258.3	1.196	.724	7.52	9.83	.508	393.4	.0	.0	78.02
161.1	263.8	1.177	.728	8.00	10.30	.506	393.5	.0	.0	78.02
173.5	267.2	1.174	.732	8.33	10.64	.507	393.4	.0	.0	78.02
181.5	269.2	1.175	.734	8.54	10.86	.508	393.4	.0	.0	78.02
199.5	273.4	1.179	.740	9.00	11.35	.510	393.3	.0	.0	78.02
219.5	277.5	1.183	.747	9.50	11.89	.513	393.3	.0	.0	78.02
239.7	281.2	1.188	.755	10.00	12.42	.515	393.2	.0	.0	78.02

Table 6.5-10

DEPS Break
(minimum safeguards case)
Post-Reflood M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
239.8	218.7	272.7	208.8	137.2
244.8	218.2	272.0	208.5	136.8
249.8	217.7	271.4	208.2	136.4
254.8	217.1	270.7	207.9	136.0
259.8	216.5	269.9	207.5	135.6
264.8	216.6	270.0	207.2	135.2
269.8	215.9	269.2	206.9	134.8
274.8	215.2	268.3	206.5	134.4
279.8	215.2	268.3	206.2	134.0
284.8	214.4	267.3	205.9	133.6
289.8	214.3	267.1	205.5	133.2
294.8	213.4	266.1	205.2	132.9
299.8	213.2	265.8	204.9	132.5
304.8	212.9	265.4	204.5	132.1
309.8	211.9	264.2	204.2	131.7
314.8	211.5	263.7	203.8	131.3
319.8	211.0	263.1	203.5	130.9
324.8	210.5	262.4	203.2	130.4
329.8	210.5	262.4	202.8	130.0
334.8	209.8	261.5	202.5	129.6
339.8	209.6	261.3	202.1	129.2
344.8	208.7	260.2	201.8	128.8
349.8	208.3	259.7	201.4	128.4
354.8	207.8	259.1	201.1	128.0
359.8	207.2	258.3	200.7	127.6
364.8	207.0	258.0	200.4	127.2
369.8	206.6	257.6	200.0	126.8
374.8	206.0	256.9	199.7	126.3
379.8	205.3	255.9	199.3	125.9
384.8	204.8	255.3	199.0	125.5
389.8	204.6	255.0	198.6	125.1
394.8	204.0	254.3	198.3	124.7
399.8	203.5	253.8	197.9	124.2
404.8	202.8	252.8	197.5	123.8
409.8	202.4	252.4	197.2	123.4
414.8	202.3	252.2	196.8	123.0
419.8	201.6	251.4	196.5	122.6
424.8	201.1	250.7	196.1	122.2

Table 6.5-10 (Cont.)

DEPS Break
(minimum safeguards case)
Post-Reflood M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
429.8	205.4	256.1	199.9	126.6
434.8	205.1	255.7	199.5	126.1
439.8	204.6	255.1	199.1	125.7
444.8	204.0	254.3	198.8	125.3
449.8	203.4	253.6	198.4	124.8
454.8	85.7	106.9	310.3	154.0
627.6	85.7	106.9	310.3	154.0
627.7	87.5	108.4	308.4	147.8
629.8	87.5	108.3	308.5	147.6
1262.4	87.5	108.3	308.5	147.6
1262.5	74.8	86.1	321.1	31.0
1500.5	71.4	82.2	324.5	31.6
1500.6	71.4	82.2	167.3	63.8
2334.0	64.4	74.1	174.3	65.0
2334.1	64.4	74.1	174.3	65.0
3600.0	57.2	65.8	181.5	66.3
3600.1	54.3	62.5	184.3	48.7
10000.0	39.5	45.5	199.2	52.6
23400.0	31.9	36.7	206.8	54.6
23400.1	31.9	36.7	73.3	19.4
100000.0	21.1	24.3	84.1	22.2
1000000.0	9.1	10.4	96.1	25.4
10000000.0	2.8	3.3	102.4	27.0

Notes:

1. M&E exiting from the reactor vessel side of the break
2. M&E existing from the steam generator side of the break

Table 6.5-11

DEPS Break Mass Balance (minimum safeguards case) for IP2 SPU

		Mass Balance						
Time (seconds)		.00	26.40	26.40+δ	239.71	627.68	1262.40	3600.00
		Mass (thousand lbm)						
Initial	In RCS and ACC	714.25	714.25	714.25	714.25	714.25	714.25	714.25
Added Mass	Pumped injection	.00	.00	.00	74.24	227.83	479.16	1074.54
	Total added	.00	.00	.00	74.24	227.83	479.16	1074.54
***** TOTAL AVAILABLE *****		714.25	714.25	714.25	788.50	942.08	1193.41	1788.79
Distribution	Reactor coolant	524.25	58.26	84.53	144.26	144.26	144.26	144.26
	Accumulator	190.00	126.74	100.47	.00	.00	.00	.00
	Total contents	714.25	185.00	185.00	144.26	144.26	144.26	144.26
Effluent	Break flow	.00	529.24	529.24	644.22	801.15	1052.40	1647.79
	ECCS spill	.00	.00	.00	.00	.00	.00	.00
	Total effluent	.00	529.24	529.24	644.22	801.15	1052.40	1647.79
***** TOTAL ACCOUNTABLE *****		714.25	714.24	714.24	788.48	945.41	1196.65	1792.04

Table 6.5-12

DEPS Break Energy Balance (minimum safeguards case) for IP2 SPU

		Energy Balance						
Time (seconds)		.00	26.40	26.40+δ	239.71	627.68	1262.40	3600.00
		Energy (million Btu)						
Initial Energy	In RCS, ACC, steam generator	781.41	781.41	781.41	781.41	781.41	781.41	781.41
Added Energy	Pumped injection	.00	.00	.00	5.79	17.78	37.38	177.08
	Decay heat	.00	7.52	7.52	30.60	63.24	107.96	235.94
	Heat from secondary	.00	8.54	8.54	8.54	8.54	8.54	8.54
	Total added	.00	16.07	16.07	44.94	89.56	153.89	421.56
***** TOTAL AVAILABLE *****		781.41	797.48	797.48	826.35	870.98	935.30	1202.98
Distribution	Reactor coolant	305.58	13.36	15.98	38.17	38.17	38.17	38.17
	Accumulator	18.95	12.64	10.02	.00	.00	.00	.00
	Core stored	27.00	13.89	13.89	3.95	3.78	3.55	2.71
	Primary metal	166.68	158.73	158.73	131.10	92.65	69.95	53.20
	Secondary metal	40.99	41.26	41.26	38.20	29.44	20.05	15.21
	Steam generator	222.23	237.68	237.68	217.15	162.06	107.20	80.60
	Total contents	781.41	477.56	477.56	428.56	326.09	238.91	189.88
Effluent	Break flow	.00	319.44	319.44	389.59	537.57	680.69	998.74
	ECCS spill	.00	.00	.00	.00	.00	.00	.00
	Total effluent	.00	319.44	319.44	389.59	537.57	680.69	998.74
***** TOTAL ACCOUNTABLE *****		781.41	797.01	797.01	818.15	863.66	919.61	1188.62

Table 6.5-13

DEPS Break
(maximum safeguards case)
Blowdown M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
.00000	.0	.0	.0	.0
.00102	84849.3	45618.7	38460.5	20632.5
.00204	40171.1	21550.8	39870.6	21387.9
.00312	40159.2	21545.3	39615.5	21250.0
.101	39922.8	21502.1	19730.1	10574.2
.201	40979.6	22261.5	22314.8	11972.4
.301	44573.7	24485.7	23347.7	12533.3
.401	44733.5	24903.4	23497.2	12619.1
.502	43754.1	24707.6	23117.1	12420.9
.602	44173.0	25273.0	22640.6	12170.9
.701	43682.7	25278.5	22266.0	11975.5
.802	42297.2	24725.0	22065.4	11873.0
.901	40947.3	24164.9	21967.8	11824.8
1.00	39767.1	23698.6	21919.4	11802.2
1.10	38584.1	23240.8	21873.8	11780.4
1.20	37249.5	22702.1	21840.4	11764.4
1.30	35816.5	22086.0	21815.1	11752.1
1.40	34574.8	21534.6	21810.6	11750.5
1.50	33613.4	21111.2	21839.1	11766.5
1.60	32853.5	20790.6	21835.5	11764.7
1.70	32154.3	20501.2	21717.1	11700.6
1.80	31432.8	20195.4	21569.0	11620.3
1.90	30640.2	19839.4	21417.8	11538.5
2.00	29904.6	19514.1	21273.0	11460.3
2.10	29142.5	19163.4	21131.5	11384.2
2.20	28311.2	18759.2	20976.6	11300.9
2.30	27339.3	18251.3	20805.2	11208.8
2.40	26109.9	17561.4	20624.4	11111.9
2.50	24628.0	16684.9	20436.4	11011.2
2.60	22707.6	15483.7	20235.9	10904.1
2.70	21119.5	14495.7	20024.8	10791.3
2.80	20677.2	14277.7	19830.0	10687.6
2.90	20092.1	13921.6	19641.2	10587.2
3.00	19618.0	13629.3	19452.2	10486.9
3.10	19439.5	13538.0	19253.4	10381.6
3.20	19277.2	13446.3	19038.5	10267.5
3.30	18887.7	13195.0	18811.0	10146.7

Table 6.5-13 (Cont.)

DEPS Break
(maximum safeguards case)
Blowdown M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
3.40	18582.5	13000.8	18590.3	10029.7
3.50	18234.9	12768.3	18376.3	9916.6
3.60	17794.6	12467.4	18158.9	9801.7
3.70	17321.2	12144.4	17944.9	9688.6
3.80	16848.6	11822.4	17737.4	9579.2
3.90	16396.6	11512.6	17538.0	9474.3
4.00	15955.2	11206.7	17348.5	9374.9
4.20	15135.8	10639.0	16986.7	9185.4
4.40	14450.3	10162.8	16655.3	9012.6
4.60	13877.1	9759.9	16344.2	8850.7
4.80	13404.7	9422.8	16060.8	8703.8
5.00	13011.8	9135.4	15789.2	8563.0
5.20	12732.5	8922.1	15540.3	8434.4
5.40	12478.1	8723.0	15299.7	8310.1
5.60	12276.0	8558.7	15078.5	8196.2
5.80	12100.2	8407.6	14860.7	8083.8
6.00	11970.8	8286.4	14662.3	7982.1
6.20	11897.3	8198.2	14466.7	7881.5
6.40	11941.5	8184.6	14308.5	7801.8
6.60	12573.4	8577.2	14836.8	8099.2
6.80	12267.5	8335.5	14923.1	8152.2
7.00	11115.9	8022.4	14778.1	8077.9
7.20	9516.4	7395.4	14698.4	8039.6
7.40	9167.6	7210.5	14540.2	7957.4
7.60	9127.7	7152.4	14376.2	7872.8
7.80	9064.7	7090.9	14232.0	7799.4
8.00	8976.5	7009.1	13995.3	7674.5
8.20	8938.7	6944.0	13762.8	7551.6
8.40	8944.4	6890.1	13558.1	7442.5
8.60	8964.5	6847.8	13401.7	7357.6
8.80	8988.7	6815.2	13346.4	7326.0
9.00	8943.5	6738.5	13265.6	7276.9
9.20	8873.3	6665.8	13116.3	7188.5
9.40	8771.5	6589.3	12977.4	7107.1
9.60	8643.0	6499.3	12853.6	7035.2
9.80	8522.7	6406.2	12713.2	6954.8
10.0	8413.6	6305.9	12565.4	6870.7

Table 6.5-13 (Cont.) DEPS Break (maximum safeguards case) Blowdown M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
10.2	8325.3	6213.6	12425.6	6791.3
10.4	8236.9	6121.7	12287.2	6712.8
10.6	8147.1	6034.8	12149.6	6634.7
10.8	8049.5	5950.5	12013.2	6557.5
11.0	7943.5	5868.5	11876.1	6480.1
11.2	7827.4	5787.9	11739.4	6403.4
11.4	7703.1	5708.8	11602.8	6327.0
11.6	7571.4	5630.8	11466.2	6251.0
11.8	7432.6	5552.9	11327.9	6174.2
12.0	7287.4	5475.4	11190.3	6098.1
12.2	7143.9	5404.4	11053.9	6022.8
12.4	6998.6	5332.3	10913.4	5945.4
12.6	6857.7	5261.2	10773.8	5868.7
12.8	6722.4	5191.7	10634.2	5792.3
13.0	6590.5	5122.9	10494.3	5715.9
13.2	6461.5	5054.6	10355.0	5640.1
13.4	6336.3	4987.1	10215.8	5564.4
13.6	6213.6	4920.1	10076.1	5488.5
13.8	6095.2	4854.0	9939.9	5414.5
14.0	5981.4	4789.2	9804.5	5341.0
14.2	5870.0	4725.2	9667.8	5266.6
14.4	5762.4	4663.0	9548.5	5195.8
14.6	5651.0	4597.9	9450.6	5121.3
14.8	5528.1	4523.8	9346.8	5029.6
15.0	5387.0	4432.0	9292.2	4951.8
15.2	5233.8	4321.4	9221.7	4856.4
15.4	5084.6	4200.2	9195.7	4779.9
15.6	4953.8	4082.5	9185.0	4710.7
15.8	4851.5	3981.3	9150.7	4634.3
16.0	4763.9	3896.2	9065.1	4540.5
16.2	4679.7	3824.4	8924.7	4427.4
16.4	4594.8	3761.2	8849.9	4352.3
16.6	4509.0	3704.4	8774.1	4281.8
16.8	4423.0	3654.2	8624.3	4179.2
17.0	4338.2	3611.9	8508.6	4095.2
17.2	4249.9	3574.2	8449.4	4040.2
17.4	4161.1	3542.1	8334.7	3961.4

Table 6.5-13 (Cont.) DEPS Break (maximum safeguards case) Blowdown M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
17.6	4070.4	3517.3	8134.0	3843.5
17.8	3977.6	3498.7	7692.7	3611.5
18.0	3879.6	3483.8	7925.8	3695.0
18.2	3766.4	3467.4	8408.4	3898.6
18.4	3647.5	3459.9	7772.1	3588.3
18.6	3523.9	3459.4	6787.4	3115.0
18.8	3316.3	3392.0	6792.5	3085.4
19.0	3075.5	3296.5	6861.7	3076.2
19.2	2851.5	3194.6	6591.0	2917.6
19.4	2653.8	3085.2	6186.4	2705.8
19.6	2481.6	2968.9	5760.6	2488.9
19.8	2319.3	2822.8	5425.6	2309.7
20.0	2157.3	2647.7	5521.7	2301.1
20.2	1997.9	2463.6	5978.1	2425.4
20.4	1849.6	2288.4	6479.4	2560.5
20.6	1720.9	2134.5	6024.8	2338.2
20.8	1601.7	1991.3	5615.7	2151.8
21.0	1491.6	1857.6	5333.2	2015.1
21.2	1391.2	1735.8	5105.3	1895.5
21.4	1296.0	1619.3	4883.9	1777.2
21.6	1207.7	1511.3	4646.3	1654.6
21.8	1116.9	1399.1	4398.9	1531.3
22.0	1047.8	1314.8	4158.4	1414.7
22.2	968.7	1216.9	3926.8	1305.8
22.4	922.7	1160.5	3713.1	1207.8
22.6	883.2	1111.8	3507.5	1116.8
22.8	840.3	1058.5	3309.9	1032.7
23.0	800.0	1008.2	3105.1	950.2
23.2	755.0	952.2	2893.1	869.5
23.4	706.2	891.1	2726.2	805.7
23.6	656.7	829.2	2523.1	734.2
23.8	606.2	765.9	2294.3	658.1
24.0	555.7	702.5	2035.7	576.5
24.2	504.9	638.5	1740.4	487.3
24.4	454.2	574.8	1396.2	387.3
24.6	403.2	510.5	1003.6	276.4
24.8	350.6	444.2	589.7	161.7

Table 6.5-13 (Cont.) DEPS Break (maximum safeguards case) Blowdown M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
25.0	296.1	375.3	226.7	62.1
25.2	239.7	303.9	.0	.0
25.4	183.3	232.7	.0	.0
25.6	130.6	165.9	.0	.0
25.8	75.7	96.4	.0	.0
26.0	.0	.0	.0	.0

Notes:

1. M&E exiting the steam generator side of the break
2. M&E exiting the pump side of the break

Table 6.5-14

DEPS Break
(maximum safeguards case)
Reflood M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
26.0	.0	.0	.0	.0
26.5	.0	.0	.0	.0
26.7	.0	.0	.0	.0
26.8	.0	.0	.0	.0
26.9	.0	.0	.0	.0
26.98	.0	.0	.0	.0
27.03	.0	.0	.0	.0
27.1	30.3	35.7	.0	.0
27.2	13.8	16.3	.0	.0
27.3	13.5	15.9	.0	.0
27.4	19.0	22.3	.0	.0
27.6	25.3	29.8	.0	.0
27.7	31.6	37.3	.0	.0
27.8	36.5	43.0	.0	.0
27.9	41.1	48.4	.0	.0
28.0	45.6	53.7	.0	.0
28.1	49.1	57.9	.0	.0
28.2	52.5	61.9	.0	.0
28.3	56.3	66.4	.0	.0
28.4	59.4	70.0	.0	.0
28.5	62.9	74.1	.0	.0
28.6	65.8	77.5	.0	.0
28.7	69.0	81.3	.0	.0
28.9	71.7	84.5	.0	.0
29.0	74.8	88.1	.0	.0
29.1	77.2	91.0	.0	.0
30.1	98.6	116.2	.0	.0
31.1	116.3	137.1	.0	.0
32.1	131.8	155.3	.0	.0
33.1	145.5	171.6	.0	.0
33.3	149.0	175.7	.0	.0
34.1	309.5	365.9	3205.7	507.5
35.2	385.5	456.4	4116.3	676.5
36.2	382.0	452.3	4075.8	675.4
37.2	375.5	444.5	4003.6	667.0
38.2	369.0	436.7	3930.6	658.2
38.3	368.3	435.9	3923.4	657.3

Table 6.5-14 (Cont.)

DEPS Break
(maximum safeguards case)
Reflood M&E Releases for IP2 SPU

Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
39.2	362.6	429.1	3858.8	649.4
40.2	356.4	421.7	3788.7	640.8
41.2	350.4	414.6	3720.5	632.4
42.2	344.7	407.8	3654.2	624.2
43.2	339.2	401.2	3589.9	616.2
44.2	333.8	394.9	3527.5	608.5
44.3	333.3	394.2	3521.4	607.7
45.2	360.2	426.3	3903.8	631.3
46.2	243.9	288.3	1429.4	375.3
47.2	426.1	504.8	354.5	244.8
48.2	423.1	501.2	352.9	242.9
49.2	416.5	493.4	349.9	238.6
50.2	410.1	485.8	347.0	234.6
50.4	408.9	484.3	346.4	233.8
51.2	404.0	478.4	344.2	230.6
52.2	398.0	471.3	341.5	226.8
53.2	392.2	464.4	338.8	223.1
54.2	386.3	457.4	336.2	219.4
55.2	381.2	451.2	333.8	216.1
56.2	376.2	445.4	331.6	213.0
56.9	372.9	441.3	330.0	210.9
57.2	371.5	439.7	329.4	210.0
58.2	366.8	434.1	327.3	207.0
59.2	362.3	428.7	325.2	204.2
60.2	357.9	423.5	323.2	201.5
61.2	353.6	418.4	321.3	198.8
62.2	349.5	413.4	319.4	196.2
63.2	345.4	408.6	317.6	193.6
64.2	341.4	403.8	315.8	191.2
65.2	337.5	399.2	314.0	188.8
66.2	333.7	394.7	312.3	186.4
67.2	330.0	390.3	310.7	184.1
68.2	326.4	386.0	309.0	181.9
69.2	322.9	381.8	307.5	179.8
70.2	319.4	377.7	305.9	177.7
71.2	316.0	373.7	304.4	175.6
71.7	314.4	371.7	303.7	174.6

Table 6.5-14 (Cont.) DEPS Break (maximum safeguards case) Reflood M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
72.2	312.7	369.7	303.0	173.6
73.2	309.5	365.9	301.5	171.7
74.2	306.4	362.2	300.2	169.8
75.2	303.3	358.5	298.8	168.0
76.2	300.3	355.0	297.6	166.2
77.2	297.4	351.5	296.4	164.5
78.2	294.5	348.1	295.2	162.7
79.2	291.6	344.6	294.0	161.0
80.2	288.8	341.3	292.8	159.4
81.2	286.1	338.1	291.7	157.8
82.2	283.5	335.0	290.6	156.2
83.2	280.9	331.9	289.6	154.7
84.2	278.4	329.0	288.5	153.2
85.2	276.0	326.1	287.6	151.8
86.2	273.7	323.3	286.6	150.5
87.2	271.4	320.7	285.7	149.2
89.1	267.3	315.8	284.0	146.8
89.2	267.1	315.5	283.9	146.7
91.2	263.1	310.8	282.3	144.4
93.2	259.4	306.3	280.8	142.2
95.2	255.9	302.2	279.4	140.3
97.2	252.7	298.4	278.1	138.4
99.2	249.7	294.9	276.9	136.8
101.2	246.9	291.6	275.8	135.2
103.2	244.4	288.6	274.8	133.8
105.2	242.0	285.8	273.9	132.5
107.2	239.9	283.2	273.0	131.3
109.1	238.0	281.0	272.3	130.2
109.2	237.9	280.9	272.2	130.2
111.2	236.1	278.8	271.5	129.2
113.2	234.5	276.9	270.9	128.3
115.2	233.0	275.1	270.3	127.5
117.2	231.7	273.6	269.8	126.8
119.2	230.5	272.2	269.3	126.1
121.2	229.5	270.9	268.9	125.5
123.2	228.6	269.8	268.5	125.0
125.2	227.8	268.9	268.2	124.6

Table 6.5-14 (Cont.) DEPS Break (maximum safeguards case) Reflood M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
127.2	227.0	268.0	267.9	124.2
129.2	226.4	267.3	267.7	123.9
131.2	225.9	266.7	267.5	123.6
133.2	225.5	266.2	267.3	123.3
135.2	225.1	265.7	267.2	123.1
137.2	224.8	265.4	267.0	122.9
139.2	224.6	265.1	266.9	122.8
141.2	224.4	264.9	266.9	122.7
143.2	224.3	264.8	266.8	122.6
145.2	224.3	264.7	266.8	122.6
147.2	224.3	264.7	266.8	122.6
149.2	224.6	265.1	266.9	122.7
151.2	225.6	266.4	268.2	123.4
153.2	226.9	267.9	270.5	124.2
154.7	228.0	269.2	272.8	124.9
155.2	228.4	269.6	273.7	125.2
157.2	230.0	271.5	277.6	126.3
159.2	231.6	273.4	281.8	127.4
161.2	233.0	275.1	286.2	128.5
163.2	234.2	276.5	290.6	129.4
165.2	235.2	277.8	295.1	130.4
167.2	236.0	278.7	299.7	131.2
169.2	236.6	279.4	304.2	131.9
171.2	237.0	279.8	308.7	132.5
173.2	237.1	280.0	313.2	133.0
175.2	237.0	279.9	317.8	133.5
177.2	236.8	279.6	322.4	133.9
178.9	236.5	279.2	326.4	134.2

Notes:

1. M&E exiting the steam generator side of the break
2. M&E exiting the pump side of the break

Table 6.5-15

DEPS Break
(maximum safeguards case)
Principle Parameters during Reflood for IP2 SPU

Time Seconds	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction (-----)	Core Height (feet)	Downcomer Height (feet)	Flow Fraction (-----)	Injection Total (pounds mass per second)	Injection Accumulator (pounds mass per second)	Injection Spill (pounds mass per second)	Enthalpy Btu/lbm
26.0	189.3	.000	.000	.00	.00	.250	.0	.0	.0	.00
26.7	187.9	20.997	.000	.50	1.18	.000	6759.0	6759.0	.0	99.50
27.0	186.5	24.771	.000	1.10	1.24	.000	6692.1	6692.1	.0	99.50
27.4	186.3	2.530	.127	1.34	2.12	.228	6571.3	6571.3	.0	99.50
27.7	186.4	2.571	.171	1.38	2.64	.279	6514.3	6514.3	.0	99.50
28.4	186.6	2.434	.294	1.50	4.35	.321	6369.9	6369.9	.0	99.50
29.1	186.9	2.368	.379	1.59	5.82	.333	6242.7	6242.7	.0	99.50
33.3	188.8	2.663	.616	2.00	14.96	.353	5562.9	5562.9	.0	99.50
35.2	189.6	4.048	.667	2.19	16.12	.553	4902.2	4902.2	.0	99.50
37.2	190.6	3.818	.694	2.40	16.12	.549	4684.2	4684.2	.0	99.50
38.3	191.2	3.718	.703	2.50	16.12	.546	4579.7	4579.7	.0	99.50
44.3	194.8	3.360	.726	3.00	16.12	.530	4093.2	4093.2	.0	99.50
45.2	195.4	3.530	.727	3.08	16.12	.553	4511.9	3864.4	.0	96.42
46.2	196.1	2.828	.729	3.15	16.12	.410	1858.2	1183.4	.0	91.70
47.2	196.8	3.885	.733	3.23	16.00	.581	629.7	.0	.0	78.02
50.4	199.3	3.715	.737	3.50	15.54	.579	635.3	.0	.0	78.02
56.9	205.1	3.407	.741	4.00	14.78	.573	647.7	.0	.0	78.02
64.2	212.5	3.145	.743	4.52	14.17	.566	658.0	.0	.0	78.02
71.7	220.5	2.923	.745	5.00	13.75	.560	666.2	.0	.0	78.02
80.2	229.4	2.716	.747	5.51	13.47	.552	674.5	.0	.0	78.02
89.1	237.4	2.544	.749	6.00	13.38	.544	682.4	.0	.0	78.02

Table 6.5-15 (Cont.)

DEPS Break
(maximum safeguards case)
Principle Parameters During Reflood for IP2 SPU

Time Seconds	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction (-----)	Core Height (feet)	Downcomer Height (feet)	Flow Fraction (-----)	Injection Total (pounds mass per second)	Injection Accumulator (pounds mass per second)	Injection Spill (pounds mass per second)	Enthalpy Btu/lbm
99.2	245.0	2.401	.751	6.52	13.46	.537	688.5	.0	.0	78.02
109.1	251.4	2.305	.754	7.00	13.67	.531	692.3	.0	.0	78.02
121.2	258.0	2.230	.757	7.56	14.04	.527	695.0	.0	.0	78.02
131.2	262.8	2.194	.761	8.00	14.41	.525	696.1	.0	.0	78.02
143.2	267.8	2.170	.765	8.52	14.90	.524	696.7	.0	.0	78.02
147.2	269.3	2.165	.767	8.69	15.06	.524	696.7	.0	.0	78.02
154.7	272.0	2.179	.770	9.00	15.37	.528	695.5	.0	.0	78.02
165.2	275.4	2.200	.773	9.44	15.70	.536	693.0	.0	.0	78.02
167.2	276.0	2.200	.774	9.52	15.74	.538	692.6	.0	.0	78.02
178.9	279.3	2.168	.778	10.00	15.95	.544	692.1	.0	.0	78.02

Table 6.5-16 DEPS Break (maximum safeguards case) Post-Reflood M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
179.0	241.6	297.6	487.0	160.2
184.0	241.0	296.8	487.7	159.9
189.0	239.7	295.2	488.9	159.9
194.0	238.9	294.2	489.7	159.7
199.0	238.0	293.1	490.6	159.6
204.0	237.3	292.2	491.4	159.4
209.0	236.4	291.2	492.2	159.3
214.0	235.5	290.0	493.1	159.1
219.0	234.5	288.8	494.2	159.1
224.0	233.8	287.9	494.9	158.9
229.0	232.5	286.3	496.1	158.8
234.0	232.0	285.7	496.6	158.6
239.0	230.9	284.3	497.8	158.5
244.0	230.0	283.3	498.6	158.4
249.0	229.0	281.9	499.7	158.3
254.0	228.1	280.9	500.6	158.1
259.0	235.9	290.5	492.7	159.5
264.0	234.8	289.1	493.9	159.4
269.0	233.9	288.1	494.7	159.2
274.0	232.9	286.8	495.8	159.1
279.0	231.9	285.6	496.7	158.9
284.0	230.8	284.2	497.9	158.8
289.0	229.8	283.0	498.8	158.7
294.0	228.9	281.9	499.7	158.5
299.0	227.9	280.6	500.8	158.4
304.0	226.7	279.1	502.0	158.3
309.0	225.7	277.9	503.0	158.1
314.0	224.5	276.5	504.1	158.0
319.0	92.7	114.1	636.0	192.4
483.3	92.7	114.1	636.0	192.4
483.4	91.8	112.5	636.9	188.1
484.0	91.8	112.4	636.9	188.5
1007.9	91.8	112.4	636.9	188.5
1008.0	78.0	89.7	650.7	54.7
1085.0	76.9	88.5	651.8	54.9
1085.1	76.9	88.5	732.4	211.7

Table 6.5-16 (Cont.) DEPS Break (maximum safeguards case) Post-Reflood M&E Releases for IP2 SPU				
Time	Break Path No. 1[1]		Break Path No. 2[2]	
	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
3600.0	56.7	65.3	752.6	215.4
3600.1	54.3	62.5	755.0	199.4
10000.0	39.5	45.5	769.8	203.3
100000.0	21.1	24.3	788.2	208.1
1000000.0	9.1	10.4	800.2	211.3
10000000.0	2.8	3.3	806.5	213.0

Notes:

1. M&E exiting the steam generator side of the break
2. M&E exiting the pump side of the break

Table 6.5-17

DEPS Break Mass Balance (maximum safeguards case) for IP2 SPU

		Mass Balance						
Time (seconds)		.00	26.00	26.00+δ	178.90	483.36	1007.89	3600.00
		Mass (thousand lbm)						
Initial	In RCS and ACC	714.25	14.25	714.25	714.25	714.25	714.25	714.25
Added Mass	Pumped injection	.00	.00	.00	91.33	313.11	695.31	2786.89
	Total added	.00	.00	.00	91.33	313.11	695.31	2786.89
*** TOTAL AVAILABLE ***		714.25	14.25	714.25	805.58	1027.36	1409.57	3501.15
Distribution	Reactor coolant	524.25	58.13	84.86	152.59	152.59	152.59	152.59
	Accumulator	190.00	28.61	101.88	.00	.00	.00	.00
	Total contents	714.25	86.74	186.74	152.59	152.59	152.59	152.59
Effluent	Break flow	.00	527.50	527.50	652.98	874.76	1256.96	3348.54
	ECCS spill	.00	.00	.00	.00	.00	.00	.00
	Total effluent	.00	527.50	527.50	652.98	874.76	1256.96	3348.54
*** TOTAL ACCOUNTABLE ***		714.25	14.24	714.24	805.57	1027.35	1409.55	3501.13

Table 6.5-18

DEPS Break Energy Balance (maximum safeguards case) for IP2 SPU

		Energy Balance						
Time (seconds)		.00	26.00	26.00+δ	178.90	483.36	1007.89	3600.00
		Energy (million Btu)						
Initial Energy	In RCS, ACC, Steam generator	782.85	782.85	782.85	782.85	782.85	782.85	782.85
Added Energy	Pumped injection	.00	.00	.00	7.13	24.43	54.25	596.15
	Decay heat	.00	7.45	7.45	24.75	51.81	90.88	235.28
	Heat from secondary	.00	0.79	0.79	0.79	3.00	6.13	6.13
	Total added	.00	8.24	8.24	32.67	79.24	151.25	837.56
***** TOTAL AVAILABLE *****		782.85	791.10	791.10	815.52	862.10	934.11	1620.41
Distribution	Reactor coolant	305.58	13.38	16.04	40.37	40.37	40.37	40.37
	Accumulator	18.95	12.82	10.16	.00	.00	.00	.00
	Core stored	27.00	14.02	14.02	3.95	3.78	3.60	2.71
	Primary metal	166.68	158.94	158.94	129.82	93.62	72.69	55.44
	Secondary metal	40.99	41.78	41.78	37.23	29.84	21.09	16.19
	Steam generator	223.66	229.09	229.09	200.11	157.97	112.25	86.71
	Total contents	782.85	470.03	470.03	411.48	325.58	250.00	201.42
Effluent	Break flow	.00	320.59	320.59	395.86	528.34	666.12	1407.59
	ECCS spill	.00	.00	.00	.00	.00	.00	.00
	Total effluent	.00	320.59	320.59	395.86	528.34	666.12	1407.59
***** TOTAL ACCOUNTABLE *****		782.85	790.62	790.62	807.34	853.91	916.11	1609.01

Table 6.5-19 DEHL Break Sequence of Events for IP2 SPU	
Time (sec)	**Event Description**
0.0	Break occurs, LOOP is assumed
0.44	Reactor trip on pressurizer low pressure of 1874.7 psia
3.8	Low pressurizer pressure SI setpoint -1695 psia reached in blowdown
14.0	Broken-loop accumulator begins injecting water
14.2	Intact-loop accumulator begins injecting water
29.2	End-of-blowdown phase

Table 6.5-20 DEPS Break (minimum safeguards case) Sequence of Events for IP2 SPU	
Time (sec)	**Event Description**
0.0	Break occurs, and LOOP is assumed
0.604	Reactor trip on pressurizer low pressure of 1860. psia
4.1	Low pressurizer pressure SI setpoint 1695 psia reached in blowdown
7.75	Main Feedwater Flow Control Valve closed
13.9	Broken-loop accumulator begins injecting water
14.2	Intact-loop accumulator begins injecting water
26.40	End-of-blowdown phase
49.1	SI begins
45.806	Broken-loop accumulator water injection ends
46.256	Intact-loop accumulator water injection ends
239.706	End of reflood phase
1500.46	Cold leg recirculation begins
23400.	Hot leg recirculation begins
1.0E+07	Transient modeling terminated

Table 6.5-21 DEPS Break (maximum safeguards case) Sequence of Events for IP2 SPU	
Time (sec)	**Event Description**
0.0	Break occurs, and LOOP are assumed
0.54	Reactor trip on pressurizer low pressure of 1874.7 psia
4.1	Low pressurizer pressure SI setpoint 1695 psia reached in blowdown
14.0	Broken-loop accumulator begins injecting water
14.3	Intact-loop accumulator begins injecting water
26.0	End-of-blowdown phase
45.	SI begins
45.6	Broken-loop accumulator water injection ends
46.15	Intact-loop accumulator water injection ends
178.9	End-of-reflood phase
1085.	Cold leg recirculation begins
23400.	Hot leg recirculation begins
1.0E+07	Transient modeling terminated

Table 6.5-22 LOCA M&E Release Analysis for IP2 SPU Core Decay Heat Fraction	
Time (sec)	Decay Heat Generation Rate (Btu/Btu)
1.00E+01	0.053876
1.50E+01	0.050401
2.00E+01	0.048018
4.00E+01	0.042401
6.00E+01	0.039244
8.00E+01	0.037065
1.00E+02	0.035466
1.50E+02	0.032724
2.00E+02	0.030936
4.00E+02	0.027078
6.00E+02	0.024931
8.00E+02	0.023389
1.00E+03	0.022156
1.50E+03	0.019921
2.00E+03	0.018315
4.00E+03	0.014781
6.00E+03	0.013040
8.00E+03	0.012000
1.00E+04	0.011262
1.50E+04	0.010097
2.00E+04	0.009350
4.00E+04	0.007778
6.00E+04	0.006958
8.00E+04	0.006424
1.00E+05	0.006021
1.50E+05	0.005323
4.00E+05	0.003770
6.00E+05	0.003201
8.00E+05	0.002834
1.00E+06	0.002580

Table 6.5-23 LOCA Containment Response Analysis Parameters	
SW Temperature (°F)	95
RWST Water Temperature (°F)	110
Initial Containment Temperature (°F)	130
Initial Containment Pressure (psia)	16.7
Initial Relative Humidity (%)	20
Net-Free Volume (ft^3)	2.61×10^6
Reactor Containment Air Recirculation Fan Coolers	
Total	5
Analysis Maximum	4
Analysis Minimum	3
Containment Hi-1 Setpoint (psig)	10.0
Delay Time (sec) with Offsite Power without Offsite Power	 NA 60.0
Containment Spray Pumps	
Total	2
Analysis Maximum	1
Analysis Minimum	1
Flowrate (gpm) Injection Phase (per pump)- see Table 6.5-25 Recirculation Phase (total)	 2180 0
Containment Hi-3 Setpoint (psig)	30.
Delay Time (sec) with Offsite Power (delay after high-high setpoint) without Offsite Power (total time from t=0)	 NA 60.0
ECCS Recirculation Switchover, sec Minimum Safeguards Maximum Safeguards	 1500. 1085.
Containment Spray Termination on LO-LO RWST Level, (sec) Minimum Safeguards Maximum Safeguards	 2345. 1536.

Table 6.5-23 (Cont.) LOCA Containment Response Analysis Parameters	
Emergency Core Cooling System (ECCS) Flows (gpm)	
Minimum ECCS	
Injection Alignment	2871.2
Recirculation Alignment	1864.0
Maximum ECCS	
Injection Alignment	5394.5
Recirculation Alignment	6320.5
Residual Heat Removal System	
RHR Heat Exchangers	
Modeled in Analysis(1)	1
Recirculation Switchover Time, sec	
Minimum Safeguards Maximum Safeguards	1500. 1085.
UA,10^6 * Btu/hr-°F	0.767
Flows – Tube-Side and Shell-Side - gpm	
Minimum Safeguards Maximum Safeguards	4936. 9871.
Component Cooling Water Heat Exchangers	
Modeled in Analysis	2
UA,10^6 * Btu/hr-°F	2.40
Flows – Shell-Side and Tube-Side - gpm	
Shell-side(1)	4936.
Tube-side(1) (SW)	5000.
Additional Heat Loads, (Btu/hr)	19.675×10^6

Note:

1. Minimum safeguard data representing 1 EDG.

Table 6.5-24 RCFC Performance	
Containment Temperature (°F)	Heat Removal Rate [Btu/hr] per RCFC
271	68,703,760
250	56,417,650
230	42,939,170
210	29,265,000
190	23,988,390
170	18,709,700
150	13,471,840
130	8,332,114
110	3,414,706

Table 6.5-25 Containment Spray Performance	
Containment Pressure (psig)	With 1 Pump (gpm)
0-47	2180

Table 6.5-26 DEPS Break Sequence of Events IP2 SPU (minimum safeguards case)	
Time (sec)	**Event Description**
0.0	Break occurs, reactor trip and LOOP power are assumed
0.604	Reactor trip on pressurizer low pressure of 1860. psia
1.75	Containment HI-1 pressure setpoint reached
4.1	Low pressurizer pressure SI setpoint = 1695 psia reached (SI begins coincident with low pressurizer pressure SI setpoint)
7.75	Main Feedwater Flow Control Valve closed
11.43	Containment HI-3 pressure setpoint reached
13.9	Broken-loop accumulator begins injecting water
14.2	Intact-loop accumulator begins injecting water
26.40	End-of-blowdown phase
45.806	Broken-loop accumulator water injection ends
46.256	Intact-loop accumulator water injection ends
49.1	SI begins
61.75	Reactor containment air recirculation fan coolers actuate
71.48	Containment spray pump(s) (RWST) start
239.706	End-of-reflood for MIN SI Case
1264.1.	Peak pressure and temperature occur
1500.46	RHR/HHSI alignment for recirculation
2354	Containment spray is terminated due to RWST LO-LO signal
23400	Hot leg recirculation
1.0×10^7	Transient modeling terminated

Table 6.5-27 DEPS Break Sequence of Events IP2 SPU (maximum safeguards case)	
Time (sec)	**Event Description**
0.0	Break occurs, reactor trip and loss of offsite power are assumed
1.75	Containment HI-1 pressure setpoint reached
4.1	Low pressurizer pressure SI setpoint =1695 psia reached (safety injection begins coincident with low pressurizer pressure SI setpoint)
11.32	Containment HI-3 pressure setpoint reached
14.0	Broken-loop accumulator begins injecting water
14.3	Intact-loop accumulator begins injecting water
26.0	End-of-blowdown phase
45.0	SI begins
45.6	Broken loop accumulator water injection ends
46.15	Intact loop accumulator water injection ends
61.75	Reactor containment air recirculation fan coolers actuate
71.32	Containment spray pump(s) (RWST) start
178.9	End of reflood for MIN SI case
319.	Peak pressure and temperature occur
1085.	RHR/HHSI alignment for recirculation
1536	Containment spray is terminated due to RWST LO-LO signal
23400	Hot leg recirculation begins
1.0×10^7	Transient modeling terminated

Table 6.5-28 DEHL Break Sequence of Events IP2 SPU	
Time (sec)	**Event Description**
0.0	Break occurs, reactor trip and LOOP are assumed
1.86	Containment HI-1 pressure setpoint reached
3.8	Low pressurizer pressure SI Setpoint =1695 psia reached
9.87	Containment HI-3 pressure setpoint reached
14.0	Broken-loop accumulator begins injecting water
14.2	Intact-loop accumulator begins injecting water
23.5	Peak pressure and temperature occur
29.2	End of blowdown phase
29.2	Transient modeling terminated

Table 6.5-29 Containment Heat Sinks			
No.	Material	Heat Transfer Area (ft^2)	Thickness (inch)
1.	Carbon Steel Concrete	41530	0.375 54.0
2.	Carbon Steel Concrete	26012	0.5 42.0
3.	Concrete	13636	12.0
4.	Concrete	55454	12.0
5.	Stainless Steel Concrete	9091	0.375 12.0
6.	Carbon Steel	62538	0.5
7.	Carbon Steel	74276	0.375
8.	Carbon Steel	25407	0.25
9.	Carbon Steel	63454	0.1875
10.	Carbon Steel	2727	0.125
11.	Carbon Steel	20000	0.138
12.	Carbon Steel	9090	0.0625
13.	Stainless Steel PVC Insulation Carbon Steel Concrete	714	0.019 1.25 0.75 54.0
14.	Stainless Steel PVC Insulation Carbon Steel Concrete	6226	0.019 1.25 0.5 54.0
15.	Stainless Steel Foam Insulation Carbon Steel Concrete	3469	0.025 1.5 0.5 54.0
16.	Stainless Steel Foam Insulation Carbon Steel Concrete	3965	0.025 1.5 0.375 54.0

Notes:

1. All carbon steel exterior surfaces are modeled with 0.00033-ft layer of paint on top of a 0.000258-ft layer of carbozinc primer.
2. Approximately 25-ft^2 of the PVC insulation was replaced with fiberglass. As described in Section14.3.5.1.1, modeling the PVC insulation instead of the fiberglass insulation was determined to be conservative and bounding.
3. Approximately 7100-ft^2 of the liner top coat material (Phenoline305) was replaced with Carboline890. As described in Section14.3.5.1.1, modeling the Phenoline305 top coat material instead of the Carboline890 top coat material was determined to be conservative and bounding.

Table 6.5-30 Thermo-Physical Properties of Containment Heat Sinks		
Material	**Thermal Conductivity (Btu/hr-ft - °F)**	**Volumetric Heat Capacity (Btu/ft^3 - °F)**
Paint Layer 1, Phenoline	0.08	28.8
Paint Layer 2, Carbozinc	0.9	28.8
Carbon Steel	26.0	56.35
Stainless Steel	8.6	56.35
Concrete	0.8	28.8
PVC Insulation	0.0208	1.20
Foam Insulation	0.0417	1.53

Table 6.5-31 LOCA Containment Response Results for IP2 SPU (LOOP assumed)				
Case	Peak Press. (psig)	Peak Steam Temp. (°F)	Pressure (psig) @24 hours	Steam Temperature (°F) @24 hours
DEPS MINSI	45.71 @ 1264.1 sec	266.81 @ 1264.1 sec	17.05 @ 24 hrs	204.989 @ 24 hrs
DEPS MAXSI	39.67 @ 319 sec	257.596 @ 319 sec	21.38 @ 24 hrs	216.192 @ 24 hrs
DEHL MINSI	40.62 @ 23.50 sec	259.98 @ 23.49 sec	NA	NA

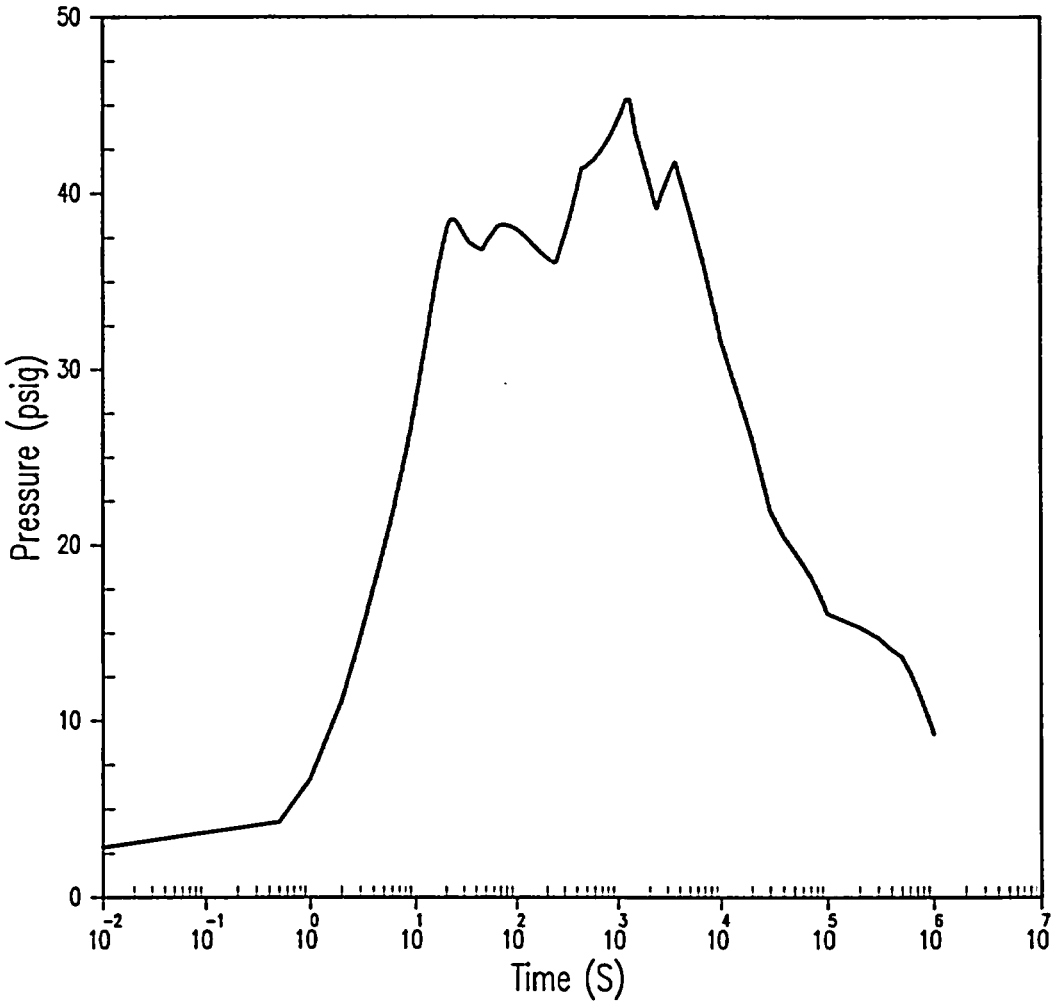
Table 6.5-32 DEPS Break IP2 SPU (minimum safeguards case)			
Time (sec)	Pressure (psig)	Steam Temperature (°F)	Sump Temperature (°F)
.10	2.00	130.00	130.00
.50	4.29	148.20	187.07
1.00	6.71	166.82	202.15
2.00	11.12	195.08	215.04
3.00	14.72	212.53	221.81
4.00	17.55	222.55	226.33
5.00	19.79	228.04	229.73
6.00	21.67	231.04	232.51
7.00	23.41	233.00	235.00
8.00	25.05	234.26	236.99
10.00	28.00	236.57	240.51
11.00	29.43	239.63	242.07
12.00	30.75	242.34	243.47
13.00	31.98	244.78	244.71
14.00	33.12	246.97	245.81
15.00	34.16	248.92	246.81
16.00	35.07	250.58	247.76
17.00	35.84	251.96	248.68
19.00	37.19	254.30	250.27
20.00	37.78	255.31	250.86
21.00	38.21	256.03	251.34
22.00	38.44	256.43	251.77
23.00	38.53	256.57	252.10
25.00	38.48	256.48	252.49
30.00	37.75	255.25	252.37
34.00	37.29	254.47	252.24
42.00	36.93	253.85	247.95
46.00	36.82	253.65	246.31
47.00	36.86	253.68	246.36
53.00	37.37	254.32	246.30
65.00	38.09	255.06	246.24
71.00	38.21	255.22	246.23
78.00	38.22	255.23	246.27
98.00	38.01	254.88	246.40
158.00	36.91	252.96	246.72
198.00	36.39	252.05	246.88
239.00	36.09	251.51	247.01
259.00	36.45	252.15	247.08
349.00	38.88	256.30	247.62
439.00	41.11	259.94	248.44
449.00	41.36	260.33	248.54

Table 6.5-32 (Cont.) DEPS Break IP2 SPU (minimum safeguards case)			
Time (sec)	Pressure (psig)	Steam Temperature (°F)	Sump Temperature (°F)
599.00	41.94	261.22	250.85
699.00	42.45	262.00	252.15
1199.00	45.32	266.25	257.27
1299.00	45.30	266.21	256.73
1499.00	43.31	263.20	250.94
1599.00	42.71	262.27	250.90
2299.00	39.30	256.76	253.98
2399.00	39.18	256.56	254.23
2499.00	39.51	257.10	254.44
2699.00	40.06	258.01	254.88
2999.00	40.73	259.10	255.56
3299.00	41.29	259.99	256.28
3599.00	41.74	260.72	257.02
4999.00	38.99	256.25	260.47
5999.00	37.28	253.33	262.24
6999.00	35.73	250.58	263.52
7999.00	34.29	247.94	263.39
9999.00	31.68	242.88	265.21
19999.00	26.08	230.69	262.49
29999.00	21.94	220.06	255.95
39999.00	20.44	215.76	250.46
49999.00	19.64	213.37	247.11
59999.00	18.89	211.05	244.90
79999.00	17.51	206.54	242.09
89999.00	16.81	204.12	241.13
99999.00	16.12	201.65	240.22
199999.00	15.32	198.16	241.81
299999.00	14.72	195.60	239.95
399999.00	14.06	193.03	237.55
499999.00	13.67	190.41	234.14
599999.00	12.75	186.49	235.85
699999.00	11.81	182.19	232.98
799999.00	10.92	177.82	231.38
899999.00	10.07	173.38	229.56
1000000.00	9.27	168.88	228.34

Table 6.5-33 DEPS Break IP2 SPU (maximum safeguards case)			
Time (sec)	Pressure (psig)	Steam Temperature (°F)	Sump Temperature (°F)
.00	2.00	130.00	130.00
.50	4.29	148.20	187.07
1.00	6.71	166.84	202.15
2.00	11.12	195.07	215.02
3.00	14.74	212.52	221.80
4.00	17.59	222.58	226.34
5.00	19.86	228.14	229.76
6.00	21.79	231.25	232.58
7.00	23.58	233.36	235.11
8.00	25.26	234.73	237.13
9.00	26.79	235.39	238.95
10.00	28.29	237.19	240.64
11.00	29.74	240.27	242.20
12.00	31.08	243.00	243.62
13.00	32.32	245.45	244.88
14.00	33.48	247.66	246.00
15.00	34.55	249.64	247.00
16.00	35.49	251.34	247.94
17.00	36.30	252.77	248.85
19.00	37.69	255.16	250.37
20.00	38.23	256.07	250.88
21.00	38.59	256.68	251.37
22.00	38.78	256.98	251.77
23.00	38.83	257.08	252.09
25.00	38.75	256.93	252.41
30.00	37.99	255.67	252.26
35.00	37.48	254.79	251.48
46.00	37.10	254.13	245.88
66.00	38.09	255.02	246.01
73.00	38.16	255.13	246.08
139.00	38.08	254.98	247.03
161.00	38.13	255.06	247.33
259.00	38.96	256.54	249.91
319.00	39.67	257.60	251.42
359.00	39.13	256.70	252.59
409.00	38.61	255.83	253.80
499.00	37.92	254.63	255.52
599.00	37.29	253.51	256.98
699.00	36.81	252.66	258.07
799.00	36.42	251.94	258.92
999.00	35.80	250.80	260.15

Table 6.5-33 (Cont.) DEPS Break IP2 SPU (maximum safeguards case)			
Time (sec)	Pressure (psig)	Steam Temperature (°F)	Sump Temperature (°F)
1099.00	34.83	249.00	254.94
1399.00	33.29	246.08	257.18
1499.00	32.83	245.19	257.70
1599.00	32.68	244.90	258.21
1699.00	32.93	245.37	258.82
1899.00	33.28	246.05	260.08
2199.00	33.62	246.70	261.87
2499.00	33.82	247.07	263.47
2899.00	33.90	247.24	265.30
3599.00	33.70	246.85	267.92
3699.00	33.40	246.28	268.21
3999.00	32.36	244.26	268.90
4999.00	29.94	239.35	269.65
5999.00	28.42	236.08	269.10
6999.00	27.38	233.73	268.12
7999.00	26.58	231.91	267.06
8999.00	25.93	230.36	266.02
9999.00	25.33	228.92	265.14
19999.00	24.09	225.25	262.40
29999.00	23.86	223.98	262.13
39999.00	23.42	222.86	260.84
49999.00	23.77	222.65	261.82
59999.00	22.77	219.98	260.59
69999.00	22.19	218.41	260.46
99999.00	20.67	214.15	257.92
199999.00	20.57	212.54	257.72
299999.00	20.40	211.29	257.54
399999.00	20.18	210.09	255.93
599999.00	19.80	207.45	256.28
799999.00	19.36	204.87	254.17
899999.00	18.94	203.80	255.22
999999.00	18.66	202.90	254.71
4299999.00	18.40	201.94	252.90
7499999.00	17.90	200.18	252.55
9999999.00	17.45	198.68	250.15
10000000.00	17.24	197.64	250.15

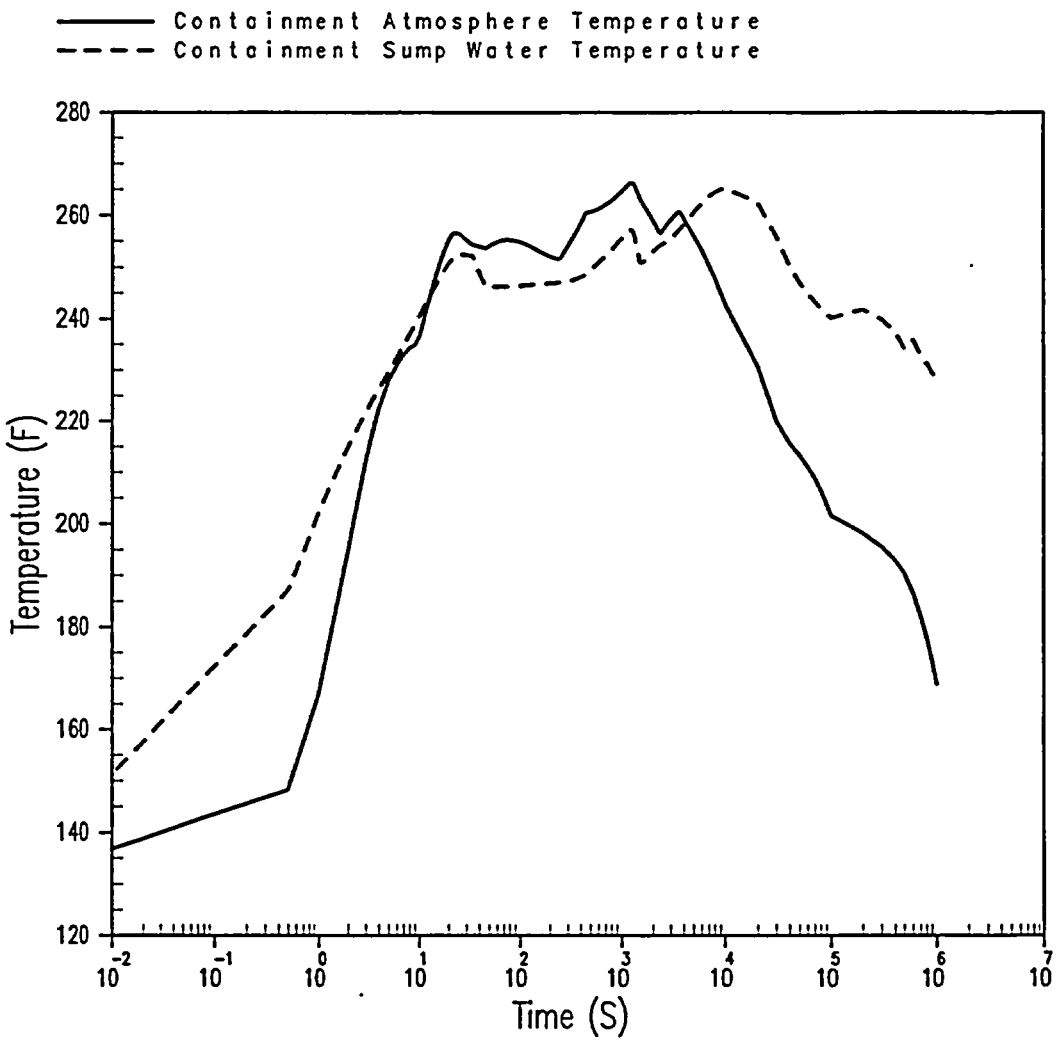
Table 6.5-34 DEHL Break IP2 SPU			
Time (sec)	Pressure (psig)	Steam Temperature (°F)	Sump Temperature (°F)
.00	2.00	130.00	130.00
.50	4.60	150.75	181.36
1.00	6.68	166.64	196.08
2.00	10.56	191.77	210.73
3.00	14.12	209.90	219.20
4.00	17.27	222.38	225.20
5.00	20.06	230.96	229.98
6.00	22.62	237.16	234.04
7.00	24.74	240.63	236.87
8.00	26.73	243.09	239.39
9.00	28.54	244.60	241.69
10.00	30.23	245.54	243.86
11.00	31.69	245.56	245.70
12.00	33.07	246.89	247.25
13.00	34.38	249.33	248.58
14.00	35.56	251.46	249.71
15.00	36.61	253.31	250.63
16.00	37.55	254.94	251.35
17.00	38.39	256.34	251.80
18.00	39.11	257.54	252.05
19.00	39.67	258.45	252.17
20.00	40.06	259.09	252.24
21.00	40.35	259.56	252.26
22.00	40.55	259.86	252.23
24.00	40.62	259.98	252.17
29.00	40.46	259.72	252.04
30.00	40.30	259.48	252.01
→			



DG23 fails and 3 RCFCs and 1 spray pump operate.

Figure 6.5-1

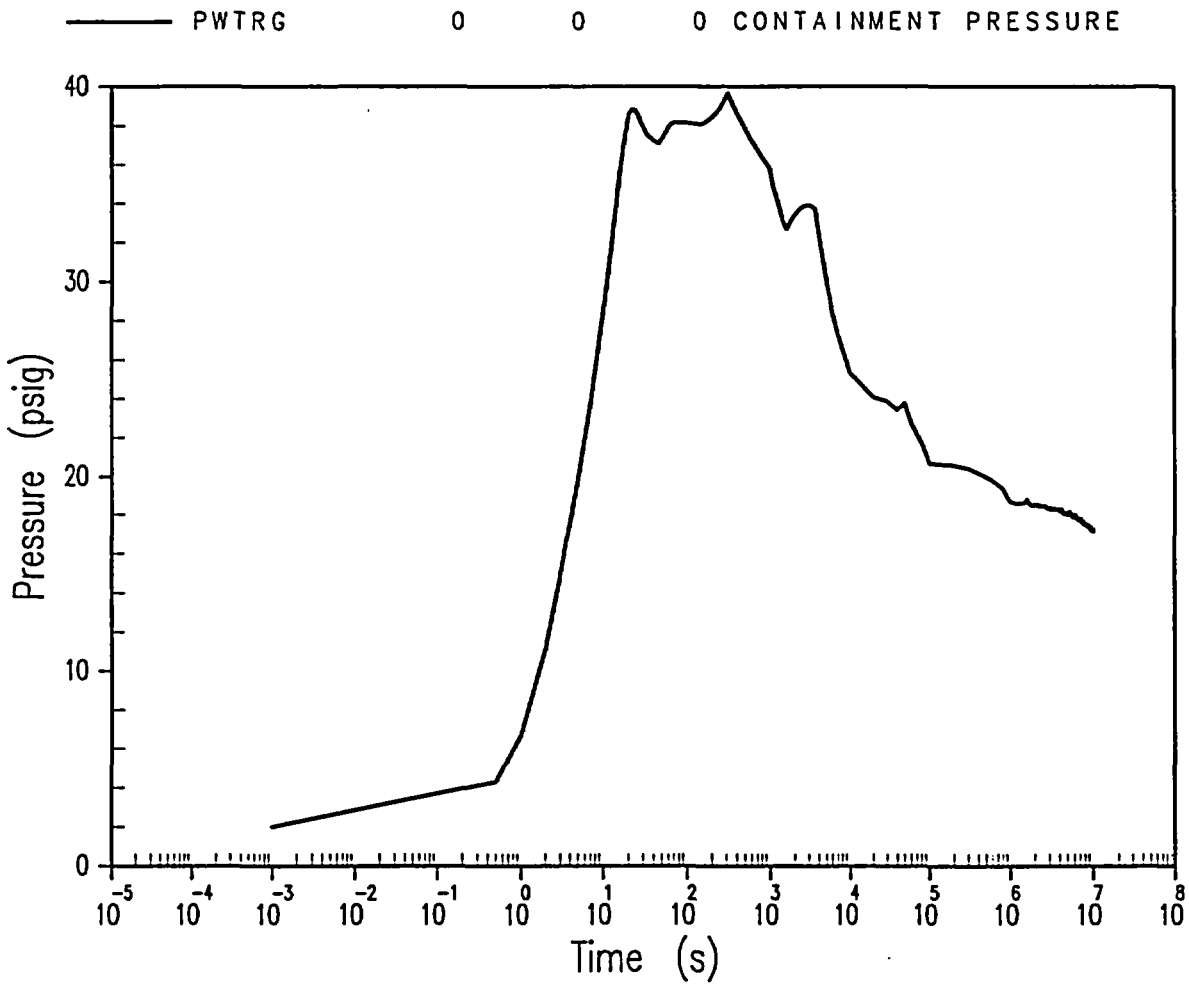
DEPS Break (minimum safeguards) Containment Pressure



DG23 fails and 3 RCFCs and 1 spray pump operate.

Figure 6.5-2

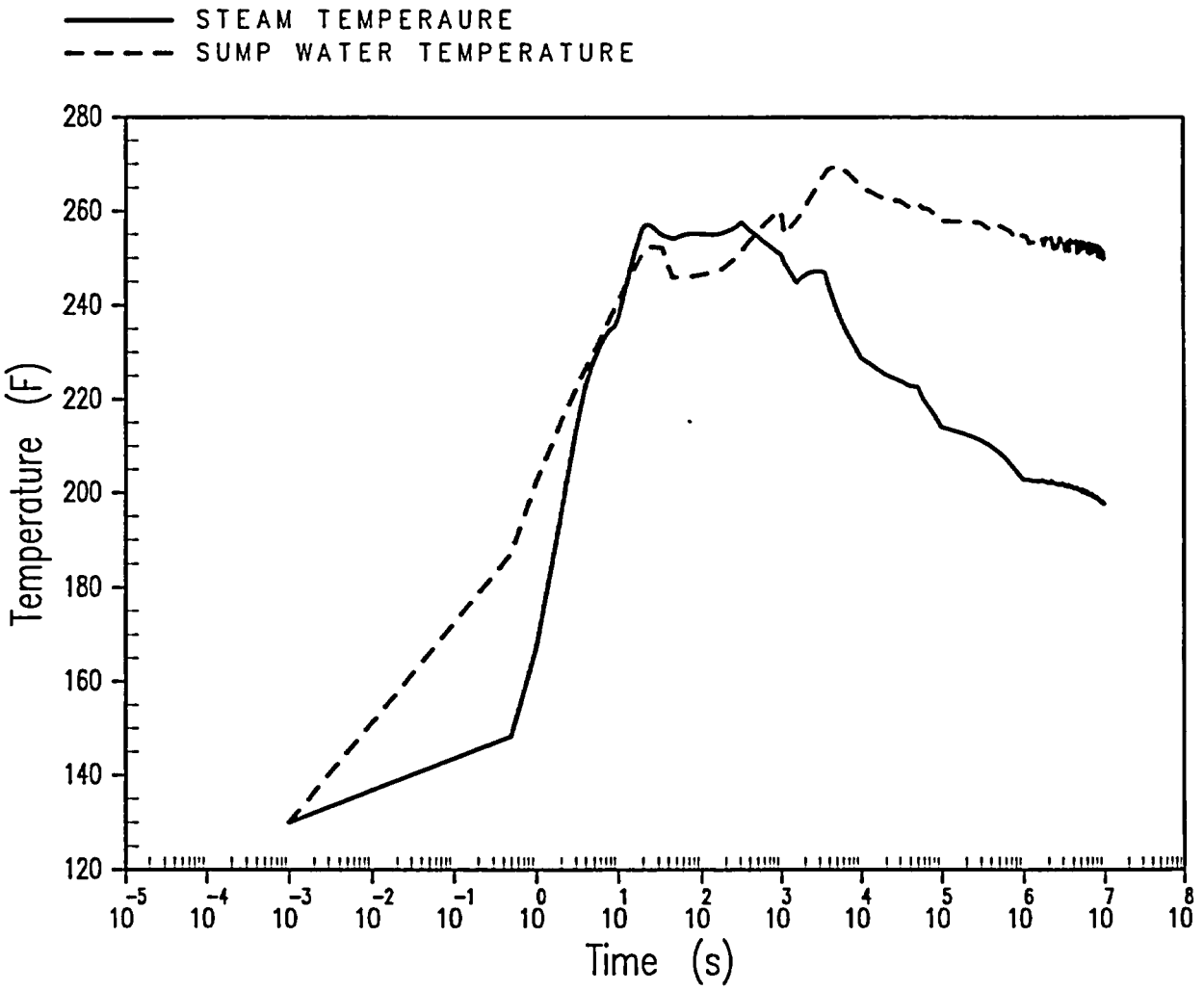
DEPS Break (minimum safeguards) Containment Temperature



Single failure is 1 spray pump and 4 RCFCs run.

Figure 6.5-3

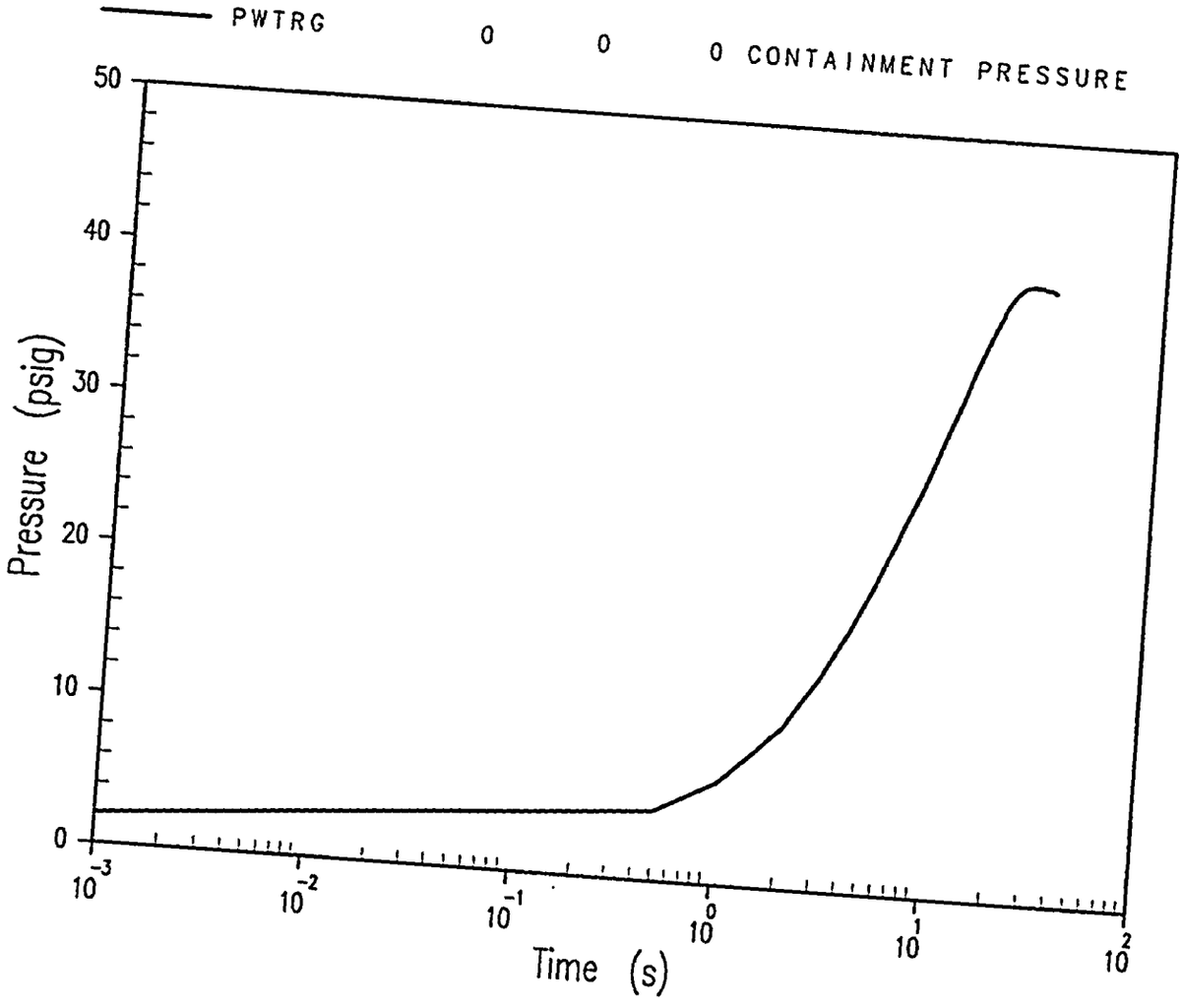
DEPS Break (maximum safeguards) Containment Pressure



Single failure is 1 spray pump and 4 RCFCs run.

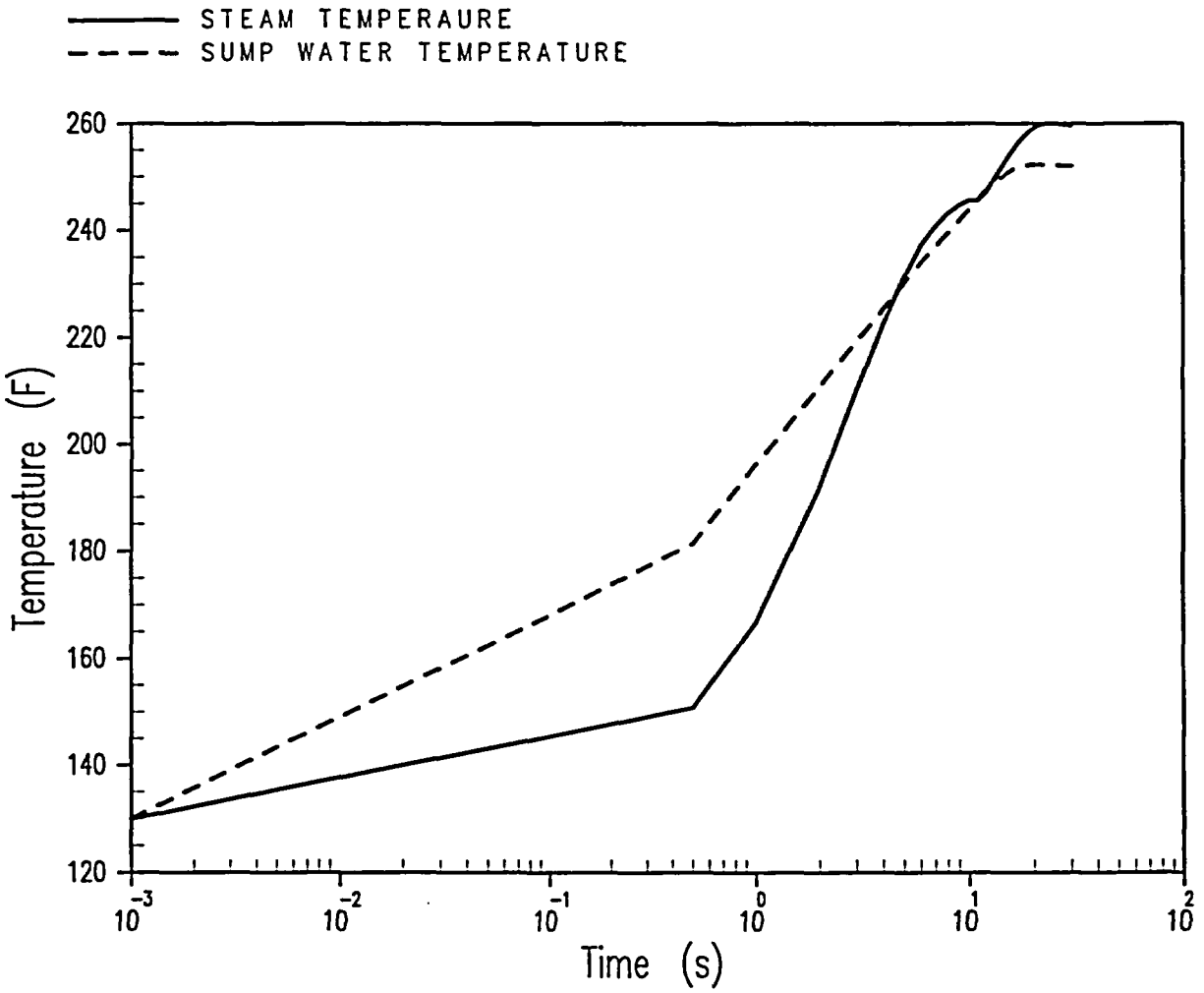
Figure 6.5-4

DEPS Break (maximum safeguards) Containment Temperature



DEHL break is blowdown-limited, thus no containment ESF is assumed.

Figure 6.5-5
DEHL (blowdown limited) Containment Pressure



DEHL break is blowdown-limited, thus no containment ESF is assumed.

Figure 6.5-6

DEHL Break (blowdown safeguards) Containment Temperature

6.6 Main Steamline Break Mass and Energy Releases

6.6.1 Main Steamline Break M&E Releases inside Containment Responses

6.6.1.1 Introduction

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the plant operating conditions and the size of the rupture as well as the configuration of the plant steam system and containment design. The analysis considers a postulated pipe break with limiting consequences, thereby 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.

6.6.1.2 Input Parameters and Assumptions

To assess the effect of break area on the M&E releases from a ruptured steamline, the limiting rupture of the main steamline has been evaluated. This analysis returns to the assumption of a 2-percent power uncertainty by assuming 102 percent of Nuclear Steam Supply System (NSSS) power as the initiating condition for the MSLB event. At a plant power level of 102-percent nominal full-load power, a full double-ended rupture (DER) has been analyzed based on the results of the analyses presented in Section 6.2.1.4 of the *IP2 Updated Final Safety Analysis Report* (UFSAR) (Reference 1).

The DER is postulated in one steamline downstream of the steam generator flow restrictor. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. The effective break area for IP2 (with Westinghouse Model 44F steam generators) is 1.4 ft^2 because the flow from the steam generator is limited by the steam generator flow restrictor.

The important plant conditions and features that were assumed for the stretch power uprate (SPU) analysis case are discussed in the following paragraphs.

Initial Power Level

This analysis returns to the assumption of a 2-percent power uncertainty by assuming 102 percent of NSSS power as the initiating condition for the MSLB event. Full-power conditions have been investigated for IP2 as presented in the UFSAR (Reference 1).

NSSS power is used in this analysis since the reactor coolant pumps (RCPs) continue to run during the event. Net heat addition is conservatively modeled at 20 MWt (See Table 6.6-1). For the uprating analysis, the approach demonstrated that the containment response at uprated power conditions does not exceed the containment pressure limit delineated in the technical specifications.

Initial Plant Conditions

In general, plant initial conditions are assumed to be at their nominal values corresponding to the initial power for that case, with appropriate uncertainties included. Tables 6.6-1 and 6.6-2 identify the values assumed for Reactor Coolant System (RCS) pressure, RCS vessel average temperature, pressurizer water volume, steam generator water level, and feedwater enthalpy at 102-percent uprated power. Steamline break M&E 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.

Single-Failure Assumptions

The analyzed case considered a single failure. That single failure is the failure of the feedwater control valve (FCV) in the faulted loop. If the FCV in the feedwater line to the faulted steam generator is assumed to fail in the open position, the unisolatable volume of feedwater piping is increased. The fluid inventory in this additional unisolatable feedwater piping is available to flash and be released to the containment as the piping depressurizes.

Main Feedwater System

The rapid depressurization that occurs following a steamline rupture typically results in large amounts of water being added to the faulted steam generator through the Main Feedwater System. Rapid-closing feedwater isolation valves (FIV) in the main feedwater lines can limit this effect.

Following initiation of the MSLB, main feedwater flow is conservatively modeled as increasing in response to the decreasing steam pressure. This maximizes the total mass addition prior to feedwater isolation. The feedwater isolation response time, following the safety injection (SI) signal, was assumed to be a total of 60 seconds.

Following feedwater isolation, as the steam generator pressure decreases, the fluid in the feedwater lines downstream from the isolation valve may flash to steam if the feedwater temperature is at or exceeds the saturation temperature associated with the pressure of the feedwater piping. This unisolatable feedwater line volume is an additional source of

high-energy fluid that was assumed to be discharged from the break. The unisolatable volume in the faulted loop feedwater line is conservatively maximized.

Auxiliary Feedwater System

Addition of auxiliary feedwater (AFW) to the steam generators will increase the secondary mass available for release to containment and increase the heat transferred to the secondary fluid. Within the first minute following a steamline break, the Auxiliary Feedwater System (AFWS) is initiated on any one of several protection system signals. The AFW flow to the faulted and intact steam generators has been assumed to be a constant value, based on maximum AFW pump performance. A higher AFW flowrate to the faulted loop steam generator is assumed, consistent with a depressurizing steam generator. Conversely, a lower AFW flowrate to the intact steam generators is assumed, consistent with the intact-loop steam generators remaining at a pressurized condition.

Steam Generator Secondary Side Fluid Mass

A maximum initial steam generator mass in the faulted-loop steam generator was used in the analyzed case. The use of a high faulted-loop initial steam generator mass maximizes the steam generator inventory available for release to containment. The initial mass was calculated as the value corresponding to the programmed level +10-percent narrow-range span (NRS) and assuming 0-percent steam generator tube plugging (SGTP), plus an uncertainty on steam generator water mass.

Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, the steam generators in the intact loops become sources of energy, which can be transferred to the steam generator with the broken line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes drops 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 for transfer to the steam generator with the broken steamline. The effects of reverse steam generator heat transfer are included in the results.

Break Flow Model

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

Steamline Volume Blowdown

The contribution to the M&E releases from the secondary plant steam piping was included in the M&E release calculations. For the analyzed case, the steamline check valves were credited to prevent break flow from the intact steam generators. Therefore, the mass and energy available for release from the secondary plant steam piping is limited to that contained in the volume between the faulted steam generator and the steamline check valve. The flowrate was determined using the Moody critical mass flow model.

Main Steamline Isolation

Steamline isolation is not considered, as the steamline check valve in the faulted loop is credited to prevent blowdown from the three intact steam generators.

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. (Subsequent analysis of the containment response to the MSLB models the operation of the emergency fan coolers and containment spray.) The protection system actuation signals and associated setpoints that were modeled in the analysis are identified in Table 6.6-3. The setpoints used are conservative with respect to the IP2 plant-specific values presented in the Technical Specifications (Reference 2).

For the DER MSLB for IP2 at 102-percent power, the first protection system signal actuated is high differential steamline pressure between two loops 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.

Safety Injection System

Minimum Emergency Core Cooling System (ECCS) flowrates corresponding to the failure of one ECCS train were assumed in this analysis. A minimum ECCS flow is conservative since the reduced boron addition maximizes a return to power resulting from RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to start ECCS pumps was assumed to be 16 seconds for this analysis with offsite power available. A coincident loss of offsite power (LOOP) is not assumed for the analysis since the assumed LOOP would reduce the M&E releases. This is due to the loss of forced reactor coolant flow, which results in a consequential reduction in primary-to-secondary heat transfer.

RCS 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, and the RCPs. 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 does not have a major effect on the calculated M&E releases. The effects of this RCS metal heat are included in the results using conservative thick metal masses and heat transfer coefficients.

Core Decay Heat

Core decay heat generation assumed in calculating the steamline break M&E releases was based on 1979 American National Standard (ANS) Decay Heat with 2σ uncertainty (Reference 3).

Rod Control

The Rod Control System was conservatively assumed to be in manual operation for all steamline break analyses. Rods in automatic control would step into the core prior to reactor trip, due to the increased steam flow. This would reduce nuclear power and core heat flux, reducing the primary-to-secondary heat transfer.

Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions were 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.

6.6.1.3 Description of Analysis

The break flows and enthalpies of the steam release through the steamline break inside containment are analyzed with the LOFTRAN computer code (Reference 4). Blowdown M&E releases determined using LOFTRAN include the effects of core power generation, main and AFW additions, engineered safeguards systems, RCS thick metal heat storage, and reverse steam generator heat transfer.

The IP2 NSSS is analyzed using LOFTRAN to determine the transient steam M&E releases inside containment following a steamline break event. The M&E releases are used as input conditions to the analysis of the containment response.

The licensing-basis case of the MSLB inside containment that has been analyzed at the uprated power is the full double-ended rupture at 102-percent power, with the FCV in the faulted loop assumed to be failed open. Selection of this case was based on the results of the analyses presented in the IP2 UFSAR, Section 6.2.1.4 (Reference 1).

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 for IP2 (44F steam generators). Reverse flow from the three intact steam generators is prevented by the steamline check valve located downstream of the break site.

6.6.1.4 Acceptance Criteria

The MSLB is classified as an ANS Condition IV event, an infrequent fault. Additional clarification of the ANS classification of this event is presented in subsection 6.3.11 of this report, which discusses the core response to a steamline break event. The acceptance criterion associated with the steamline break event resulting in a M&E release inside containment is not based on the M&E analysis itself. It is based on an analysis for containment response which provides sufficient conservatism to ensure that the containment design margin is maintained. The containment response analysis is discussed in subsection 6.6.2.

The specific criterion applicable to this analysis is related to the assumptions regarding power level, stored energy, the break flow model including entrainment, 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 M&E release analysis as discussed in Reference 5 and subsection 6.6.1.2 of this report.

The M&E release data for each of the MSLB cases was used as input to a containment response calculation to confirm the design parameters of the IP2 containment structure.

6.6.1.5 Results

Using Reference 1 as a basis, including parameter changes associated with the SPU, the M&E release rates for the MSLB case noted in subsection 6.6.1.3 were developed for use in containment pressure and temperature response analysis. The containment pressure response was, in turn, used for evaluation of containment integrity. Table 6.6-4 provides the sequence of events for IP2, for the large double-ended rupture at 102-percent power with feedwater control valve failure assumed.

6.6.1.6 Conclusions

The M&E releases from the MSLB case have been analyzed at the SPU power conditions. The assumptions discussed in subsection 6.6.1.2 have been included in the MSLB analysis such that the applicable acceptance criteria are met. The M&E releases discussed in this section have been provided for use in the containment response analysis (see subsection 6.6.2) in support of the IP2 SPU.

6.6.2 Steamline Break Containment Response Evaluation

6.6.2.1 Introduction

The IP2 containment systems are designed such that for all steamline break sizes, up to and including the double-ended severance of a steamline, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical steamline break. The containment response analysis uses the long-term M&E release data from subsection 6.6.1.5.

6.6.2.2 Input Parameters and Assumptions

The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis as shown in Table 6.6-5.

Also, values for the refueling water storage tank (RWST) temperature have been specified, along with containment spray (CS) pump flowrate and reactor containment fan cooler (RCFC) heat removal performance. These values are chosen conservatively, as shown in Table 6.6-5 and 6.6-6. The heat sink modeling is specified in Table 6.6-7 and 6.6-8, and is consistent with the values used for the LOCA containment response analysis, as documented in Section 6.5.

Subsection 6.6.1.5 discusses the M&E releases for the SPU MSLB case. The M&E release analysis includes the single failure of the faulted-loop feedwater control valve, as discussed in subsection 6.6.1.2. Since a single failure is included in the M&E release analysis, no single failure is modeled in the containment response analysis.

6.6.2.3 Description of Analysis

Calculation of containment pressure and temperature is accomplished by use of the computer code COCO (Reference 6). COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for a particular containment design. The values used in the specific model for different aspects of the

containment are derived from plant-specific input data. The COCO code has been used and found acceptable to calculate containment pressure and temperature transients for previous IP2 containment response analyses.

6.6.2.4 Acceptance Criteria

The design basis MSLB is an ANS Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the IP2 UFSAR, Revision 18 (Reference 7) for long-term containment response, the relevant General Design Criteria (GDC) (Reference 8) requirements are listed below.

GDC 16, Containment Design

To satisfy the requirements of GDC 16, the peak calculated containment pressure must be less than the containment design pressure of 47 psig for IP2.

GDC 38, Containment Heat Removal

To satisfy the requirement of GDC 38, the calculated pressure at 24 hours must be less than 50 percent of the peak calculated value.

6.6.2.5 Analysis Results

The peak containment pressure is listed in Table 6.6-9 the for the uprated full-power case with offsite power available.

6.6.2.6 Conclusions

An evaluation of the MSLB containment pressure response has been performed as part of the IP2 SPU Program. The analysis included the long-term pressure profile for the limiting case. The analyzed case results in a peak containment pressure that is less than the containment design pressure of 47 psig. The long-term pressures are well below 50 percent of the peak value within 24 hours. Based on these results, the GDC criteria for IP2 have been met.

6.6.3 Main Steamline Break M&E Releases outside Containment Responses

6.6.3.1 Introduction

Main steamline breaks (MSLBs) outside the Containment Building were considered for the IP2 SPU Program to define conditions for equipment qualification (EQ) for electrical equipment that is needed to mitigate the consequences of high-energy line breaks and is located near the steam and feed penetration area.

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 that is released. The effect of the steam release depends on the plant configuration at the time of the break, plant response to the break, and the size and location of the break. Because of the interrelationship among many of the factors that influence steamline break M&E releases, an appropriate determination of a single limiting case with respect to M&E releases cannot be made. Therefore, it was necessary to analyze the steamline break event outside containment for a range of conditions. The resulting M&E releases were used as input to the Auxiliary Building temperature analysis (see subsection 6.5.2.7) for equipment environmental qualification (see subsection 10.9.3).

6.6.3.2 Input Parameters and Assumptions

To determine the effects of NSSS power level and break area on M&E releases from a ruptured steamline, spectra of both variables were evaluated as part of the methodology development program documented in WCAP-10961 (Reference 9). At 102 and 70 percent of NSSS power levels, various break sizes were analyzed, ranging from 0.1 ft^2 to 4.6 ft^2.

A full-break spectrum at both power levels (102 and 70 percent) has been analyzed at the SPU conditions for IP2. Other assumptions regarding important plant conditions and features are discussed in the following paragraphs.

Initial Power Level

The initial power assumed for steamline break analyses outside containment affects the M&E 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 at lower power levels will cause 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.

Therefore, the following power levels were analyzed:

- Full power - maximum NSSS power (3236 MWt based on 3216 MWt plus 20 MWt for RCP heat addition) plus uncertainty, that is, 102 percent of rated NSSS power

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

For this IP2 SPU analysis, the power levels and steamline break sizes are noted in subsection 6.6.3.3 of this report.

In general, plant initial conditions were assumed to be the nominal values corresponding to the initial power for that case, with appropriate uncertainties included. Table 6.6-10 lists nominal 100-percent power NSSS conditions. Table 6.6-11 lists initial plant condition assumptions for the cases 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, the calculated values of the superheated steam enthalpy available for release outside containment are larger than at the low end. The thermal design flowrate has been used for the RCS flow input. This is consistent with the assumptions documented in Reference 5 and with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties and RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the SPU Program have been applied only to RCS average temperature (7.5°F), steam generator mass (10-percent NRS), and power fraction (2 percent) at full power. Nominal values are adequate for the initial pressurizer pressure and water level. Uncertainty conditions were only applied to those parameters that could increase the enthalpy of superheated steam discharged from the break.

Single-Failure Assumptions

The steamline break analyses outside containment were designed to encompass two separate single failures consistent with the current licensing basis for IP2.

The first single failure is one AFW pump resulting in minimum AFW flow to the steam generators. Variations in AFW flow can affect steamline break M&E releases in a number of ways including break mass flowrate, RCS temperature, tube bundle uncovery time and steam superheating. The minimum AFW flow used in the analysis was conservatively based on only one motor-driven AFW pump.

The second single failure is the main steamline isolation valve (MSIV) in the loop with the faulted steamline. This permits blowdown of the entire mass inventory of the steam generator in the loop with the faulted steamline. This single failure was limited to the steamline with the postulated break.

Main Feedwater System

The rapid depressurization that 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 increased steam flow following the steamline break. This minimizes total mass addition and the associated cooling effects in the steam generators, which causes the earliest onset of superheated steam released from the break.

Isolation of main feedwater flow was conservatively assumed to be coincident with reactor trip, irrespective of the function that produced the trip signal. This assumption reduces the total mass addition to the steam generators. The main feedwater flow isolation valves were assumed to close instantaneously with no consideration of associated signal processing or valve stroke time.

Auxiliary Feedwater System

Within the first few minutes following a steamline break, AFW is initiated on one of several protection system signals. Addition of AFW to the steam generators will increase the secondary mass available to cover the tube bundle and reduce the amount of superheated steam produced. For this reason, AFW flow is minimized while the actuation delay is maximized to accentuate depletion of the initial secondary side inventory.

The volume of the AFW piping up to the isolation valve closest to the steam generator was maximized and purging of the AFW piping was assumed. This maximizes the amount of preheated water resident in the AFW piping and ensures that this preheated water was injected into the steam generator first. The less dense resident AFW decreases initial mass addition to the faulted-loop steam generator. The large volume also delays the introduction of colder AFW

into any steam generator, which reduces the cooldown effect on the primary side of the RCS. AFW assumptions used in the analysis are presented in Table 6.6-12.

Steam Generator Fluid Mass

A minimum initial fluid mass in all steam generators has been used in each of the analyzed cases. This minimizes the capability of the heat sink afforded by the steam generators and leads to earlier tube bundle uncovery. The initial mass has been calculated as that corresponding to the programmed water level, minus 10-percent NRS, minus a mass uncertainty. All steam generator fluid masses were calculated assuming 0-percent SGTP. This assumption is conservative with respect to the RCS cooldown through the steam generators resulting from the steamline break.

Steam Generator Reverse Heat Transfer

Once 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 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 were included in the results.

Break Flow Model

The flowrate from the break is maximized by assuming a critical flowrate for saturated steam based on the Moody correlation for fL/D=0 (Reference 5). The upstream pressure is based on the steam generator pressure, with no credit for line losses or piping discharge resistance. The downstream pressure is assumed to be atmospheric throughout the blowdown.

Steamline Volume Blowdown

There is no contribution to M&E releases from the steam in the secondary plant loop piping and header because the initial volume is saturated steam. With the focus of the MSLB analysis outside containment on maximizing superheated steam enthalpy, it is presumed that the saturated steam in the loop piping and header has no adverse effects on the results. The blowdown of steam in this volume serves to delay the time of tube uncovery in the steam generators and is conservatively ignored.

Main Steamline Isolation

Steamline isolation was assumed to terminate blowdown from the intact-loop steam generators for the header break cases. For the loop break cases, the faulted-loop steamline check valve was assumed to prevent blowdown from the three intact steam generators. The main steamline isolation function was accomplished via the MSIV in each of the four steamlines. The MSIV actuation signal was received from a dynamically compensated low steamline pressure signal. A delay time of 9 seconds, accounting for delays associated with signal processing plus MSIV stroke time has been assumed. Unrestricted steam flow through the valve during valve stroke has been assumed.

Protection System Actuations

The protection systems available to mitigate the effects of a MSLB outside containment include reactor trip, safety injection, steamline isolation, and AFW injection. The protection system actuation signals and associated setpoints that were modeled in the analysis are identified in Table 6.6-13. The setpoints are conservative values with respect to the plant-specific values delineated in the IP2 Technical Specifications.

Tables 6.6-14 through 6.6-17 provide the protection system actuation times for the various steamline break sizes for IP2, at 102- and 70-percent NSSS power.

In all cases, the turbine stop valve was assumed to close instantly following the reactor trip signal.

Safety Injection System

Minimum Emergency Core Cooling System (ECCS) flowrates corresponding to failure of one ECCS train have been assumed in this analysis. Minimum ECCS flow is conservative since the reduced boron addition maximizes a return to power resulting from RCS cooldown. The return to power increases heat transfer to the secondary side, maximizing steam flow from the break. The delay time to achieve full safety injection flow was assumed to be 16 seconds for this analysis with offsite power available. A coincident LOOP was not assumed for the analysis since the M&E releases would be reduced due to loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

RCS Metal Heat Capacity

As the primary side of the plant cools, the reactor coolant temperature drops below that of the reactor coolant piping, reactor vessel, RCPs, and 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 does not have a major effect on the calculated M&E releases, but the effects were included in the results using conservative thick-metal masses and heat transfer coefficients.

Core Decay Heat

Core decay heat generation assumed in calculating steamline break M&E releases was based on the 1979 ANS Decay Heat + 2σ model (Reference 3).

Rod Control

The Rod Control System was conservatively assumed to be in manual operation for all steamline break analyses. Rods in automatic control would step in prior to reactor trip due to the increase in steam flow, reducing nuclear power and core heat flux. However, sensitivity analyses performed when WCAP-10961 (Reference 9) was written, investigating the effects on steamline break M&E releases of manual versus automatic rod control, have shown negligible effect on calculated results.

Core Reactivity Coefficients

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

6.6.3.3 Description of Analysis

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 M&E releases determined using LOFTRAN include the effects of core power generation, main feedwater and auxiliary feedwater additions, engineered safeguards systems, RCS thick-metal heat storage, and reverse steam generator heat transfer. The use of the LOFTRAN code for analysis of the MSLB with superheated steam M&E releases is documented in Supplement 1 of WCAP-8822 (Reference 5), which has been reviewed and approved by the NRC for use in analyzing MSLBs, and in WCAP-10961 (Reference 9) for MSLBs outside containment.

The IP2 NSSS has been analyzed to determine the transient mass releases and associated superheated steam enthalpy values outside containment following a steamline break event.

The resulting tables of mass flowrates and steam enthalpies were used as input conditions to the calculation of outside-containment compartment conditions (see subsection 6.6.4) for the environmental evaluation of safety-related electrical equipment.

The following licensing-basis cases of the MSLB outside containment were analyzed at the noted conditions for the SPU Program.

- At 102-percent power, break sizes of 4.6, 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 4.6, 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 was represented as a non-mechanistic split rupture (crack area). The largest break was postulated as a crack area equivalent to a single-ended pipe rupture. The break flowrate was limited by the total cross-sectional flow area of the steam pipe; the maximum break size was thus limited to 4.6 ft^2. Prior to steamline isolation, the break area was represented by the spectrum noted above. After steamline isolation, the break area was limited by the area of the integral steam generator flow restrictor.

6.6.3.4 Acceptance Criteria

The MSLB is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steamline break event resulting in a M&E release outside containment are based on an analysis that provides sufficient conservatism to ensure that the equipment remains qualified for the temperature and pressure profiles from the compartment analyses. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, 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 assumptions have been included in this steamline break M&E release analysis as discussed in subsection 6.6.3.2 of this report. The tables of mass flowrates and steam enthalpy values for each of the steamline break cases analyzed were used as input to calculation of outside-containment compartment conditions (see subsection 6.6.4) for the environmental evaluation of safety-related electrical equipment.

6.6.3.5 Results

Using the MSLB analysis methodology documented in WCAP-10961 (Reference 9) as a basis, including parameter changes associated with the SPU, the M&E release rates for each

steamline break case have been developed for use in calculating outside-containment compartment conditions (see subsection 6.6.4) for the environmental evaluation of safety-related electrical equipment. Tables 6.6-14 through 6.6-17 provide the sequences of events for the various steamline break sizes for IP2, at 102- and 70-percent NSSS power.

6.6.3.6 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 IP2 SPU Program. The assumptions discussed in subsection 6.6.3.2 have been included in the analysis such that conservative M&E releases were calculated. The resulting mass releases and associated steam enthalpy values have been provided for use in the calculation of outside-containment compartment conditions (see subsection 6.6.4) for the environmental evaluation of safety-related electrical equipment outside containment in support of the IP2 SPU Program.

6.6.4 Main Steamline Break outside Containment Compartment Response

6.6.4.1 Introduction

This section of the report presents the results of a study to determine the effects of superheated steam releases, during postulated main steamline ruptures, on outside containment equipment qualification for IP2. For this study, the compartment temperature profiles for the steam and feed penetration area were calculated as required by 10CFR50.49 (Reference 10).

NRC IE Information Notice 84-90, *Main Steam Line Break Effect on Environmental Qualification of Equipment*, (Reference 11) informed licensees of potential issues related to the release of superheated steam following a postulated MSLB. Specifically, such superheated blowdowns have the potential to raise the compartment temperatures, and, therefore, the equipment surface and internal temperatures, above those originally used for the environmental qualification of such equipment needed to mitigate the consequences of high energy line breaks.

The report describes the methods and assumptions used in modeling the IP2 compartments in the steam and feed penetration area. The M&E releases from the postulated MSLBs were discussed in subsections 6.6.3.1 through 6.6.3.6. The results from these calculated compartment temperature profiles are discussed here.

6.6.4.2 Input Parameters and Assumptions

This study used MSLB M&E releases (see subsections 6.6.3.1 through 6.6.3.6) in calculations of the outside containment compartment temperatures resulting from those releases. The RCS conditions used for determining the steamline break M&E releases were described in subsection 6.6.3.2. The compartment model was developed for the GOTHIC code (Reference 12) from engineering drawings and plant information.

Some ventilation louvers in the steam and feed penetration area are closed and covered during winter conditions. Because of this difference and because of the different temperature and humidity conditions for winter and summer, cases were divided into winter and summer conditions. Table 6.6-18 provides the GOTHIC initial conditions for the winter and summer cases.

6.6.4.3 Description of Analysis

This analysis of the temperature and pressure response in the steam and feed penetration area was performed with the GOTHIC code. The M&E releases for loop breaks at 102- and 70-percent power and header breaks at 102- and 70-percent power were provided by the analysis discussed in subsections 6.6.3.1 through 6.6.3.6 at the SPU conditions. The compartment response was determined for two sets of initial temperature conditions, winter and summer. In addition to the initial temperature and relative humidity, the cases for the winter and summer conditions also modeled several louvers differently. The louvers were modeled as closed for both the winter and summer conditions, but for the winter, Entergy covers the louvers in order to reduce the possibility of freezing in the compartment. The compartment response was calculated for a period of 100,000 seconds. This duration was sufficient for the compartment temperatures to decrease to ambient conditions.

6.6.4.4 Acceptance Criteria

The acceptance criteria for the outside containment compartment temperature evaluation was defined for the equipment qualification program at IP2 as the qualification limits for each piece of equipment because each piece of equipment has its own qualification conditions. The peak temperature in the compartment and the duration at elevated temperatures are of interest for the EQ program. Refer to subsection 10.9.3 for the equipment qualification discussion.

6.6.4.5 Results

The M&E releases for IP2 were provided by the analysis in subsections 6.6.3.1 through 6.6.3.6 at the SPU conditions. These cases were analyzed for IP2 at initial conditions for winter and initial conditions for summer.

Since these results were used for equipment qualification, the temperature and pressure profiles for each case were provided for the EQ evaluations. In some cases, pressure and temperature composite profiles were also provided. All of the temperature transients returned to pre-accident temperatures within 40,000 seconds (~11 hours).

The computer simulations performed for the M&E release analysis were originally run assuming an operator action time of 900 seconds to terminate auxiliary feedwater flow and close the MSIVs. The operator action time of 900 seconds was later determined to be excessively conservative and a time of 600 seconds was determined to be conservative, but more realistic.

For the subsequent compartment temperature analysis in the steamline and feedline penetration area, each of the break cases were assessed for the inclusion of the 600-second operator action time. To accomplish this, the following approach was taken:

- The M&E release transients were examined to determine when the start of steady-state steaming of auxiliary feedwater flow occurred.

- For the cases in which steady-state conditions prevailed between 600 and 900 seconds, the time of steady-state M&E releases was shortened by 300 seconds. This method captures the M&E release transient including steam generator blowdown, plant system actuations, generation of superheated steam, and steaming of auxiliary feedwater flow. The result was a conservative representation of the M&E releases when a 600-second operator action time was assumed.

- For the cases in which steady-state conditions were not reached by 600 seconds, the M&E releases were not modified.

6.6.4.6 Conclusions

Since these results were used for equipment qualification, the temperature and pressure profiles for each case were provided for the EQ evaluations. In some cases, composites of temperature cases were also provided.

Composite temperature and pressure profiles for the 600-second operator action time are provided in Figures 6.6-2 and 6.6-3 for the header and loop breaks for winter and summer conditions. Section 10.9.3 uses this information and the individual case profiles to address the qualification of the equipment for IP2 at the SPU conditions.

6.6.5 Steam Releases for Radiological Dose Analysis

The vented steam releases have been calculated for the locked rotor and steamline break events. Table 6.6-19 summarizes the vented steam releases from the operable steam generators as well as auxiliary feedwater flows for the 0- to 2-hour time period, the 2- to 8-hour time period, and the 8- to 40-hour time period for each of these events.

No explicit assumption is considered in these analyses regarding Steam Generator Blowdown System isolation. The implied assumption is that the entire inventory of the steam generators is released to the environment and no loss of inventory through the blowdown line is considered. This provides a conservative calculation of the quantity of steam vented during the noted time periods.

The steam releases discussed in this section have been provided as inputs to the radiological dose analyses (See subsection 6.11.9) in support of the IP2 SPU Program.

6.6.6 References

1. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report*, Docket No. 50-247.

2. *Indian Point Unit 2 Technical Specifications, Amendment 235*, February 6, 2003.

3. ANSI/ANS-5.1-1979, *American National Standard for Decay Heat Power in Light Water Reactors*, The American Nuclear Society Standards Institute, Inc., LaGrange Park, Illinois, August 1979.

4. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), *LOFTRAN Code Description*, T. W. T. Burnett, et al., April 1984.

5. 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 (Non-Proprietary), *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.

6. WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), *Containment Pressure Analysis Code (COCO), July 1974.*

7. Indian Point Power Station Unit 2, *Updated Final Safety Analysis Report*, Rev. 18, June 2003.

8. 10CFR50 Appendix A, *General Design Criteria for Nuclear Power Plants.*

9. WCAP-10961 *(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*, Rev. 1, October 1985.

10. 10CFR50.49, *Environmental Qualification Of Electric Equipment Important To Safety For Nuclear Power Plants*, 66 FR 64738, December 14, 2001.

11. NRC IE Information Notice 84-90, *Main Steam Line Break Effect on Environmental Qualification of Equipment*, December 07, 1984.

12. NAI 8907-02, *GOTHIC Containment Analysis Package User Manual*, Version 7.0, Rev. 13, July 2001.

Table 6.6-1 Nominal Plant Parameters for IP2 SPU[1] (MSLB M&E Releases)	
Nominal Conditions	**IP2**
NSSS Power, MWt	3236
Core Power, MWt	3216
Net Heat Addition, MWt	20
Reactor Coolant Flow (total), gpm	322,800
Pressurizer Pressure, psia	2250
Core Bypass, %	6.5
Reactor Coolant Temperatures, °F	
Core Outlet	610.0
Vessel Outlet	605.8
Core Average	575.9
Vessel Average	572.0
Vessel/Core Inlet	538.2
Steam Generator	
Steam Temperature, °F	516.4
Steam Pressure, psia	788
Steam Flow (total), 10^6 lbm/hr	14.08
Feedwater Temperature, °F	436.2

Note:

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

Table 6.6-2 IP2 Initial Condition Assumptions for SPU[1]	
MSLB M&E Releases Inside Containment	
Parameter	**Value**
NSSS Power (% Nominal Uprated)	102
RCS Average Temperature (°F)	579.5
RCS Flowrate (gpm)	322,800
RCS Pressure (psia)	2250
Pressurizer Water Volume (ft^3)	1164.8
Feedwater Enthalpy (Btu/lbm)	415.1
SG Water Level (% span)	62

Note:

1. Noted values correspond to plant conditions defined for 0% steam generator tube plugging and the high end of the RCS T_{avg} window; the temperature includes the applicable calorimetric uncertainties.

Table 6.6-3 Protection System Actuation Signals and Safety System Setpoints for IP2 SPU Analysis	
MSLB M&E Releases Inside Containment	
Safety Injection High Differential Steamline Pressure: 270 psig	Conservatively high value used Safety Injection signal results in reactor trip, feedwater isolation, and actuation of the reactor containment fan coolers
Containment Sprays High-High Containment Pressure: 30 psig	Conservatively high value used

Table 6.6-4 1.4 ft^2 MSLB Hot Full Power With FCV Failure Assumed Sequence of Events for IP2 SPU	
Time (sec)	**Event Description**
0.0	MSLB occurs
1.0	SI setpoint reached on high steamline differential pressure (270 psi)
3.0	Rod motion starts (high steamline differential pressure actuates SI, which initiates reactor Trip) Intact loop FCVs close
8.0	Main feedwater pumps trip
18.0	MFW pumps stopped; continued flow from condensate pumps
63.0	BFD-2 feedwater pump discharge valve closes (following SI signal)
128.0	BFD-5 feedwater block valve closes (following SI signal)

Table 6.6-5 MSLB Containment Response Analysis Initial Containment Conditions and Parameters	
RWST water temperature (°F)	110
Initial containment temperature (°F)	130
Initial containment pressure (psia) Maximum	16.7
Initial relative humidity (%)	20
Net free volume (ft^3)	2.61×10^6
Number of Containment Air Recirculation Fan Coolers	5
Start of Containment Fan Coolers (sec)	61
Number of Containment Spray Pumps	2
Containment Spray Flowrate, total (gpm)	4200
Start of Containment Spray (sec)	206.1

Table 6.6-6 Reactor Containment Fan Cooler Performance	
Containment Temperature (°F)	Heat Removal Rate [Btu/sec] Per RCFC
130	2314
150	3742
170	5197
190	6663
210	8129
230	11928
271	19084

Table 6.6-7 Containment Heat Sinks			
No.	Material	Thickness (ft)	Surface Area (ft^2)
1	Carbon Steel Concrete	0.03125 4.5	41530
2	Carbon Steel Concrete	0.0417 3.5	26012
3	Concrete	1.0	13636
4	Stainless Steel Concrete	0.03125 1.0	9091
5	Concrete	1.0	55454
6	Carbon Steel	0.0417	62538
7	Carbon Steel	0.03125	74276
8	Carbon Steel	0.0208	25407
9	Carbon Steel	0.01563	63454
10	Carbon Steel	0.01042	2727
11	Carbon Steel	0.0115	20000
12	Carbon Steel	0.00521	9090
13	Stainless Steel PVC Insulation Carbon Steel Concrete	0.0016 0.1042 0.0625 4.5	714
14	Stainless Steel PVC Insulation Carbon Steel Concrete	0.0016 0.1042 0.0417 4.5	6226
15	Stainless Steel Foam Insulation Carbon Steel Concrete	0.00208 0.125 0.0417 4.5	3469
16	Stainless Steel Foam Insulation Carbon Steel Concrete	0.00208 0.125 0.03125 4.5	3965
Coatings	Top (Paint) Bottom (Carbozinc)	0.00033 0.000258	Equal to carbon steel surface area

Table 6.6-8 Thermophysical Properties of Containment Heat Sinks		
Material	Thermal Conductivity (Btu/hr-ft - °F)	Volumetric Heat Capacity (Btu/ft^3 - °F)
Paint	0.09	28.8
Carbozinc	0.9	28.8
Carbon Steel	26.0	56.35
Stainless Steel	8.6	56.35
Concrete	0.8	28.8
PVC Insulation	0.0208	1.20
Foam Insulation	0.0417	1.53

Table 6.6-9 MSLB Peak Containment Pressure for IP2		
Break	**Single Failure**	**Peak Pressure @ Time (sec)**
Full DER	FCV	38.89 psig @ 280.8

Table 6.6-10 Nominal Plant Parameters for SPU[1] (MSLB M&E releases inside and outside containment)	
Nominal Conditions	
NSSS Power, MWt	3236.0[2]
Core Power, MWt	3216.0
Net Heat Addition, MWt	20[2]
Reactor Coolant Flow (total), gpm TDF	322,800
Pressurizer Pressure, psia	2250
Core Bypass, %	6.5
Reactor Coolant Vessel Average Temperature, °F	572.0[1]
Steam Generator	
Steam Temperature, °F	516.4
Steam Pressure, psia	788
Steam Flow, 10^6 lbm/hr (plant total)	14.08
Feedwater Temperature, °F	436.2
Zero-Load Temperature, °F	547

Notes:

1. Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window.
2. A conservative assumption of 20 MWt RCP heat addition was used for MSLB M&E analyses to determine NSSS power.

Table 6.6-11 Initial Condition Assumptions for SPU[1] (MSLB M&E releases outside containment)		
Initial Conditions	**102% Power**	**70% Power**
RCS Average Temperature (°F)	579.5[1]	579.5[1]
RCS Flowrate (gpm TDF)	322,800	322,800
RCS Pressure (psia)	2250	2250
Pressurizer Water Volume (ft^3)	1164.8	1164.8
Feedwater Enthalpy (Btu/lbm)	415.1	370.6
SG Pressure (psia)[2]	836.2	962.6
SG Water Level (% NRS)	42	42

Notes:

1. 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.
2. The noted SG pressures were determined at the steady-state conditions defined by the RCS average temperatures, including applicable uncertainties.

Table 6.6-12 Main and AFWS Assumptions for SPU (MSLB M&E releases outside containment)	
Main Feedwater System	
Flowrate – Both Power Levels (102% and 70%)	nominal flow to all loops
Unisolable Volume from SG Nozzle to MFIV (all loops)	none assumed
AFW	
Single-Failure Assumption, Split Evenly Between Two Steam Generators	380 gpm
Manual Isolation Assumption	600 or 900 seconds[1]
Temperature (maximum value)	120°F
Piping Volume (faulted loop)	268.8 ft^3
Actuation Delay Time	60 seconds

Note:

1. See Section 6.6.4.5 for discussion of use of 600 seconds or 900 seconds.

Table 6.6-13 Protection System Actuation Signals and Safety System Setpoints for SPU (MSLB M&E releases outside containment)
Reactor Trip Low-Low Steam Generator Water Level in any loop – 0% NRS Low Pressurizer Pressure – 1860 psia Power-Range High Neutron Flux – 118% rated thermal power(1) Overtemperature ΔT $K_1 = 1.42$ $K_2 = 0.020$ $K_3 = 0.00070$ Dynamic Compensation lead – 25 seconds lag – 3 seconds Overpower ΔT $K_4 = 1.164$ $K_5 = 0.0$ $K_6 = 0.188$ Dynamic Compensation rate lag – 10 seconds
Safety Injection Low Pressurizer Pressure – 1715 psia Low Steamline Pressure in any Loop – 415 psia[1]
Steamline Isolation Low Steamline Pressure in any Loop – 415 psia[1]
Feedwater Isolation Reactor Trip (conservative assumption)
AFW Initiation Low-Low Steam Generator Water Level in any Loop – 0% NRS Safety Injection

Note:

1. The setpoint values modeled for high neutron flux and low steamline pressure are more conservative than those used for the steamline break core response analysis. The corresponding values for the core response analysis are documented in Table 6.10-1.

Table 6.6-14

IP2 MSLB Outside Containment M&E Release
Time Sequence Summary – 102% Power, Header Break

Break Size (ft^2, Before/ After Steamline Isolation)	Reactor Trip		Safety Injection[1]		MSIV Closure		Auxiliary Feedwater		Time Faulted SG Tubes Uncover (sec)	Time Break Releases Stop (sec)
	Signal	Time Rod Motion Starts (sec)	Signal	Time of Signal (sec)	Signal	Time Fully Closed (sec)	Signal	Time Flow Starts (sec)		
0.1 / 0.1	LSGL	331.0	HDP	1027.9	Manual	900.0	LSGL	389.0	942.6	1255.5
0.2 / 0.2	LSGL	170.6	LPP	525.6	Manual	900.0	LSGL	278.6	468.7[2]	987.7
0.3 / 0.3	LSGL	116.8	LPP	326.2	Manual	900.0	LSGL	174.8	323.5[2]	933.5
0.4 / 0.4	LSGL	89.7	LPP	238.9	Manual	900.0	LSGL	147.7	253.2[2]	921.9
0.5 / 0.5	OPΔT	39.0	LPP	155.1	Manual	900.0	LSGL	121.1	216.0[2]	915.6
0.6 / 0.6	OPΔT	31.2	LPP	125.1	Manual	900.0	LSGL	111.3	185.2[2]	911.7
0.7 / 0.7	OPΔT	27.4	LPP	105.7	Manual	900.0	LSGL	104.8	162.8[2]	909.3
0.8 / 0.8	OPΔT	25.0	LPP	91.6	Manual	900.0	LSGL	100.1	146.4[2]	907.4
0.9 / 0.9	OPΔT	23.2	LPP	80.7	Manual	900.0	LSGL	95.9	133.2[2]	906.1
1.0 / 1.0	OPΔT	21.9	LPP	72.6	Manual	900.0	LSGL	93.4	123.2[2]	905.1
1.2 / 1.2	OPΔT	20.0	LPP	60.6	Manual	900.0	LSGL	89.2	107.8[2]	903.9
1.4 / 1.4	OPΔT	18.6	LPP	52.1	Manual	900.0	LSGL	86.0	96.4[2]	903.0
2.0 / 1.4	HNFLX	12.3	LPP	34.6	Manual	900.0	LSGL	79.2	79.4[2]	902.9
4.6 / 1.4	HNFLX	9.3	LPP	20.8	HSF/LT_{avg}	34.51	LSGL	72.7	50.2	902.1

Note:
1. The SI signal is generated, but the RCS pressure remains too high for delivery of SI flow to the RCS.
2. The intact SG tubes also uncover and contribute superheated steam out the break.

Key
LPP ≡ Low pressurizer pressure
HDP ≡ High differential steam pressure
HSF/LT_{avg} ≡ High steam flow + low T_{avg}
LSGL ≡ Low-low steam generator water level
OPΔT ≡ Overpower ΔT
HNFLX ≡ High nuclear flux

Table 6.6-15

IP2 MSLB Outside Containment M&E Release
Time Sequence Summary – 70% Power, Header Break

Break Size (ft^2, Before/ After Steamline Isolation)	Reactor Trip		Safety Injection[1]		MSIV Closure		Auxiliary Feedwater		Time Faulted SG Tubes Uncover (sec)	Time Break Releases Stop (sec)
	Signal	Time Rod Motion Starts (sec)	Signal	Time of Signal (sec)	Signal	Time Fully Closed (sec)	Signal	Time Flow Starts (sec)		
0.1 / 0.1	LSGL	255.4	HDP	1142.5	Manual	900.0	LSGL	313.4	1062.5	1349.7
0.2 / 0.2	LSGL	131.6	LPP	453.8	Manual	900.0	LSGL	189.6	643.7[2]	1042.8
0.3 / 0.3	LSGL	90.0	LPP	284.4	Manual	900.0	LSGL	148.0	409.8[2]	936.4
0.4 / 0.4	LSGL	69.1	LPP	207.6	Manual	900.0	LSGL	127.1	306.6[2]	922.9
0.5 / 0.5	LSGL	56.4	LPP	163.3	Manual	900.0	LSGL	114.4	250.0[2]	916.4
0.6 / 0.6	LSGL	47.9	LPP	134.4	Manual	900.0	LSGL	105.9	212.6[2]	912.4
0.7 / 0.7	LSGL	41.8	LPP	113.9	Manual	900.0	LSGL	99.8	187.0[2]	910.0
0.8 / 0.8	LSGL	37.1	LPP	98.5	Manual	900.0	LSGL	95.1	167.6[2]	908.1
0.9 / 0.9	LSGL	33.5	LPP	86.6	Manual	900.0	LSGL	91.5	152.0[2]	906.9
1.0 / 1.0	LSGL	30.6	LPP	77.1	Manual	900.0	LSGL	88.6	140.8[2]	905.7
1.2 / 1.2	LSGL	26.1	LPP	63.2	Manual	900.0	LSGL	84.2	123.0[2]	904.4
1.4 / 1.4	LSGL	23.0	LPP	53.6	Manual	900.0	LSGL	81.0	110.0[2]	903.6
2.0 / 1.4	LSGL	17.2	LPP	37.2	Manual	900.0	LSGL	75.2	86.4[2]	903.8
4.6 / 1.4	LSGL	10.0	LPP	19.3	HSF/LT_{avg}	33.5	LSGL	68.0	56.4	902.1

Note:
1. The SI signal is generated, but the RCS pressure remains too high for delivery of SI flow to the RCS.
2. The intact SG tubes also uncover and contribute superheated steam out the break.

Key
LPP ≡ Low pressurizer pressure
HDP ≡ High differential steam pressure
HSF/LT_{avg} ≡ High steam flow + low T_{avg}
LSGL ≡ Low-low steam generator water level
OPΔT ≡ Over power ΔT
HNFLX ≡ High nuclear flux

Table 6.6-16

IP2 MSLB Outside Containment M&E Release
Time Sequence Summary – 102% Power, Loop Break

Break Size (ft^2)	Reactor Trip		Safety Injection[1]		MSIV Closure		Auxiliary Feedwater		Time Faulted SG Tubes Uncover (sec)	Time Break Releases Stop (sec)
	Signal	Time Rod Motion Starts (sec)	Signal	Time of Signal (sec)	Signal	Time Fully Closed (sec)	Signal	Time Flow Starts (sec)		
0.1	LSGL	308.2	HDP	493.5	--	--	LSGL	366.2	434.3	1006.7
0.2	LSGL	158.2	HDP	188.6	--	--	LSGL	216.2	238.4	943.0
0.3	LSGL	107.9	HDP	125.4	--	--	LSGL	165.9	172.4	923.1
0.4	LSGL	82.6	HDP	95.2	--	--	LSGL	140.6	140.8	914.5
0.5	OPΔT	39.1	HDP	49.4	--	--	LSGL	105.5	105.6	909.7
0.6	OPΔT	31.1	HDP	39.6	--	--	LSGL	97.0	92.2	906.2
0.7	OPΔT	27.3	HDP	34.4	--	--	LSGL	92.4	83.2	904.8
0.8	OPΔT	24.9	HDP	31.2	--	--	LSGL	89.3	76.6	904.2
0.9	OPΔT	23.1	HDP	28.8	--	--	LSGL	87.0	71.4	903.7
1.0	OPΔT	22.1	HDP	27.0	--	--	LSGL	84.9	66.6	903.4
1.2	OPΔT	20.7	HDP	23.8	--	--	LSGL	81.3	58.6	902.9
1.4	LSGL	18.7	HDP	20.8	--	--	LSGL	76.7	52.4	902.1

Note:

1. The SI signal is generated, but the RCS pressure remains too high for delivery of SI flow to the RCS.
2. MSIV Closure is irrelevant in these cases.

Key

LPP ≡ Low pressurizer pressure
HDP ≡ High differential steam pressure
HSF/LT_{avg} ≡ High steam flow + low T_{avg}
LSGL ≡ Low-low steam generator water level
OPΔT ≡ Over power ΔT
HNFLX ≡ High nuclear flux

Table 6.6-17

IP2 MSLB Outside Containment M&E Release
Time Sequence Summary – 70% Power, Loop Break

Break Size (ft^2)	Reactor Trip		Safety Injection[1]		MSIV Closure		Auxiliary Feedwater		Time Faulted SG Tubes Uncover (sec)	Time Break Releases Stop (sec)
	Signal	Time Rod Motion Starts (sec)	Signal	Time of Signal (sec)	Signal	Time Fully Closed (sec)	Signal	Time Flow Starts (sec)		
0.1	LSGL	246.6	HDP	466.2	--	--	LSGL	304.6	427.9	1009.8
0.2	LSGL	126.8	HDP	158.0	--	--	LSGL	184.8	237.8	944.4
0.3	LSGL	86.6	HDP	104.6	--	--	LSGL	144.6	173.4	923.8
0.4	LSGL	66.4	HDP	79.4	--	--	LSGL	124.4	140.4	915.3
0.5	LSGL	53.5	HDP	63.6	--	--	LSGL	111.5	119.6	910.5
0.6	LSGL	35.2	HDP	41.4	--	--	LSGL	93.2	96.0	907.7
0.7	LSGL	26.3	HDP	30.2	--	--	LSGL	84.3	84.4	906.1
0.8	LSGL	21.1	HDP	23.8	--	--	LSGL	79.1	79.2	905.0
0.9	LSGL	17.7	HDP	19.4	--	--	LSGL	75.7	74.6	903.7
1.0	LSGL	15.3	HDP	16.2	--	--	LSGL	73.3	69.2	903.4
1.2	LSGL	12.2	HDP	11.4	--	--	LSGL	70.2	61.4	903.0
1.4	SI	9.00	HDP	7.0	--	--	SI	67.0	55.6	902.1

Note:

1. The SI signal is generated, but the RCS pressure remains too high for delivery of SI flow to the RCS.
2. MSIV Closure is irrelevant in these cases.

Key

LPP ≡ Low pressurizer pressure
HDP ≡ High differential steam pressure
HSF/LT$_{avg}$ ≡ High steam flow + low T$_{avg}$
LSGL ≡ Low-low steam generator water level
OPΔT ≡ Over power ΔT
HNFLX ≡ High nuclear flux

Table 6.6-18 IP2 Outside Containment Steam & Feed Penetration Area Initial Conditions			
	Pressure (psia)	Temperature (°F)	Relative Humidity (%)
Winter			
Inside	14.7	110.0	100.0
Outside	14.7	84.0	70.0
Summer			
Inside	14.7	125.0	100.0
Outside	14.7	100.0	90.0

Table 6.6-19 Vented Steam Releases from Operable Steam Generators and Auxiliary Feedwater Flows for the 0 – 2, 2 – 8, and 8 – 40 Hr Time Periods						
	Vented Steam Release			Auxiliary Feedwater Injection		
Event	0-2 hours	2-8 hours	8-40 hours	0-2 hours	2-8 hours	8-40 hours
Locked Rotor	384,000 lbm	860,000 lbm	1,488,000 lbm	568,000 lbm	943,000 lbm	1,488,000 lbm
Steamline Break	381,000 lbm	830,000 lbm	1,488,000 lbm	519,000 lbm	892,000 lbm	1,488,000 lbm

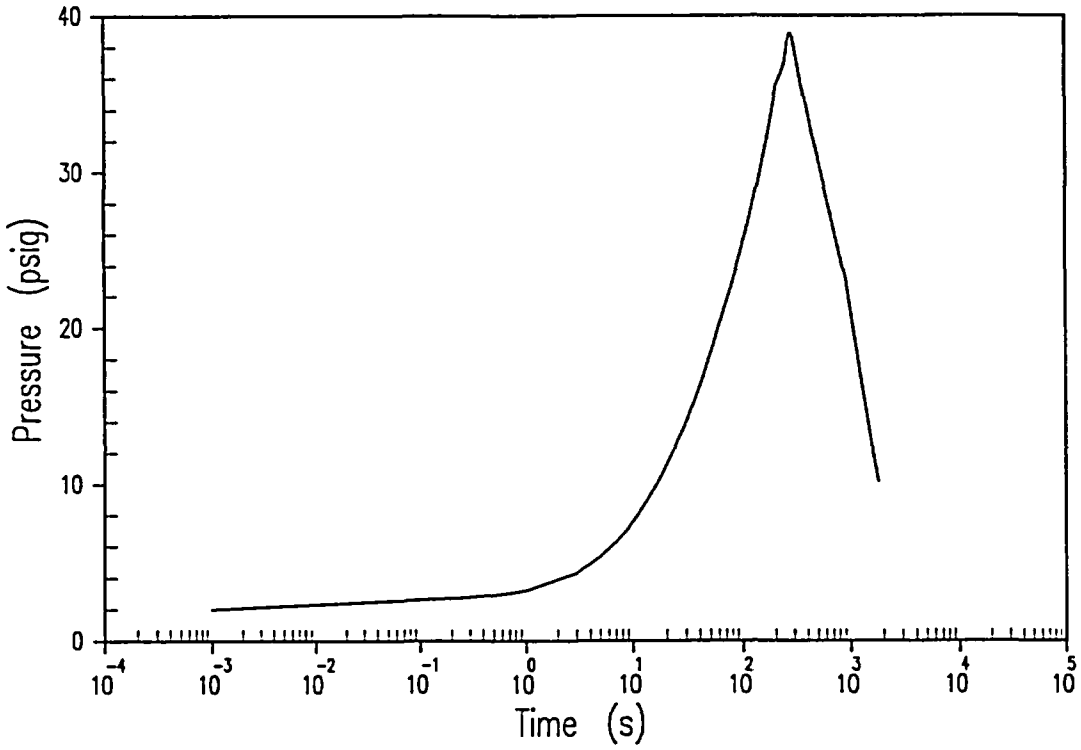


Figure 6.6-1

Containment Pressure Curve for Steamline Break for IP2

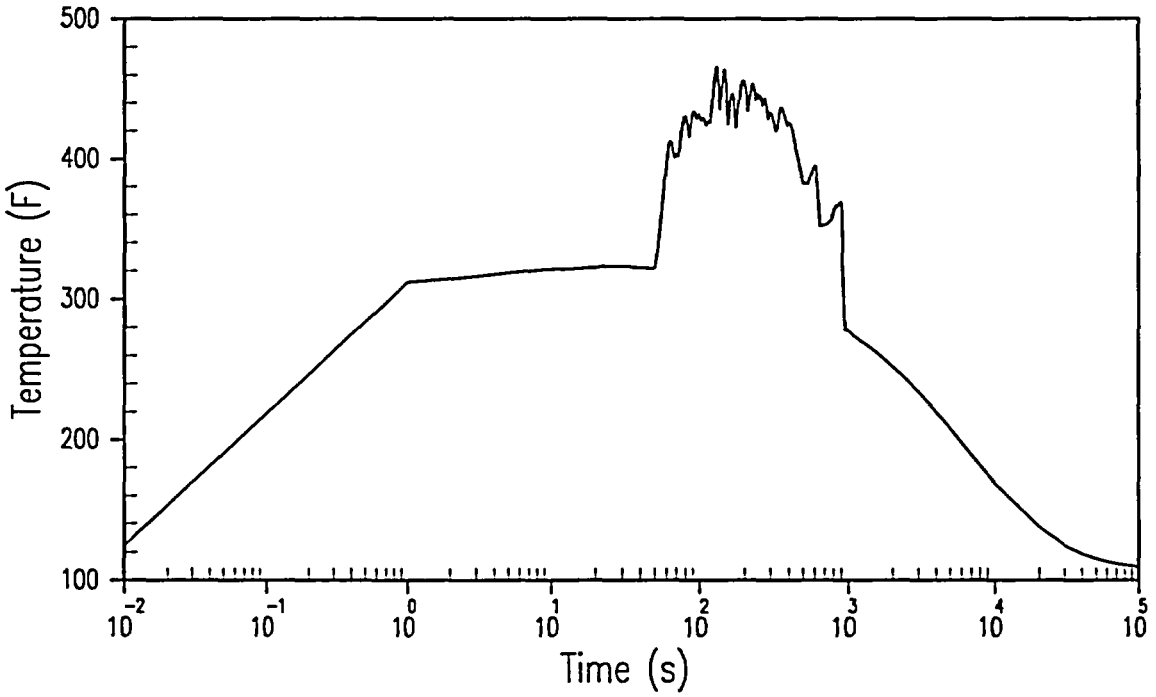


Figure 6.6-2

IP2 MSLB Outside Containment Composite Temperature Profile

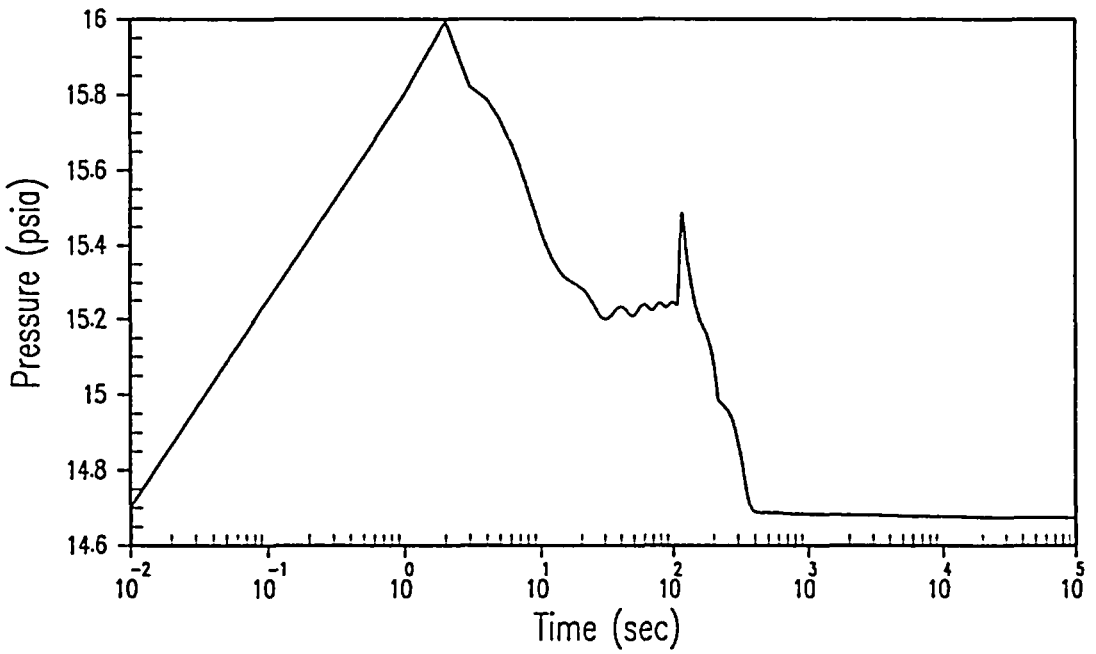


Figure 6.6-3

IP2 MSLB Outside Containment Composite Pressure Profile

6.7 Loss-of-Coolant Accident Hydraulic Forces

6.7.1 Introduction

The loss-of-coolant accident (LOCA) hydraulic forces analysis generates the hydraulic forcing functions that would act on Reactor Coolant System (RCS) components as a result of a postulated LOCA. The LOCA hydraulic forces were calculated for conditions consistent with minimum thermal design flow and maximum RCS power. The most recent qualification of the vessel internals and fuel was performed using an advanced beam model version of MULTIFLEX (3.0), Reference 1, in accordance with methodology approved by the NRC in Reference 2. This same version of the MULTIFLEX code was used in the LOCA hydraulic forces analysis for the Indian Point Unit 2 (IP2) Stretch Power Uprate (SPU) Project.

6.7.2 Input Parameters and Assumptions

To conservatively calculate LOCA hydraulic forces for Indian Point Unit 2, the following operating conditions were considered in establishing the limiting temperatures and pressures:

- Initial RCS conditions associated with a minimum thermal design flow of 80,700 gpm per loop
- Uprated core power of 3216 MWt (Nuclear Steam Supply System [NSSS] power of 3230 MWt)
- A nominal RCS hot full power (HFP) T_{avg} range of 549.0 to 572.0°F. This provides an RCS T_{cold} range of 514.3 to 538.2°F (see Table 2.1-2).
- An RCS temperature uncertainty of ±6.6°F. (The minimum analyzed T_{cold} was 507.7°F.)
- A feedwater temperature range of 390.0 to 436.2°F
- A nominal RCS pressure of 2250 psia
- A pressurizer pressure uncertainty of ±50 psi

General Design Criterion 4 (GDC-4) (Reference 3) 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). The technical justification for application of LBB to IP2 is documented in WCAP-10977 (Reference 4).

LBB licensing allows RCS components to be evaluated for LOCA integrity considering the next most limiting auxiliary line breaks, that for IP2, are the accumulator line, the pressurizer surge line, and the residual heat removal line.

6.7.3 Description of Evaluation

LOCA forces were generated with a focus on the component of interest; loop, vessel, steam generator, or rod control cluster assembly (RCCA) guide tubes using the advanced beam model version of MULTIFLEX (3.0) (Reference 1), assuming a conservative break opening time of 1 millisecond.

Generally, this improved modeling results in lower, more realistic, but still conservative hydraulic forces on the core barrel.

The MULTIFLEX computer code calculated the thermal-hydraulic transient within the RCS and considers subcooled, transition, and early two-phase (saturated) blowdown regimes. The code used the method of characteristics to solve the conservation laws, assuming one-dimensional (1-D) flow and a homogeneous liquid-vapor mixture. The RCS was divided into subregions in which each subregion was regarded as an equivalent pipe. A complex network of these equivalent pipes was used to represent the entire primary RCS.

For the reactor pressure vessel (RPV) and specific vessel internal components, the MULTIFLEX code generated the LOCA thermal-hydraulic transient that was input to the LATFORC and FORCE2 post-processing codes (Reference 5). These codes, in turn, were used to calculate the actual forces on the various components.

These forcing functions for horizontal and vertical LOCA hydraulic forces, combined with seismic, thermal, and system shaking loads, were used by the cognizant structural groups to determine the resultant mechanical loads on the RPV and vessel internals.

The loop forces analysis use the THRUST post-processing code to generate the X, Y, and Z directional component forces during a LOCA blowdown from the RCS pressure, density, and mass flux calculated by the MULTIFLEX code. The THRUST code is described and documented in WCAP-8252 (Reference 6).

The hydraulic transient time history data were extracted directly from the MULTIFLEX output. For steam generator and some reactor vessel internal components, such as baffle bolts or RCCA guide tubes.

6.7.4 Acceptance Criteria

LOCA hydraulic forces were provided as input to structural qualification analyses, and as such, had no independent regulatory acceptance criteria. The structural analyses performed using these forcing functions were performed to demonstrate compliance with 10CFR50, Appendix A, (Reference 3) GDC 4.

6.7.5 Results

For the IP2 SPU Project, all relevant LOCA hydraulic forces analyses were performed directly at the uprated power operating conditions using models specific to the IP2 NSSS design. These analyses included reactor vessel internals and fuel, loop piping, steam generator, and RCCA guide tube forces. The results of these analyses were then used as input to the structural analyses for component qualification.

6.7.6 Conclusions

LOCA hydraulic forces were generated for IP2 for the uprate conditions specified in subsection 6.7.2 of this document.

6.7.7 References

1. WCAP-9735, Rev. 2, (Proprietary), and WCAP-9736, Rev. 1, (Nonproprietary), *MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model,* K. Takeuchi, et al., February 1998.

2. WCAP-15029-P-A (Proprietary), and WCAP-15030-NP-A (Nonproprietary), *Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions*, R. E. Schwirian, et al., January 1999.

3. 10CFR50, Appendix A, *General Design Criteria for Nuclear Power Plants*, NRC.

4. WCAP-10977, Rev. 2, (Proprietary), and WCAP-10976, (Nonproprietary), *Technical Bases for Eliminating Large Primary Loop Rupture as the Structural Design Basis for Indian Point Unit 2*, F. J. Witt, et al., December 1986.

5. WCAP-8708-P-A (Proprietary) and WCAP-8709-A (Nonproprietary), *MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics*, K. Takeuchi, et al., September 1977.

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

6.8 Anticipated Transients Without Scram

6.8.1 Introduction

For Westinghouse-designed pressurized water reactors (PWRs), the licensing requirements related to anticipated transients without scram (ATWS) are specified in the Final ATWS Rule, 10CFR50.62(c) (Reference 1). The requirement set forth in 10CFR50.62(c) is that all Westinghouse-designed PWRs must install AMSAC (ATWS Mitigation System Actuation Circuitry), and in compliance with this, AMSAC has been installed and implemented at Indian Point Unit 2 (IP2).

As documented in SECY-83-293 (Reference 2), the analytical bases for the Final ATWS Rule are the generic ATWS analyses for Westinghouse PWRs generated by Westinghouse in 1979. These generic ATWS analyses were formally transmitted to the Nuclear Regulatory Commission (NRC) via letter NS-TMA-2182 (Reference 3) and were performed based on the guidelines provided in NUREG-0460 (Reference 4).

In the generic ATWS analyses documented in NS-TMA-2182 (Reference 3), ATWS analyses were performed with the LOFTRAN computer code for the various American Nuclear Society (ANS) Condition II events (that is, anticipated transients), considering various Westinghouse PWR configurations applicable at that time. These analyses addressed two-, three-, and four-loop PWRs with various steam generator models. For IP2, the generic ATWS analyses applicable at that time were those for a four-loop PWR with Model 44 steam generators and a core power of 3025 MWt. These conditions are summarized in Table 3-1-d of NS-TMA-2182 (Reference 3). For this plant configuration, the peak Reactor Coolant System (RCS) pressure reported in NS-TMA-2182 for the limiting loss-of-load ATWS event is 2979 psia.

The generic ATWS analyses documented in NS-TMA-2182 (Reference 3) also support the analytical basis for the NRC-approved generic AMSAC designs generated for the Westinghouse Owners Group (WOG), as documented in WCAP-10858P-A, Revision 1 (Reference 5). For the purpose of these AMSAC designs, the generic ATWS analyses for the four-loop PWR configuration with Model 51 steam generators were used to conservatively represent all of the various Westinghouse PWR configurations contained in NS-TMA-2182. For IP2, WCAP-10858P-A AMSAC Logic 2, AMSAC Actuation on Low Main Feedwater Flow was used.

For the subject power uprating, an increase from a Nuclear Steam Supply System (NSSS) power of 3127 MWt to an NSSS power of 3230 MWt is proposed. This reflects a power increase of 6.8 percent above that considered in the generic ATWS analysis for the four-loop PWRs with Model 44 steam generators. As documented in NS-TMA-2182 (Reference 3), an increase in core thermal power adversely affects the results of the ATWS analyses. As reported

for the generic four-loop PWR with Model 51 steam generators, an increase in power of 2 percent increases peak RCS pressure by 44 psi in the limiting loss-of-load ATWS. As demonstrated in NS-TMA-2182, the peak RCS pressure with the 2-percent increase in power remains below 3200 psig. This ATWS sensitivity analysis was performed assuming a 2-percent variation in power consistent with the typical calorimetric measurement uncertainty on power at the time of these analyses. Based on this sensitivity, the proposed increase in power of 6.8 percent would increase the RCS pressure in the limiting loss of load ATWS event by 150 psia.

As prescribed by NUREG-0460 (Reference 4), the 1979 generic ATWS analyses for Westinghouse PWRs documented in NS-TMA-2182 (Reference 3) assumed a full-power moderator temperature coefficient (MTC) of -8 pcm/°F. A sensitivity analysis including the use of an MTC of -7 pcm/°F was also provided as prescribed by NUREG-0460. In 1979, the MTC values of -8 pcm/°F and -7 pcm/°F represented MTCs that Westinghouse PWRs would be more negative than for 95 and 99 percent of the cycle, respectively. The base case of 95 percent represents a 95-percent confidence limit on favorable MTC for the fuel cycle. For IP2, the Technical Specification requirement on MTC is limited to < 0 pcm/°F at all power levels. The current MTC Technical Specification for IP2 remains the same as that which was applicable for most Westinghouse PWRs in 1979. Therefore, the reactivity feedback for IP2 remains sufficiently negative to be comparable to the generic Westinghouse ATWS analyses presented in NS-TMA-2182.

Relative to the other conditions important to the ATWS analyses, the pressurizer power-operated relief valve (PORV) relief capacity, safety valve relief capacity, and auxiliary feedwater (AFW) capacity is unaffected by the proposed stretch power uprate (SPU). The design capacity of each IP2 pressurizer PORV (179,000 lbm/hr) and pressurizer safety relief valve (408,000 lbm/hr) are consistent with the relief capacities assumed in the 1979 generic ATWS analysis for this plant configuration.

The design capacities of the IP2 AFW pumps are as follows.

- Motor-driven AFW pump - 400 gpm
- Turbine-driven AFW pump - 800 gpm

The IP2 Auxiliary Feedwater System (AFWS) has two motor-driven AFW pumps (each pump aligned to 2 steam generators) and a turbine-driven AFW pump that requires operator action to initiate flow to all 4 steam generators. Therefore, the total design capacity of the IP2 AFWS, originally designed for 1600-gpm flow, can only be credited for a total flow of 800 gpm. The reduced flow results in an overall peak pressure penalty when compared to the total AFWS capacity of 1760 gpm, assumed in the 1979 generic ATWS analyses for the Westinghouse

four-loop plant configuration with Model 44 steam generators (as documented in Table 3-1-d of NS-TMA-2182 [Reference 3]). As reported for the generic four-loop PWR with Model 51 steam generators, the impact of a reduced AFW flow of 60 percent (that is, 960 gpm) increases the peak RCS pressure in the limiting loss of load ATWS event by 76 psi.

6.8.2 Conclusions

As a result of the IP2 SPU conditions, a 6.8-percent higher reactor power and conservative lower auxiliary feedwater flow rate of 60 percent may be assumed as when compared to the generic limiting loss-of-load ATWS event applicable for IP2. The higher reactor power and lower auxiliary feedwater flow results in a combined overall peak RCS pressure penalty (increase) of 226 psi (150 psi + 76 psi) relative to the peak RCS pressure of 2979 psia reported in the generic ATWS analysis. This results in a net peak RCS pressure of 3205 psia (2979 psia + 226 psi), or 10 psi margin to the ATWS peak RCS pressure limit of 3215 psia (3215 psia – 3205 psia).

Based on the above, it is concluded that operation of IP2 at an uprated NSSS power of 3230 MWt remains in compliance with the Final ATWS Rule, 10CFR50.62(c) (Reference 1).

6.8.3 References

1. 10CFR50.62, *Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.*

2. SECY-83-293, *Amendments to 10CFR50 Related to Anticipated Transients Without Scram (ATWS) Events*, W. J. Dircks, July 19, 1983.

3. Letter NS-TMA-2182, T. M. Anderson (Westinghouse) to S. H. Hanauer (NRC), *ATWS Submittal,* December 30, 1979.

4. NUREG-0460, *Anticipated Transients Without Scram for Light Water Reactors,* December 1978.

5. WCAP-10858-P-A, *AMSAC Generic Design Package*, Westinghouse Topical Report, Rev. 1, M. R. Adler, July 1987.

6.9 Natural Circulation Cooldown Capability

6.9.1 Introduction

Certain initiating events such as a loss-of-offsite power can cause a reactor trip with loss of forced circulation. As the reactor coolant pumps (RCPs) coast down, a coolant density difference is established between the Reactor Coolant System (RCS) hot-leg and cold-leg sides that causes flow to circulate, allowing residual heat to be transferred to and removed by the steam generators. This process of natural circulation cooling has been observed in Westinghouse-designed pressurized water reactors (PWRs) in startup tests as well as in actual events. In addition, Diablo Canyon Unit 1, a 4-loop PWR similar to Indian Point Unit 2 (IP2), has performed a test to demonstrate the capability to cooldown the RCS to residual heat removal (RHR) initiation conditions (below 350°F and ~400 psig) via this natural circulation cooling process. The recovery guidance used for this test as well as the IP2 plant-specific Emergency Operating Procedures (EOPs) has been based on the Westinghouse Owners Group Emergency Response Guidelines (ERGs), specifically ES-0.2, Natural Circulation Cooldown.

To demonstrate that the stretch power uprate (SPU) does not adversely affect the natural circulation cooling capability of the IP2 plant, an analysis simulation was performed. In addition to supporting the technical basis for the EOPs, this simulation demonstrated the following:

- The maximum temperature differential ($T_{hot} - T_{cold}$) and maximum hot-leg temperatures are bounded by full-power operation,
- The capacity of the steam generator power-operated relief valves (steam generator ARVs) does not limit the capability to cooldown to RHR cut-in conditions (350°F, 370 psig), and
- RHR can be placed in service prior to depletion of the Technical Specification volume in the condensate storage tank (CST) (360,000 gallons).

6.9.2 Analysis Inputs

The IP2 plant EOPs, which are based on the ERGs, were followed in performing the analysis simulation. This analysis was performed in a conservative manner using realistic time delays and equipment limitations. For example, the simulation assumed a "locked-rotor" RCP hydraulic resistance following RCP coastdown, a 4-hour delay at hot standby to allow boration to cold shutdown, a natural circulation cooldown rate of 20°F/hr (versus a maximum 25°F/hr allowed for a T_{hot} upper-head plant), and an 8-hour delay to allow the upper head to cool or "soak" before depressurizing to the RHR cut-in pressure. As per the ERG generic analysis, this upper-head

soak delay is included to allow the upper-head region sufficient time to cool due to the assumed loss of control rod drive mechanism (CRDM) fans. If the CRDM fans were operating, the upper-head region would cool down at a rate comparable to the rest of the RCS and this 8-hour delay to preclude steam void formation in the upper head would not be necessary.

Other important assumptions were:

- Decay heat rate approximately the same as the ANSI/ANS-5.1-1979 standard (Reference 1), including +2 sigma uncertainty, with full-power operation at 3287.5 MWt core power for an extended period of time (3.2 years average fuel exposure). (This power level bounds 102 percent of 3216 MWt.)

- Total capacity for all four SG ARVs = 1.369×10^6 lbm/hr at the valve inlet pressure of 1005 psig = 1020 psia.

- Condensate storage tank (CST) available inventory of 360,000 gallons.

6.9.3 Simulation Results

For the short-term maximum temperature response, the decay heat is approximately 3 percent of full power by the time the RCPs coast down and the core/hot-leg side heats up to quasi steady-state conditions. This condition occurs approximately 5 minutes after the RCPs and the reactor trip. Results calculated for this situation are as follows:

- Hot-leg/core exit temperature = 593°F
- Hot-to-cold leg ΔT = 37°F
- Cold-leg temperature = 556°F
- Core flow rate $\cong 6.1 \times 10^6$ lbm/hr (approximately 4.5 percent of nominal)

For this maximum temperature condition, the cold-leg temperatures are assumed to be controlled by the lowest main steam safety valve (MSSV) pressure set-point (1080 psia, T_{sat} = 554°F). Soon after reactor trip, the operator would control this temperature to no-load (547°F), as instructed in the EOPs, by operation of the steam generator ARVs. Thus, the above temperatures for T_{hot} and T_{cold} would be reduced accordingly by about 9°F. The above hot-leg/vessel-outlet temperature is approximately 13°F less than the maximum Performance Capability Working Group (PCWG) temperature of 605.8°F (see Table 2.1-2). Since the RCS is initially controlled to ~2100 to 2250 psia (T_{sat} = 643 to 653°F), it would typically be subcooled by more than 50°F at the core exit/hot legs at this maximum temperature condition.

It was determined that the capacities of the steam generator ARVs did not restrict the cooldown capability of the RCS. After borating to cold-shutdown boron concentration, the cooldown was simulated by controlling the pressure setpoints for the four steam generator ARVs. At approximately 15.6 hours after reactor trip, the RCS hot-leg and cold-leg temperatures had reached 346 and 320°F, respectively, conditions that would allow the RHR to be placed in service once the RCS is depressurized. Based on saturated critical flow from the four SG ARVs, the cooldown could be maintained at the assumed 20°F/hr rate with the valves slightly less than full open (~93 percent calculated).

The simulation was then extended to include the 8-hour upper-head soak, followed by depressurization and stabilization at RHR entry conditions. At the end of the 27-hour transient, the RCS pressure was stabilized at 375 psia (360 psig), T_{hot} = 339°F in all hot legs and at the core exit, and T_{cold} = 318°F in the cold legs. At that time, approximately 128 gpm of auxiliary feedwater was being used to remove decay heat (approximately 20.9 MWt, or ~0.64 percent of full power). After the 27-hour transient, approximately 342,000 gallons from the CST had been used, about 95 percent of the assumed available inventory (360,000 gallons). Of the 342,000 gallons, roughly 82 percent had been used for decay heat removal, approximately 9 percent had been used to fill the steam generators back to the normal narrow-range level (46- to 52-percent range), and the remaining 9 percent was used for sensible heat removal as required for the cooldown.

6.9.4 Conclusion

By demonstrating the natural circulation flow rates and temperature differentials are reasonable and that the steam generator ARVs are adequately sized to allow cooldown to RHR within the desired time frame, it is concluded that the IP2 SPU does not adversely affect the natural circulation cooldown capability of the plant.

6.9.5 References

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

6.10 Reactor Trip System/Engineered Safety Feature Actuation System Setpoints

6.10.1 Introduction

The Reactor Trip System (RTS)/Engineered Safety Feature Actuation System (ESFAS) nominal trip setpoints (NTS) and Technical Specification allowable values (AVs) have been reviewed for operation at the power uprate conditions. As a result of this review, several NTS and AV changes have been identified.

6.10.2 Description of Analyses and Evaluations

The setpoint analysis uses the square-root-sum-of-the-squares (SRSS) technique to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components, that 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. The methodology used for the IP2 SPU Program is the same as used for the recently NRC-approved 1.4-percent uprating (Reference 1). Where Technical Specification AVs were affected, these were recalculated in accordance with the Entergy methodology (Reference 2).

The IP2 RTS/ESFAS uncertainty calculations were evaluated based on operation at the uprate operating conditions, along with the plant-specific instrumentation and plant calibration procedures, and any revisions to the safety analysis limits (SALs) values that were required to support operation at the uprate conditions. Several setpoint calculations were affected due to revised SALs or changes in instrumentation hardware and scaling/calibration.

6.10.3 Acceptance Criteria and Results

The setpoint methodology defines the Total Allowance (TA) as the difference between the limiting SAL and the NTS. The Channel Statistical Allowance (CSA) is the statistical combination of the instrument channel uncertainty components. Margin is defined as the difference between the TA and the CSA. The acceptance criterion for the RTS/ESFAS setpoints is that margin is greater than or equal to zero.

Setpoint calculations were performed for the affected RTS/ESFAS parameters. Table 6.10-1 summarizes the most limiting SALs, NTS, and Technical Specification AVs for the parameters that were affected by the IP2 SPU Program. Incorporation of these AVs and NTS changes will support operation at power uprate conditions in a manner consistent with the UFSAR assumptions. Functions not listed in Table 6.10-1 were not affected by the IP2 SPU Program.

The Steam Generator Water Level uncertainty calculations included the resolution of the generic uncertainty issues (References 3-6), which are unrelated to the power uprate.

6.10.4 Conclusions

With the setpoint and allowable value changes as shown on Table 6.10-1, all of the RTS/ESFAS functions have acceptable margins and, therefore, are acceptable for operation at the uprated core power of 3216 MWt.

6.10.5 References

1. WCAP-15904-P, *Power Calorimetric Uncertainty for the 1.4-Percent Uprating of Indian Point Unit 2*, Rev. 1, May 2003.

2. ENN Specification No. FIX-95-A-001, *Guidelines for Preparation of Instrument Loop Accuracy and Setpoint Determination Calculations*, Rev. 1, November 2001.

3. NSAL-02-03, *Steam Generator Mid-deck Plate Pressure Loss Issue*, Rev. 1, April 2002.

4. NSAL-02-04, *Maximum Reliable Indicated Steam Generator Water Level*, Rev. 0, February 2002.

5. NSAL-02-05, *Steam Generator Water Level Control System Uncertainty Issue*, Rev. 1, April 2002.

6. NSAL-03-09, *Steam Generator Water Level Uncertainties*, Rev. 0, September 2003.

Table 6.10-1 IP2 SPU Summary of RTS/ESFAS Setpoint Calculations			
Protection Function	**NTS**	**SAL Value**	**Tech. Spec. AV**
Nuclear Instrumentation System (NIS) Power Range Reactor Trip High Setpoint	≤109% rated thermal power (RTP)	116% RTP	≤110.6% RTP
Overtemperature ΔT Reactor Trip			≤4.9%ΔT span above computed setpoint
K_1 Max		1.42	
K_1 Nominal	≤1.22		
K_2	0.020 /°F	0.020 /°F	
K_3	0.00070 /psi	0.00070 /psi	
f(ΔI) Function Between (-30% and +7%)	0	0	
Positive Slope (ΔI>7%)	2.25% RTP/%ΔI	2.25% RTP/%ΔI	
Negative Slope (ΔI<-30%)	1.97% RTP/%ΔI	1.97% RTP/%ΔI	
Overpower ΔT Reactor Trip			≤2.4% ΔT span above computed setpoint
K_4 Max		1.164	
K_4 Nominal	≤1.074		
K_5 (decreasing T_{avg})	0	0	
K_5 (increasing T_{avg})	0.0188/°F	0.0188/°F	
K_6 (T≥T")	0.0015/°F	0.0015/°F	
K_6 (T<T")	0	0	
RCS Flow Low Reactor Trip	≥92% loop flow	85.0% flow	≥88.7% loop flow
Steam Generator Water Level – Low-Low Reactor Trip	≥7% span	0% span	≥3.4% span
Steam Generator Water Level – High-High Feedwater Isolation	≤73% span	90% span	≤88.3% span
Steamline Pressure Low (safety injection/steamline [SI/SL] actuation)	≥565.3 psig	515.3 psig	≥540.3 psig

Table 6.10-1 (Cont.) IP2 SPU Summary of RTS/ESFAS Setpoint Calculations			
Protection Function	**NTS**	**SAL Value**	**Tech. Spec. AV**
Steam Flow in Two Steamlines – High (SI/SL actuation)	≤40% full flow between 0 and 20% load, increasing linearly to ≤110% full flow at 100% load	64% full flow between 0 and 20% load, increasing linearly to 144% full flow at 100% load	≤45.9% full flow between 0 and 20% load, increasing linearly to ≤122% full flow at 100% load
T_{avg} – Low (SI/SL actuation)	≥542°F	537°F	≥540.5°F

6.11 Radiological Assessments

6.11.1 Introduction

This section addresses the radiological effects of power uprate at Indian Point Unit 2 (IP2). The current licensing basis core power level is 3114.4 MWt. The stretch power uprate (SPU) core power level is 3216 MWt (that is, an increase of approximately 3.26 percent with respect to the current power level).

Additionally, as holder of an operating license issued prior to January 10, 1997, and in accordance with 10CFR50.67 (Reference 1) and *Standard Review Plan* (SRP) 15.0.1 (Reference 2), the accident source terms used in the IP2 uprated design basis offsite and control room dose analyses have been revised to reflect the full implementation of alternative source terms (ASTs) as detailed in Regulatory Guide (RG) 1.183 (Reference 3).

The first use of the AST for IP2 was reviewed and approved by the Nuclear Regulatory Commission (NRC) in its SER for Operating License (OL) Amendment No. 211 (Reference 4).

The SPU was evaluated for its effect on the following radiological areas:

- Normal operation dose rates and shielding
- Normal operation annual radwaste effluent releases
- Radiological environmental doses for equipment qualification (EQ)
- Post-loss-of-coolant accident (LOCA) access to vital areas
- Post-accident offsite and control room doses

In accordance with regulatory guidance, radiological evaluations for accident-related issues are assessed at a core power level of 3216 MWt plus 2 percent to address power measurement uncertainties (for a total of 3280.3 MWt). Installation of improved feedwater measurement instrumentation used for calorimetric power calculation allows for instrument error to be reduced from the traditional 2 percent as recommended in RG 1.49 (Reference 5). The reduction of the uncertainty allowance for calorimetric thermal power measurement to 0.6 percent was approved by the NRC in its SER for License Amendment No. 237 for IP2 (Reference 6). However, IP2 has decided to return to the use of the traditional 2 percent uncertainty.

Except as noted, radiological evaluations for normal operation-related issues were assessed for power uprate at a core power level of 3216 MWt. In accordance with regulatory guidance, the radwaste effluent assessment assumed a core power level of 3280.3 MWt, but used flow rates and coolant masses at the Nuclear Steam Supply System (NSSS) power level.

With the exception of the offsite and control room dose assessments, the uprate evaluations discussed in this section (that is, those associated with normal operation dose rate/shielding adequacy, normal operation radwaste effluents, environmental levels for equipment qualification, and vital access) were based on scaling techniques. The scaled increase in radiation levels also included the effect of the change in fuel cycle length and the use of current computer codes, methodology, and nuclear data in developing the uprated core and reactor coolant inventory, versus the methodology computer tools, and nuclear data used in the development of the original licensing basis core/reactor coolant inventory. Note that for the most part, the percentage of the estimated increase that can be attributed directly to the power uprate was approximately the percentage of the core uprate.

The radiological consequences for the following design basis accidents (DBAs) were reanalyzed to support the SPU Program:

- Main steamline break (MSLB)
- Locked reactor coolant pump (RCP) rotor
- Rod ejection
- Steam generator tube rupture (SGTR)
- Small-break loss-of-coolant accident (SBLOCA)
- Large-break loss-of-coolant accident (LBLOCA)
- Waste gas decay tank (GDT) rupture
- Volume control tank (VCT) rupture
- Holdup tank (HT) failure
- Fuel-handling accident (FHA)

The radiological consequences of all of these accidents (except for the failure of the VCT, GDT, and HT) have been approved by the NRC (Reference 4) for the implementation of AST at IP2, using the analytical methods and assumptions outlined in NUREG-1465 (Reference 7). IP2 was a pilot plant for the application of AST. Since that time, RG 1.183 (Reference 3) has been approved for use as the methodology document for AST application.

The analyses performed for the SPU Program followed the methodology outlined in RG 1.183 (Reference 3). The implementation of the RG 1.183 methodology affected the assumed gap fractions, activity in failed fuel in the rod ejection and locked rotor accidents, and the gap activity release timing in the LOCA. The analyses that had been approved by the NRC in Reference 4 have all been updated consistent with RG 1.183 guidance and using input assumptions consistent with the proposed nominal core power of 3216 MWt and are presented in subsection 6.11.9 of this document. A change in the Technical Specification for primary-to-secondary leak rate has also been incorporated.

The updated offsite and control room dose analyses reflected uprated conditions and the AST (as applicable).

6.11.2 Regulatory Approach

Summarized below are the regulatory acceptance criteria that were used for the uprate assessments.

6.11.2.1 Normal Operation Assessments

The regulatory commitments currently associated with normal operation assessments are not affected by this application and remain applicable for the uprate assessment:

- Normal operation onsite dose rates and available shielding will meet the requirements of 10CFR20 (Reference 8) as it relates to allowable operator exposure and access control.

- Normal operation offsite releases and doses will meet the requirements of 10CFR20 and 10CFR50, Appendix I (Reference 9). Performance and operation of installed equipment as well as reporting of offsite releases and doses will continue to be controlled by the requirements of the Technical Specifications (Reference 10).

6.11.2.2 Accident Assessments

The regulatory commitments associated with accident assessments were revised as noted by this application and are summarized below:

- Offsite and Control Room Doses: As part of the uprate application, IP2 is updating its implementation of the AST to be consistent with RG 1.183 (Reference 3).

  The acceptance criteria for the exclusion area boundary (EAB) and low-population zone (LPZ) doses are based on 10CFR50.67 (Reference 1) and Table 6 of RG 1.183 (Reference 3) (also noted in Table 1 of SRP 15.0.1 [Reference 2]):

  — An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release should not receive a radiation dose in excess of the accident-specific total effective dose equivalent (TEDE) value noted in RG 1.183 (Reference 3), Table 6.

— An individual located at any point on the outer boundary of the LPZ who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) should not receive a radiation dose in excess of the accident-specific TEDE value noted in RG 1.183 (Reference 3), Table 6.

— The GDT rupture, VCT rupture, and HT failure are not specifically addressed in RG 1.183 (Reference 3) or in the pilot program approved by the NRC in Reference 4). The acceptance criterion used for these events is assumed to be 0.5 rem consistent with the guidance of RG 1.26 (Reference 11). The criterion is applied as 0.5 rem TEDE to be consistent with an AST application.

- Control Room Dose: The acceptance criterion for the control room dose is based on 10CFR50.67 (Reference 1).

  — Adequate radiation protection is provided to permit occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

- Equipment Qualification: The uprate EQ assessment takes into consideration the impact of core power uprate using scaling techniques and TID-14844 source terms (Reference 12). This approach is consistent with the NRC Safety Evaluation Report (SER) (Reference 4) issued for the full implementation of the AST.

- Vital Access Doses: The vital access dose assessment for uprate takes into consideration the impact of core power uprate using scaling techniques and TID-14844 (Reference 12) source terms. This approach is acceptable based on the benchmarking study reported in SECY-98-154 (Reference 13), which concluded that results of analyses based on TID-14844 would be more limiting earlier on in the event

6.11.3 Computer Codes

The Quality Assurance (QA) Category 1 computer codes were used to support this application. The computer codes have been used extensively to support nuclear power plant design.

6.11.4 Radiation Source Terms

6.11.4.1 Introduction

This section describes the input parameters and methodology used in the calculation of radiation source terms applicable to the IP2 SPU Project. Radiation source terms for several different accident- and normal-operating conditions were determined for the power uprate conditions. These source terms were used as input to dose and balance-of-plant (BOP) analyses. The reanalyzed areas included the following:

- Core inventory and fuel-handling accident (FHA) fission product activities
- Reactor Coolant System (RCS) sources
- Volume control tank (VCT) sources
- Liquid tank sources
- Upgrade of control room direct dose

Each of these source term calculations is discussed in subsequent subsections.

6.11.4.2 Core Inventory and FHA Sources

6.11.4.2.1 Input Parameters and Assumptions

The assumptions and input parameters used in the determination of the total core inventory are summarized in Tables 6.11-1 and 6.11-2.

6.11.4.2.2 Description of Analysis

Fuel burnup and fission product production were modeled using the ORIGEN2 code. ORIGEN2 is a versatile point-depletion and radioactive decay code for use in simulating nuclear fuel cycles and calculating the nuclide concentration and characteristics of materials contained therein. The code considers the transmutation of isotopes in the material. For the relatively high fluxes in the core region of the reactor, burn-in and burn-out of isotopes can have an important effect. This is particularly true for fuel cycle designs with high-burnup regions. These important effects are modeled in the ORIGEN2 calculations.

For the transition to cycles with the uprate power level, the core inventory calculation was performed for Cycle 16 and for 3 transition cycles, labeled Cycles 17 through 19. The core inventory for these 4 cycles differed very little. For the IP2 SPU program, Cycle 19 operating at the SPU power conditions was modeled in the ORIGEN2 calculations as the base case, the

case from which results were taken. The characteristics of Cycle 19 are provided in Tables 6.11-1 and 6.11-2.

The ORIGEN2 analysis for the SPU modeled a single fuel assembly from each region of the core. Burnup calculations that reflect each of the appropriate power histories were performed, and the total inventory for each region at the end of the transition cycle was then determined by multiplying the individual assembly isotopic inventory by the number of assemblies in the respective regions. Finally, the results for each region of the core were summed to produce the total core inventory.

To accommodate variations in fuel design and fuel management, a multiplier of 1.04 was applied to the core inventory of Cycle 19. Several decay times after shutdown were calculated for use in FHA dose calculations. The inventory for 1 average fuel assembly can be obtained by dividing the core inventory by 193 assemblies.

6.11.4.2.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to various radiological evaluations.

6.11.4.2.4 Results

The total core inventory of actinide and fission product activities for use in radiological evaluations are presented in Table 6.11-3.

6.11.4.3 RCS Fission Product Activities

6.11.4.3.1 Input Parameters and Assumptions

Based on the core loading parameters in Tables 6.11-1 and 6.11-2, the parameters used in the calculation of the reactor coolant fission product concentrations, including pertinent information concerning the expected coolant cleanup flow rate, are presented in Table 6.11-4. In the RCS activity calculations, fission product escape rate coefficients were used to model a 1-percent level of small cladding defects (that is, 1 percent of the operating fission product inventory in the core is being released to the primary coolant) in all fuel regions for the fuel cycle.

6.11.4.3.2 Description of Analysis

The fission product inventory in the reactor coolant during operation of the fuel cycle with a 1-percent level of small cladding defects was computed. No credit was taken for fission product

removal due to purge of the VCT. Furthermore, in determining the RCS inventory for individual isotopes, the maximum activity occurring at any time during the fuel cycle was documented in each case. Therefore, the total set of fission product concentrations did not represent any particular time during the fuel cycle, but rather, a composite of the maximum activity concentration exhibited by each isotope. This overall approach represented a conservative treatment of the RCS.

For fission products, effects of the following variations were estimated and included conservatively in the calculation of RCS activities:

- Lower-than-expected letdown flow
- Application of a 1.04 multiplier to calculated specific activities
- Core power increased by 2 percent for power determination uncertainty
- Low RCS mass

Tritium and corrosion product values, which are not directly related to reactor power, were taken as the greater of standard Westinghouse values or nominal values from ANSI/ANS-18.1-1999 (Reference 14).

6.11.4.3.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to radiological evaluations that are presented in subsection 6.11.9 of this report.

6.11.4.3.4 Results of Analyses

The RCS-fission-product- and corrosion-product-specific activities are given in Table 6.11-5 and were provided as input to radiological evaluations that are presented in subsection 6.11.9.

6.11.4.4 VCT Inventory

6.11.4.4.1 Input Parameters and Assumptions

The input and methods for calculating the VCT inventory are the same as those used in the RCS source calculations except that the VCT purification flow rate is based on the maximum flow rate (120 gpm) as opposed to the nominal flow rate (75 gpm) that was used in the RCS calculations.

No multiplier was used for the VCT vapor activities; rather, a 1.10 multiplier was used on the 120-gpm letdown rate value. In addition, for Kr-85, it is assumed the Kr-85 in the VCT was in equilibrium with the RCS in accordance with Henry's Law.

6.11.4.4.2 Description of Analyses

Radiological inventories for the VCT were based on the calculation of RCS and VCT nuclide concentrations with the maximum letdown flow (120 gpm) conservatively increased by 10 percent (to 132 gpm) to increase the nuclide concentrations in the VCT vapor.

Values for the gas inventory in the VCT are based on a vapor volume of 270 ft^3.

6.11.4.4.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to radiological evaluations that are presented in this section.

6.11.4.4.4 Results of Analyses

The VCT radionuclide inventory is given in Table 6.11-6.

6.11.4.5 Liquid Tank Sources

6.11.4.5.1 Input Parameters and Assumptions

RCS activities from subsection 6.11.4.4 above were used as input to the Chemical and Volume Control System (CVCS) holdup tank (HT) source analysis. It was assumed that the entire noble gas inventory in the liquid introduced to the tank would be released to the vapor space. This is a conservative assumption, since some portion would be retained in the liquid.

It is assumed that the level of RCS liquid in the CVCS HT (volume of 8100 cubic feet) is filled to 80 percent of total volume.

The RCS maximum letdown rate of 120 gpm was increased by 10 percent (to 132 gpm) for conservatism. The use of conservative RCS radioactivity concentrations (all selected at maximums, μCi/g determined from minimum RCS mass) would compensate for any instances in which letdown flow exceeded 132 gpm.

6.11.4.5.2 Description of Analysis

The radioactivity inventory of the CVCS HT was calculated assuming an initially empty tank and RCS letdown to the tank at 132 gpm and considered nuclide decay but did not consider reduction in RCS activities due to the letdown flow. The calculation used a maximum liquid volume of the CVCS HT as 80-percent full. The concern for analyses of CVCS HT volume is the volatile components, therefore the noble gas and iodine activities are of interest.

6.11.4.5.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to radiological evaluations that are presented in subsection 6.11.9.

6.11.4.5.4 Results of Analyses

The analysis results are shown in Table 6.11-7.

6.11.4.6 LOCA DBA Direct Control Room Dose

6.11.4.6.1 Input Parameters and Assumptions

The assumptions and input parameters used in determining the total core inventory are summarized in Tables 6.11-1 and 6.11-2.

Other input parameters for this analysis include the reactor containment vessel and containment Shield Building dimensions, the control room location relative to the Reactor Containment Building, the location and dimensions of selected Auxiliary Building walls and floors, and the time at which removal of gaseous activity starts. These are discussed below:

- The containment dome is a 67.5-foot radius hemisphere with a thickness of 3.5 feet of concrete.
- The cylindrical portion of the containment is treated as a 67.5-foot radius cylindrical shell with a thickness of 4.5 feet of concrete. The height of the cylinder is considered to be 145 feet.
- The containment liner has a thickness of ¼ inch of steel.
- The crane support wall has a thickness of 3 feet of concrete and is 49 feet in height.

- The shielding afforded by the control room walls and structures is equivalent to ½ inch of steel.

- Removal of non-gaseous activity occurs at 3.4 hours.

6.11.4.6.2 Description of Analyses

The gamma radiation going directly from the containment into the control room was calculated using the ORIGEN2 computer code. ORIGEN2 was used to calculate the DBA sources, while the point kernel method was used to calculate the gamma dose rates at a specific control room location. In the calculation, the containment volume was treated as 2 separate source regions, that is, the containment dome and the cylindrical section of the containment. The results from these 2 sources were then summed to give the total normalized dose rate.

The detector point was placed at a point just inside the control room location.

6.11.4.6.3 Acceptance Criteria

The calculation provided a radiation source to be used as input to the LOCA control room dose. As such, there are no specific criteria for these calculated results.

6.11.4.6.4 Results of Analyses

The 1-month calculated direct dose in the control room is 0.039 rem. Applying an additional 4 percent for fuel management variations gave a control room dose of 0.041 rem.

Dose rate and dose are illustrated in Figure 6.11-1.

6.11.5 Normal Operation Dose Rates and Shielding

6.11.5.1 Introduction

Cubicle wall thickness is specified not only for structural and separation requirements, but also to provide radiation shielding in support of radiological EQ and to reduce operator exposure during all modes of plant operation, including maintenance and accidents.

Conservative estimates of the radiation sources in plant systems and components were used to form the bases of normal operation plant shielding and radiation zoning. These radiation source terms were primarily derived from conservative estimates of the reactor core and RCS isotopic inventory, and were referred to as "design basis" source terms. The core uprate will affect the

isotopic inventory in the core. In addition, since the design basis RCS source term is based on 1-percent fuel defects, the uprate will result in an increase in the design basis RCS concentration.

The expected radiation source terms in the coolant also will be affected by the core uprate. Expected source terms are less than those allowed by the Plant Technical Specifications, and are usually significantly less than the design basis source terms.

The effect of the power uprate on the normal operation dose rates and the adequacy of existing shielding were evaluated to ensure continued safe operation within regulatory limits. This section also discusses the effect of the power uprate on the normal operation component of the total integrated dose used for radiological EQ.

6.11.5.2 Description of Analysis and Evaluations

The core power uprate from 3114.4 to 3216 MWt will increase the activity inventory of fission products in the core by approximately the percentage of the power uprate. The radioactivity levels in the primary coolant, secondary coolant, and other radioactive process systems and components also will be affected.

The original shielding design for IP2 was based on a core power level of 3216 MWt, a traditional 1-year fuel cycle, and a design RCS source term based on 3216-MWt/1-percent failed fuel. The uprate core inventory was based on 3280.3 MWt and a 24-month fuel cycle. The uprate design RCS source term was based on 3280.3-MWt/1-percent failed fuel and a 24-month fuel cycle. The inclusion of the 24-month fuel cycle in the uprated core increased the inventory of the long-lived isotopes.

The uprate assessment was divided into four parts and is summarized below:

- Areas near the reactor vessel where the dose rate was dominated by the reactor core neutron flux during power operation and gamma radiation from the irradiated fuel and neutron activated sources during shutdown.

- Areas in-containment adjacent to the RCS where the dose rate was dominated by the high-energy gammas associated with nitrogen-16 (N-16).

- Areas near spent fuel assemblies where the dose rate was dominated by the gamma radiation from the irradiated fuel.

- Areas outside the containment where the dose rate was determined by radiation sources derived from primary coolant activity.

1. Dose Rates near Reactor Vessel/Primary Shield Wall: During normal operation, the radiation source in the reactor core is primarily made up of neutron and gamma fluxes, which are approximately proportional to the core power level.

   The original calculations of neutron and gamma ray leakage from the IP2 reactor were based on a design basis core configuration that included fresh fuel in the core periphery and a core power level of 3216 MWt. This fuel management approach resulted in relatively high power generation at the periphery of the core, thus maximizing the neutron and gamma radiation levels external to the reactor vessel. In actual operations, the IP2 reactor was transitioned to low-leakage fuel management, which placed burned fuel in the periphery of the core. This fuel management strategy reduced radiation leakage by a factor of 2 to 4. The equilibrium fuel cycle defined for the IP2 SPU Program also represent a low-leakage core design. Based on the application of low-leakage fuel management and the original design being based on 3216 MWt, the reactor power uprate will have no appreciable effect on the design basis of the primary shielding and the dose rates adjacent to the reactor vessel/primary wall.

2. In-Containment Areas Adjacent to RCS/Secondary Shielding: During normal operation, the major radiation source in the RCS components located within containment is N-16, as the transit times from the core to the components are not sufficient for the N-16 to decay to negligible levels.

   The secondary shielding was designed to attenuate the radiation originating from the N-16 activity. N-16 is produced as the oxygen (of the water moderator) is exposed to the neutron flux present in the reactor core. The amount of activation is defined by the flux (or power) density of the core and the amount of time the moderator is resident in the core. The N-16 source radiation levels are essentially proportional to the reactor power. Therefore, it is expected that the normal operation dose rate inside containment adjacent to RCS areas will increase from the observed values by the ratio of uprated core power to the current core power, that is, 1.033. Note that due to its short half-life, the N-16 activity level will not be affected by the use of a 24-month fuel cycle.

   During shutdown, the major radiation sources in the RCS components located within containment are the deposited corrosion products on the internal surfaces and the decayed/degassed/filtered primary coolant activity. The small increase in reactor power for the uprate is not expected to have a significant effect on the corrosion product activity deposits.

3. Near Spent Fuel Assemblies: This source depends on both the power level and, for the long-lived isotopes, on the fuel burnup. Unless a piece of equipment is specifically exposed to an old spent fuel assembly, the exposure will increase by approximately the percentage of the power uprate, that is, ~3.3 percent. Due to the 24-month fuel cycle, which will increase the inventory of long-lived isotopes, the percentage increase in dose rate from an old spent fuel assembly may be slightly higher than the percentage of the uprate. However, this is not a significant concern as the dose rate near the spent fuel pit is dominated by the freshly discharged spent fuel assemblies and/or corrosion products in the pit, which formed the basis for shielding design.

4. Outside Containment: Outside the containment, the radiation sources are either the reactor coolant itself or downstream sources originating from coolant activity. A shielding review was performed based on the power uprate design primary coolant source terms (fission and activation products) versus the original design basis primary coolant source terms. The gamma energy emission rates by energy group for the power uprate were compared to the original primary coolant source terms. The sources evaluated included total primary coolant, degassed primary coolant, and the primary coolant noble gas source. Due to the change in isotopic compositions and gamma energy spectrum between the original and the uprated RCS fluid, the comparison was based on the dose rate shielded by 0, 1, 2, and 3 feet of concrete for the representative source geometry. The uprate evaluation reflected the different computer codes used in generating the source terms and shielding analyses, the difference in nuclear libraries, and considered the conservative, simplified modeling typically employed in shielding design. In addition, the evaluation considered the operation limits imposed by the Plant Technical Specifications on the primary coolant activity.

6.11.5.3 Acceptance Criteria

Following the power uprate, normal operation dose rates/available shielding must continue to meet those requirements of 10CFR20 (Reference 8) related to allowable operator exposure and access control.

The effect of the uprate on the normal operation radiation environment must be factored into the EQ Program.

6.11.5.4 Results and Conclusions

The consequences of the power uprate on shielding/radiation zones as they relate to personnel exposure and on the normal operation component of EQ are summarized below.

Effect of the Uprate on Shielding/Radiation Zones: Personnel Exposure

The power uprate will affect the radiation source terms in the core and the "expected" radiation source terms in the coolant. Expected source terms are less than those allowed by the Plant Technical Specifications and are usually significantly less than the design basis source terms.

The actual increase in radiation levels due to the uprate will not significantly affect radiation zoning or shielding requirements in the various areas of the plant because it is expected that this increase (which is expected to be the same as that of the uprate, that is, ~3.3 percent) will be offset by the:

- Conservative analytical techniques typically used to establish shielding requirements
- Conservatism in the pre-uprate design basis RCS source terms used to establish the radiation zones
- Plant Technical Specifications that limit the RCS concentrations to levels well below (about a factor of 3 to 4) the design basis source terms

It is noted that the large difference in the calculated design basis original and uprate RCS source terms is primarily due to the more advanced fuel burnup modeling/libraries utilized to develop the uprate source terms. The increase in fuel cycle length from 12- to 24-months also increases the long-lived isotopes in the fuel and RCS.

The dose rate ratios resulting from the uprate source to the pre-uprate source for the various design basis RCS source term/shielding configurations discussed previously ranged from 0.93 to 2.8. However, since the design basis uprated primary coolant activity was a very conservative source term (that is, based on 1-percent failed fuel, and including a 2-percent margin for power uncertainty and a 4-percent margin for fuel management schemes), credit was taken for a more realistic but limiting upper bound primary coolant activity based on the Plant Technical Specification. The evaluation demonstrated that the original design basis source term was conservative with respect to the uprate design basis source term for the noble gases. It also showed that the maximum dose rate increase due to the design basis uprate total RCS and degassed RCS source term of 2.8 was bounded by the Plant Technical Specifications, which

effectively limited the RCS non-gaseous activity levels to concentrations that were a factor of 3 to 4 less than design.

Therefore, taking into consideration the limits on RCS concentrations imposed by the Plant Technical Specifications, it was concluded that the shielding design based on the original design basis primary coolant activity will remain valid at the uprated condition.

Although the calculated normal operation dose rates based on the design basis source terms could, as a worst case, increase by a factor of ~2.8, the actual radiation levels (since the plant is already operating with an 24-month cycle) in most of the plant areas are expected to increase only by approximately the percentage of the uprate, that is, ~3.3 percent.

Regardless, individual worker exposures will be maintained within acceptable limits by the site As-Low-as-is-Reasonably-Achievable (ALARA) Program, which controls access to radiation areas.

Effect of the Uprate on the Normal-Operation Component for EQ

Per the IP2 *Environmental Qualification Manual* (Reference 15), the 40-year normal operation dose component of the total EQ dose was based on survey data associated with plant operation at 2758 MWt, which was scaled up to reflect various power levels, including 3216 MWt.

For the SPU, the normal operation component of the EQ dose will utilize the value currently noted in the EQ Manual (Reference 15) for a power level of 3216 MWt.

6.11.6 Normal Operation Annual Radwaste Effluent Releases

6.11.6.1 Introduction

Liquid and gaseous effluents released to the environment during normal plant operations contain small quantities of radioactive materials.

- Liquid Radioactive Waste: Liquids from reactor process systems, or liquids that have become contaminated with these process system liquids, are considered liquid radioactive waste. These wastes are processed according to their purity level (boron concentration, conductivity, insoluble solids content, organic content, and activity) before being recycled within the plant, discharged to the environment, or reprocessed through the Radwaste System for further purification until the dose guidelines of 10CFR50 Appendix I (Reference 9) are met.

- Gaseous Radioactive Waste: Airborne particulates and gases vented from process equipment as well as the building ventilation exhaust air are considered gaseous radioactive waste. The major source of gaseous radioactive waste (processing the reactor coolant by the gas stripper and the cover gas system) is continuously decayed using separate pressurized decay tanks, filtered, and monitored prior to release to ensure that the dose guidelines of 10CFR50, Appendix I (Reference 9) are not exceeded.

The liquid and gaseous radwaste systems' design must be such that the plant is capable of maintaining normal operation offsite releases and doses within the requirements of 10CFR20 (Reference 8) and 10CFR50, Appendix I (Reference 9). (Note that actual performance and operation of installed equipment and reporting of actual offsite releases and doses continues to be controlled by the requirements of the *Offsite Dose Calculation Manual* (Reference 16).

The core uprate will not change existing Radwaste Systems (gaseous and liquid) design, operating procedures, or waste inputs. Consequently, a comparison of releases can be made based on inventories/coolant concentrations in the RCS, and secondary side steam and water inventories and concentrations. As a result, the effect of the uprate on radwaste releases and 10CFR50 Appendix I (Reference 9) doses could be estimated using scaling techniques.

Based on an existing licensed core power level of 3114.4 MWt and an uprate core power level of 3216 MWt, it is expected that the radioactive effluents and consequent offsite doses will increase by approximately the percentage increase in core power, that is, 3.3 percent.

The conservatively performed uprate analysis discussed below, considered the plant core power operating history during the years 1997 to 2001, the reported effluent and dose data during that period, NUREG-0017 assumptions, and conservative methodology, to estimate the impact of operation at the core uprate power level of 3280 MWt (including instrument uncertainty) over that of current operation, on radioactive effluents and consequent offsite doses.

6.11.6.2 Description of Analyses and Evaluations

The power uprate will increase the activity level of radioactive isotopes in the primary and secondary coolant. Due to leakage or process operations, fractions of these fluids are transported to the liquid and gaseous radwaste systems where they are processed prior to discharge. As the activity levels in the primary and secondary coolant are increased, the activity level of radwaste inputs are proportionately increased. Regulatory guidance relative to methodology used to establish whether the radwaste effluent releases from a pressurized water

reactor (PWR) meet the requirements of 10CFR20 (Reference 8), and 10CFR50 Appendix I (Reference 9) is provided in NUREG-0017, Revision 1 (Reference 17).

The methodology used in the NUREG is independent of the fuel cycle length. In determining the nominal coolant activities provided in NUREG-0017 (Reference 17), isotopic concentrations from a number of plants and power levels were combined and adjusted to yield a dataset with a resulting range of uncertainty. Adjustment factors were provided to address facilities outside a nominal range in which coolant activities could be used without adjustment. The core power levels addressed for the IP2 base and uprate cases were within the range of applicability and input data that are used to develop NUREG-0017 (Reference 17).

The Indian Point 2 Annual Radioactive Effluent Release Reports for 1997 through 2001 (Reference 18) demonstrated that the current gaseous and liquid radwaste releases from the site are well within the release/dose limits set by 10CFR20 and 10CFR50, Appendix I (References 8 and 9). The effect of the power uprate on these releases was evaluated to ensure continued operation within regulatory limits.

The licensed reactor core power level of IP2 during the 1997 to 2001 time frame (Reference 17) was 3071 MWt. The uprate will increase the core power (including margin for power uncertainty) to 3280 MWt. The system parameters for uprated conditions reflect the flow rates and coolant masses at an NSSS power level of 3228.5 MWt and a core power level of 3280 MWt. For the pre-uprate condition, the evaluation used offsite doses based on an average 5-year set of organ and whole body doses calculated from effluent reports for the years 1997 through 2001 (Reference 18), including the associated average annual core power level extrapolated to 100 percent availability. Releases occurring during periods of shutdown were conservatively combined with operational releases and included in the doses scaled for 100 percent availability.

Using the methodology and equations found in NUREG-0017, Revision 1 (Reference 18) with the plant-specific parameters for the core uprate case, the percentage change for activity classes in the reactor coolant and secondary coolant (water and steam) was calculated. To estimate a bounding effect on offsite doses, the highest factor found for any chemical group of radioisotopes pertinent to the release pathway was determined. Then this factor was applied to the representative average doses at the pre-uprate conditions (100-percent availability) to estimate the maximum potential increase in effluent doses due to the uprate. This demonstrated that the estimated offsite doses following the uprate, although increased, remained below the regulatory limits.

6.11.6.3 Acceptance Criteria

The liquid and gaseous radwaste systems' design must be such that the plant is capable of maintaining normal operation offsite releases and doses within the requirements of 10CFR50, Appendix I (Reference 9) following the power uprate. (Note that actual performance and operation of installed equipment as well as reporting of actual offsite releases and doses continue to be controlled by the requirements of the Technical Specifications and the *Offsite Dose Calculation Manual* [Reference 16]). If the resulting doses estimated after the uprate are still a small fraction of the 10CFR50 Appendix I (Reference 9) limits, then it is reasonable to conclude that the IP2 Radwaste Systems and operating procedures will meet the design objectives of 10CFR50 Appendix I (Reference 9).

6.11.6.4 Results and Conclusions

As indicated earlier, based on an existing licensed core power level of 3114.4 MWt, and an uprate core power level of 3216 MWt, it is expected that the radioactive effluents and consequent offsite doses will increase by approximately the percentage increase in core power, that is, 3.3 percent.

Utilizing NUREG-0017 (Reference 17) assumptions and conservative methodology, the uprate analysis results summarized below utilizes the plant operating history to estimate the impact of power uprate on radioactive effluents and consequent offsite doses, by comparing plant operation at the uprated core power level of 3280 MWt (which includes margin for power uncertainty) to plant operation at 2941.3 MWt (the effective core power level during the period 1997 through 2001). The estimated doses following the uprate are presented in Table 6.11-8.

Expected Reactor Coolant Source Terms

Based on a comparison of pre-uprate versus uprate input parameters and the methodology outlined in NUREG-0017 (Reference 17), the maximum expected increase in the reactor coolant source (due primarily to the decrease in RCS mass [~8 percent] and increase in the effective core power level [~11.5 percent] that is, 3280.3 MWt [uprate power level]/2941.3 MWt [average power level during 1997 through 2001] between the pre- and post-uprate conditions) is approximately 20 percent for noble gases and 12 percent for other long half-life activity. Considering the accuracy and error bounds of the operational data utilized in NUREG-0017 (Reference 17), this percentage is well within the uncertainty of the existing NUREG-0017 (Reference 17) based expected reactor coolant isotopic inventory used for radwaste effluent analyses.

Liquid Effluents

As discussed above, there was a maximum 12-percent increase in the liquid releases as input activities are based on long-term RCS activity that are proportional to the effective core power uprate percentage increase, and on waste volumes which are essentially independent of the power level within the applicability range of NUREG-0017 (Reference 17). Tritium releases in liquid effluents were assumed to increase ~11.5 percent (corresponding to the effective core power uprate percentage) since the analysis identified changes in an existing facility's power rating without changing its mode of operation.

Gaseous Effluents

For all noble gases, there will be a bounding maximum 20 percent increase in effluent releases due to the effective core power uprate percentage increase. Gaseous effluents have two components: one which is based on RCS inventory and results in an 11.5-percent increase, and the other, which is based on concentration (due primarily to the decrease in RCS mass and increase in effective core power level between the pre- and post-uprate conditions), which result in a 20-percent increase. The limiting increase was used for this evaluation.

In actuality, gaseous releases of Kr-85 will increase by approximately the effective percentage of power increase (~11.5 percent). Gaseous isotopes with shorter half-lives either will have increases slightly greater than the effective percentage increase in power level up to a bounding value of 20 percent.

The effect of the power uprate on iodine releases was approximated by the effective power level increase but was evaluated at a 12-percent increase due to the calculated increased I-131 RCS concentration. The other components of the gaseous release (that is, particulates via the building ventilation systems and water activation gases) were not affected by the power uprate using the methodology outlined in NUREG-0017 (Reference 17). Tritium releases in the gaseous effluents increased in proportion to their increased production, which was directly related to core power and was allocated in this analysis in the same ratio as the pre-uprate releases.

For particulates, the methodology of NUREG-0017 (Reference 17) specifies the release rate per year per unit per building ventilation system. This is not dependent on power level. Thus, there was no change calculated for the uprate. However, a 12-percent increase was conservatively assumed and particulates were treated similar to the iodines.

Solid Radioactive Waste

Though solid radwaste is not specifically addressed in 10CFR50, Appendix I (Reference 9), for the purposes of a complete radwaste assessment, the effect of the core uprate on solid radwaste generation is summarized below.

For a new facility, the estimated volume and activity of solid waste is linearly related to the core power level. However, for an existing facility that is undergoing a power uprate, the volume of solid waste would not be expected to increase proportionally, since the power uprate would neither appreciably impact installed equipment performance nor require drastic changes in system operation or maintenance. Only minor, if any, changes in waste generation volume would be expected. However, it is expected that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity bounded by the effective increase in core power level, that is, 11.5 percent.

Thus, while the total long-lived activity contained in the waste is expected to be bounded by the percentage for the uprate (11.5 percent), the increase in the overall volume of waste generation resulting from the uprate is expected to be minor.

In summary, and as documented in Table 6.11-8, the estimated doses due to annual radwaste effluent releases following the power uprate remain a small percentage of the allowable 10CFR50 Appendix I (Reference 9) limits. Therefore, it has been concluded that, following the uprate, the liquid and gaseous radwaste effluent treatment system will remain capable of maintaining normal operation offsite doses within the requirements of 10CFR50, Appendix I (Reference 9).

6.11.7 Post-Accident Access to Vital Areas

6.11.7.1 Introduction

In accordance with NUREG-0578, 2.1.6.b (Reference 19) and NUREG-0737, II.B.2 (Reference 20), the vital areas are those within the station that will or may require access/ occupancy to support accident mitigation or recovery following a LOCA. In accordance with the above regulatory documents, all vital areas and access routes to vital areas must be designed so that operator exposure remains within regulatory limits.

This section focuses on those areas that may require occasional access following a LOCA. Areas that require continuous occupancy, such as the control room, are addressed later in subsection 6.11.9 of this report.

The design basis vital access shielding review that supports IP2 licensing basis relative to post-LOCA vital access dose rates/doses is documented in EDS Nuclear Report No. 02-0180-1026, *Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/ Systems which may be used in Post Accident Operations,* December 1979 (Reference 21). To lower the dose rates estimated by the EDS study to acceptable limits, IP2 performed several system modifications and installed additional shielding. NRC review of the EDS Report and plant modifications is documented in NRC *Shielding Design Inspection Report* 50-247/83-14 (Reference 22). In response to the Inspection Report, post-modification dose rate assessments were performed.

The *IP2 Updated Final Safety Analysis Report* (UFSAR) (Reference 23) indicates that the EDS Report (Reference 21) is still the design basis document of record relative to post-accident vital area access. Therefore, it was concluded that the licensing basis list of target areas/dose estimates evaluated for purposes of estimating the mission dose is documented in Table 10 of the EDS Report (Reference 21) and the post-modification dose rate assessments that updated some of the dose estimates in the EDS Report and added four more locations.

The power uprate will affect the equilibrium core inventory and, therefore, the post-accident radiological source terms. Additional factors that can alter the equilibrium core inventory are fuel enrichment and burnup.

The vital access dose assessment for the stretch power (SPU) uprate utilizes scaling techniques and TID (14844) source terms.

6.11.7.2 Description of Analysis and Evaluations

The effect of the power uprate on the radiation doses received while accessing or occupying vital areas during post-LOCA conditions was evaluated based on a comparison of the original design basis source terms to the power uprate source terms.

The effect of the power uprate on the post-LOCA gamma radiation dose rates was evaluated by comparing gamma source terms. The pre-uprate source terms (derived from the original core inventory used to develop the post-LOCA dose rates for the vital access review) were compared to the gamma source terms that were determined based on the uprate core inventory. The approach used scaling techniques based on a source term comparison, rather than using the new core inventory to develop new dose rate estimates at the various locations.

The core power uprate will increase the activity level in the core by the percentage of the uprate. The estimated radiation source terms in equipment/structures containing post-accident fluids and the corresponding post-LOCA environmental dose rates will increase by the percentage of

the uprate relative to the power level used in the analyses of record. Additional factors that can impact the equilibrium core inventory and, consequently, the estimated operator dose, are fuel enrichment and burnup. Theoretically, with all things being equal, the post-LOCA environmental gamma dose rates and the operator dose per identified mission should increase by approximately 24 percent:

$$3280.3 \text{ MWt} \times 1.04 / 2758 \text{ MWt} = 1.24$$

Note: The 1.04 multiplier was applied to the SPU inventory by Westinghouse to account for variation in fuel design parameters.

However, the calculated uprate scaling factor values deviated from the core power ratio because the uprated core reflected extended burnup and the more advanced fuel burnup modeling/libraries used to develop the uprated core (compared to the computer code used in the original analyses).

The calculation was essentially a two-step process. The first part of the calculation developed a bounding SPU dose rate scaling factor, and the second part multiplied the pre-uprate personnel dose/dose rates at personnel target areas identified in the licensing basis by the bounding SPU scaling factor.

Radiological source terms for both the pre-uprate and the uprate cases were developed for the following post-accident sources.

- Containment atmosphere (100-percent core noble gases, 25-percent core halogens) sampled by the Containment Air Sampling System

- Sump water/reactor coolant liquid (100-percent core noble gases, 50-percent core halogens and 1 percent of core remainder) assumed to be in the Chemical and Volume Control System (CVCS), the Residual Heat Removal System (RHRS), the Containment Spray System (CSS), the Safety Injection System (SIS), and the RCS Sampling System

- 100-percent noble gas only assumed to be in the Gaseous Waste Disposal System (GWDS)

For the "unshielded" case, the factor effect on post-accident gamma dose rates was estimated by ratioing the gamma energy release rates (weighted by the flux-to-dose rate conversion factor as a function of time) for the uprate power level to the corresponding weighted source terms based on the pre-uprate power level. To address outside containment locations, the unshielded values included the shielding effect of a 2-inch nominal diameter pipe wall thickness. This

ensured that the results were not skewed by photons at energies less than 25 keV, which would be substantially attenuated by any piping.

To evaluate the factor effect of the power uprate on post-LOCA gamma dose rates (versus time) in areas that were shielded, the pre-uprate and the uprate source terms discussed above were weighted by the concrete shielding factors for each energy group. The concrete shielding factors, for 1 and 3 feet of concrete, provided a basis for comparing the post-LOCA spectrum hardness of source terms with respect to time for both the original design and the power uprate cases.

The uprate gamma dose rate scaling factors varied with source, time, and shielding. Therefore, to cover all types of analysis models/assessments, the maximum dose rate scaling factor developed from all of the above assessments was used. This applied, for the most part, to all source/receptor combinations, with or without shields, and at all time periods after LOCA.

6.11.7.3 Acceptance Criteria

1. When the pre-uprate vital area assessment establishes operator mission doses, demonstrate continued compliance with the operator exposure dose limits of 5 rem noted in NUREG-0578, Item 2.1.6.b and NUREG-0737, II.B.2 & II.B.3 (References 19 and 20) based on uprated conditions. Specifically, the pre-uprate calculated doses at any target location identified in the IP2 licening basis that are currently less than 5 rem whole body should not exceed 5 rem whole body following the SPU.

2. When the pre-uprate vital area assessment establishes radiation levels in the area, but does not develop operator mission doses, develop the estimated radiation levels following the uprate. There are no acceptance criteria for this case. The licensing basis for such cases is availability of the radiation dose rate information such that the licensee can factor this information into any post-accident access planning.

6.11.7.4 Results and Conclusions

The bounding SPU scaling factor for vital access dose rates was 1.36. This scaling factor included: a factor of 1.19 to address the impact of increased power from 2758 MWt (basis of current analysis) to 3280.3 MWt; a multiplier of 1.04, which was applied by Westinghouse to the SPU core activity to account for variation in fuel design parameters; and a factor of 1.10, which accounted for the effects of extended burn fuel, ORIGEN code updates, and the increase in the number of nuclides addressed in the updated core.

The evaluation concluded that following the power uprate, the post-LOCA vital area operator dose estimates will remain within the regulatory limits of 5 rem whole body in NUREG-0578, Item 2.1.6.b and NUREG-0737 II.B.2 and II.B.3 (References 19 and 20) for all target areas that currently meet this requirement at the pre-uprate conditions.

6.11.8 Radiological Environmental Qualification

6.11.8.1 Introduction

In accordance with 10CFR50.49 (Reference 24), safety-related electrical equipment must be qualified to survive the radiation environment at its specific location during normal operation and during an accident.

The effect of the power uprate on the normal operation radiation environmental dose estimates supporting environmental qualification is summarized in subsection 6.11.5. The effect of the power uprate on the post-accident radiation environmental dose estimates is discussed below. For completeness, this section will also include the conclusions of the normal operation evaluation developed in subsection 6.11.5.

Post-accident environmental doses are usually developed based on the equilibrium core inventory assuming full-power operation at the licensed power level plus margin, source term guidance (available from regulatory documents relative to post-accident core releases), and plant-specific mitigation system design features/layout. The power uprate affects the equilibrium core inventory and, therefore, the post-accident radiological source terms. Additional factors that can alter the equilibrium core inventory are fuel enrichment and burnup.

For purposes of equipment qualification, IP2 was divided into various environmental zones. The radiological environmental conditions noted for these zones were the maximum conditions expected to occur and were representative of the whole zone. *The IP2 Environmental Qualification Program Manual* (Reference 15) notes that the normal operation values represent 40 years of operation and that the post-accident component of the environmental dose is based on a 1-year integrated dose following a LOCA.

Accident Environments

In-Containment Radiation Levels

The IP2 in-containment post-accident environmental gamma and beta dose was developed based on the Division of Operating Reactors (DOR) Guidelines (Reference 25), that is, the dose values provided in the EQ Program Manual (Reference 15) for power levels 2758 MWt,

3071.4 MWt, and 3216 MWt, are based on utilization of a power level ratio on the gamma and beta dose estimates provided in Appendix D of RG 1.89, Revision 1 (Reference 25) for a 4100-MWt PWR.

Outside Containment Radiation Levels

The current post-accident environmental gamma dose outside containment was developed in 1989 by IP2 using ORIGEN2 and a core power level of 3216 MWt (Reference 15).

The IP2 EQ Manual (Reference 15) indicates that, per the current IP2 design basis, beta doses are not applicable in environmental zones outside containment. Implicit in this position is the assumption that for equipment outside containment, beta radiation is effectively attenuated to negligible values because the radioactive fluid is completely contained within the stainless steel piping.

Normal Operation Environments

As discussed previously in subsection 6.11.5, the 40-year normal operation dose component of the total EQ dose is based on survey data associated with plant operation at 2758 MWt, which is scaled-up to reflect various power levels including 3216 MWt.

6.11.8.2 Description of Analysis and Evaluation

The effect of the power uprate on the post-accident component of the total integrated environmental dose was estimated using scaling techniques and TID-14484 (Reference 12) source term.

The EQ Manual currently presents dose values at a power level of 3216 MWt. The SPU accident EQ dose scaling factor has three components: a multiplier of 1.02 to address instrument uncertainty, a factor of 1.04 to address the NSSS uncertainty for fuel management variations, and a factor to update the current values to reflect the 24-month fuel cycle.

As noted earlier, the current radiation environment inside containment is based on Appendix D of RG 1.89, Revision 1 (Reference 25), which is representative of a 12-month fuel cycle. The current radiation environment outside containment is based on an 18-month fuel cycle. The uprated core inventory is based on a 24-month fuel cycle.

To obtain the dose scaling factor to address the change in fuel cycle length, three typical PWR core activities were generated using the ORIGEN-S computer code: for 12-month, 18-month, and 24-month fuel cycles using parameters reasonably representative of IP2. Core average

enrichments were used in this assessment. Based on the vintage of RG 1.89, Revision 1, Appendix D (Reference 25), which is the basis of the inside containment doses, for the 12-month cycle, a 3.2-percent enrichment was used. For the 18-month cycle an enrichment of 5 percent was used. The enrichment for the 24-month fuel cycle was calculated based on uprated conditions. The core activities were then used to develop integrated doses versus time from a finite cloud model. No depletion other than decay was considered. The fuel cycle length scaling factors were the ratio of the finite cloud gamma and beta dose based on the 24-month fuel cycle versus the 12-month fuel cycle (for inside containment) and the 24-month fuel cycle versus 18-month fuel cycle (for outside containment).

Note that for the inside containment assessment, the noble gas and iodine doses and associated scaling factors were developed separately. This was done to account for the fact that sprays and plateout were credited in RG 1.89 (Reference 25) and, therefore, inherently considered in the current inside containment dose estimates. The fuel cycle length gamma and beta dose scaling factors associated with the iodines and noble gases, therefore, were applied to the fraction of the total gamma and beta dose attributable to iodines and noble gases, respectively, as documented in Tables D1 and D2 of RG 1.89 (Reference 25).

To evaluate the effect of attenuation on the post-LOCA gamma dose rates (versus time) in areas outside containment that are shielded, the pre-uprate, as well as the uprate, dose rates address concrete reduction factors for each energy group. For the 3 fuel burnup cycles (that is, 12, 18, and 24 months), the concrete reduction factors for 1 and 3 feet of concrete were used to provide a basis for comparison of the post-LOCA spectrum hardness of source terms, to address lightly shielded and heavily shielded conditions. Since all sources outside containment are housed in equipment casing or pipes, the unshielded values, outside containment, included the shielding effect of a 2-inch nominal diameter pipe. To model this geometry, a 2-inch zone of water was inserted between the point source and detector to account for this self-shielding effect. This trace amount of shielding ensured that the results would not be skewed by photons at energies less than 25 keV, which would be substantially attenuated by any piping or equipment wall. The most conservative scaling factor results between the unshielded and shielded case were used.

6.11.8.3 Acceptance Criteria

The equipment in the IP2 EQ Program must be qualified to actively function and/or not impair other equipment relied on to perform an active safety function in the radiation environment to which they are exposed during normal operation, as well as for the duration of the accident. This section focuses on the development of the estimated radiation environments following SPU.

6.11.8.4 Results and Conclusions

For the SPU, the normal operation component of the EQ dose will utilize the value currently noted in the EQ Manual (Reference 15) for a power level of 3216 MWt.

The SPU environmental levels for accident conditions were determined using scaling techniques. The accident gamma and beta dose scaling factors utilized to adjust the current accident environmental levels inside containment (that are presently based on a 12-month fuel cycle) and the current accident environmental levels outside containment (that are presently based on an 18-month fuel cycle) ranged from 1 to 1.23, and were based on source and shielding, as well as whether the location was inside or outside containment.

6.11.9 Radiological Consequences Evaluations (Doses)

6.11.9.1 Introduction

The radiological consequences for the following DBAs were reanalyzed to support the SPU Program:

- Main steamline break (MSLB)
- Locked reactor coolant pump (RCP) rotor
- Rod ejection
- Steam generator tube rupture (SGTR)
- Small-break loss-of-coolant accident (SBLOCA)
- Large-break loss-of-coolant accident (LBLOCA)
- Waste gas decay tank (GDT) rupture
- Volume control tank (VCT) rupture
- Holdup tank (HT) failure
- Fuel-handling accident (FHA)

The radiological consequences analyses for all of these accidents (except for the failures of the VCT, GDT, and HT) have been approved by the NRC (Reference 4) for the implementation of Alternate Source Term (AST) for IP2, using the analytical methods and assumptions outlined in NUREG-1465 (Reference 7). IP2 was a pilot plant for the application of AST. Since that time, RG 1.183 (Reference 3) has been approved for use as the methodology document for AST application.

The analyses performed for the SPU Program follow the methodology outlined in RG 1.183 (Reference 3). The implementation of the RG 1.183 methodology affects the assumed gap fractions, activity in failed fuel in the rod ejection and locked rotor accidents, and the gap activity

release timing in the LOCA. The analyses have been updated using input assumptions consistent with the proposed nominal core power of 3216 MWt and are presented in this section. Changes in the Technical Specification for primary-to-secondary leak rate have also been incorporated.

The radiological consequences analyses for the VCT, GDT and HT failures were not included in the analyses approved by the NRC in Reference 4. These analyses were analyzed with AST and were included in the IP2 UFSAR (Reference 23).

For each accident, the TEDE doses are determined at the site boundary (SB) for the limiting 0- to 2-hour period, at the LPZ boundary for the duration of the accident and in the control room for 30 days.

6.11.9.1.1 Input Parameters and Assumptions

The assumptions and inputs described in this section are common to various analyses discussed in the following sections. These assumptions and inputs are consistent with those approved by the NRC (Reference 4) except as revised to follow RG 1.183 (Reference 3) or to reflect plant operation at the uprated power. Each accident and the specific input assumptions are described in detail in subsections 6.11.9.2 through 6.11.9.11.

The TEDE dose is equivalent to the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. Effective dose equivalent (EDE) is used in lieu of DDE in determining the contribution of external dose to the TEDE consistent with RG 1.183 (Reference 3) guidance. The dose conversion factors (DCFs) used in determining the CEDE dose are from the Environmental Protection Agency (EPA) Federal Guidance Report No. 11 (Reference 26). The DCFs used in determining the EDE dose are from the EPA Federal Guidance Report No. 12 (Reference 27). The nuclide decay constants are from EPA Federal Guidance Report No. 11 (Reference 26). The nuclide data are listed in Table 6.11-9.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table 6.11-10.

The RG 1.183 (Reference 3) offsite dose acceptance limits are based on 10CFR50.67 (Reference 1) guidance of 25 rem TEDE. Depending on the event, the acceptance limit is 100 percent of 10CFR50.67 or a fraction of these guidelines. Some events are designated as having a dose limit that is 25 percent of that limit (6.3 rem TEDE); other events are specified with a dose limit that is 10 percent of that limit (2.5 rem TEDE).

Parameters modeled in the control room personnel dose calculations are provided in Table 6.11-11. These parameters include normal operation flow rates, emergency operation flow rates, control room volume, filter efficiencies, and control room operator breathing rates. Atmospheric dispersion factors are event-dependent and are listed together with the assumptions for each accident. The control room dose acceptance limit from 10CFR50.67 (Reference 1) is 5 rem TEDE.

Section 6.11.4 of this report describes the calculation of the core and coolant activity. The core fission product activity modeled in the radiological consequences analyses for the locked rotor, rod ejection, SBLOCA, and LBLOCA is provided in Table 6.11-12, and was calculated modeling the third transition cycle. To accommodate variations in fuel design and fuel management, a multiplier of 1.04 was applied to the core inventory. The core activity data in Table 6.11-12 include this multiplier. The nominal reactor coolant activity based on 1-percent fuel defects is provided in Table 6.11-13. A 1.04 multiplier was applied to the coolant activity. The reactor coolant and secondary coolant iodine activity modeled in the radiological consequences analyses, based on dose equivalent I-131 (DE I-131), is provided in Table 6.11-14.

6.11.9.1.2 Iodine Spiking Models

A number of accident analyses take iodine spiking into consideration (for example, MSLB and SGTR).

For the pre-existing iodine spike, it was assumed that a reactor transient occurs prior to the accident and raises the primary coolant iodine concentration to 60 μCi/gm of DE I-131. (This is the Technical Specification limit for transient elevated iodine activity in the primary coolant.) For the accident-initiated iodine spike, it was assumed that the reactor trip associated with the accident creates an iodine spike, which increases the iodine release rate from the fuel to the reactor coolant. The spike iodine release rate is a multiple of the maximum equilibrium release rate (where the equilibrium release rate is that rate corresponding to maintaining a primary coolant concentration of 1.0 μCi/gm of DE I-131, which is the maximum concentration allowed by the Technical Specifications for continuous operation). RG 1.183 (Reference 3) requires a spike multiplier of 500 for the steamline break, and allows a multiplier of 335 for SGTR.

The primary coolant iodine concentrations associated with a pre-existing iodine spike are provided in Table 6.11-14, and the iodine appearance rates associated with an accident-initiated iodine spike are provided in Table 6.11-15.

6.11.9.2 MSLB Radiological Consequences

In this analysis, a complete severance of a main steamline outside containment is assumed to occur. The affected steam generator rapidly depressurizes and releases iodine activity initially contained in the secondary coolant and primary coolant activity (iodines and noble gases) transferred via steam generator tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact steam generators and the activity transferred to the secondary coolant due to tube leakage is released to the atmosphere through either the atmospheric relief valves (ARVs) or the safety valves. The steamline break outside containment bounds any break inside containment since the outside containment break provides a means for direct release to the environment. This section describes the assumptions and analyses performed to determine the offsite and control room doses resulting from the release of activity associated with this event.

6.11.9.2.1 Input Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 6.11-16.

The analytical methods and assumptions outlined in RG 1.183 (Reference 3) were used in the analysis of the MSLB radiological consequences. This is consistent with the analysis approved in Reference 4, with changes made to reflect the increased power. The activity available for release to the environment included the iodine assumed to be initially present in the secondary coolant and the activity in the primary coolant (both iodine and noble gases) that could leak into the secondary coolant due to steam generator tube leakage.

Source Term

The iodine activity concentration of the secondary coolant at time an MSLB occurs was assumed to be equivalent to the Technical Specification limit of 0.15 μCi/gm of DE I-131.

The MSLB event was analyzed for two iodine spiking cases: one in which there is a pre-existing iodine spike resulting in elevated primary coolant activity, and the other in which an iodine spike is assumed to be initiated by the accident. For the pre-accident iodine spike case, it was assumed that a reactor transient occurs prior to the MSLB, and raises the Reactor Coolant System (RCS) iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of DE I-131. For the accident-initiated iodine spike case, the reactor trip associated with the MSLB creates an iodine spike in the RCS that increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap.

Based on having 8 percent of the iodine in the fuel-cladding gap, the gap inventory is depleted within 3 hours, and the accident-initiated spike is terminated at that time.

The noble gas activity concentration in the RCS at the time the accident occurs is based on operation with a fuel defect level of 1.0 percent.

Release Pathway

The primary-to-secondary steam generator tube leakage rate was assumed to be at the Technical Specification limit of 150 gpd/steam generator.

The steam generator connected to the broken steamline was assumed to boil dry within 5 minutes following the MSLB. The entire liquid inventory of this steam generator was assumed to be steamed off and all of the iodine initially in this steam generator released to the environment. Also, iodine carried over to the faulted steam generator by tube leakage was assumed to be released directly to the environment, with no credit taken for iodine retention in the steam generators.

An iodine-partition factor in the intact steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) was used. Prior to reactor trip and concurrent loss-of-offsite power (LOOP), an iodine-removal factor of 0.01 could be taken for steam released to the condenser, but this was conservatively ignored.

All noble gas activity carried over to the secondary side through steam generator tube leakage was assumed to be immediately released to the outside atmosphere.

At 30 hours after onset of the accident, the Residual Heat Removal System (RHRS) was assumed to remove all decay heat, and there were no further steam releases to the atmosphere from the intact steam generators.

Within 65 hours after the event, analysis showed that the RCS had been cooled to below 212°F, and there were no further steam releases to the atmosphere from the faulted steam generator.

No fuel failure (departure from nucleate boiling [DNB] or melt) was calculated to occur for the MSLB event.

Control Room Isolation

In the event of an MSLB, the low steamline pressure safety injection (SI) setpoint will be reached almost immediately after event initiation. The SI signal causes the control room

heating, ventilation, and air conditioning (HVAC) to switch from the normal-operation mode to the emergency mode of operation. It was conservatively assumed that the control room HVAC would not fully enter the emergency mode of operation until 1 minute after event initiation.

6.11.9.2.2 Acceptance Criteria

The offsite dose limit for an MSLB with a pre-accident iodine spike is 25-rem TEDE per RG 1.183 (Reference 3), which is also the guideline value of 10CFR50.67. For an MSLB with an accident-initiated iodine spike, the offsite dose limit is 2.5-rem TEDE per RG 1.183 (Reference 3). This is 10 percent of the guideline value of 10CFR50.67 (Reference 1). The limit for the control room dose is 5.0-rem TEDE per 10CFR50.67.

6.11.9.2.3 Results and Conclusions

The calculated doses due to the MSLB with a pre-existing iodine spike were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Pre-Accident Iodine Spike - SB	0.12	25
Pre-Accident Iodine Spike - LPZ	0.13	25
Pre-Accident Iodine Spike - Control Room	0.18	5

The calculated doses due to the MSLB with an accident-initiated iodine spike were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Accident-Initiated Iodine Spike - SB	0.12	2.5
Accident-Initiated Iodine Spike - LPZ	0.33	2.5
Accident-Initiated Iodine Spike - Control Room	0.52	5

The acceptance criteria were met.

The SB doses reported are, for the worst 2-hour period, determined to be from 0 to 2 hours for the pre-accident iodine spike, and from 3 to 5 hours for the accident-initiated iodine spike.

6.11.9.3 Locked Rotor Accident

In this analysis, an instantaneous seizure of an RCP rotor is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop (RCL). Fuel-cladding damage could be predicted as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products transfer from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the ARVs or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident, and some of this activity is assumed to be released to the atmosphere as a result of steaming from the steam generators following the accident.

6.11.9.3.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 6.11-17.

The analysis of the locked rotor radiological consequences was performed using the analytical methods and assumptions outlined in RG 1.183 (Reference 3) and, therefore, incorporated different gap activity assumptions than the analysis approved in Reference 4, in addition to the changes made to reflect the increased power.

Source Term

The analysis of the locked rotor radiological consequences assumed a pre-existing iodine spike in the RCS. For the pre-existing iodine spike, it was assumed that a reactor transient had occurred prior to the event that raised the RCS iodine concentration to 60 μCi/gm of DE I-131.

The noble gas and alkali metal activity concentration in the primary coolant when the postulated accident occurs is based on a fuel defect level of 1 percent. The iodine activity concentration of the secondary coolant when the locked rotor occurs is assumed to be 0.15 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be 15 percent of the primary side concentration.

The transient analysis performed for the uprate (subsection 6.3.14 of this report) shows that no rods in DNB are calculated for the locked rotor event. However, it was conservatively assumed that 5 percent of the fuel rods in the core suffered damage sufficient that all of their gap activity was released to the RCS. Eight percent of the total I-131 activity, 10 percent of the total Kr-85 activity, 5 percent of the total activity for other noble gases and other iodines, and 12 percent of the total activity for alkali metals were assumed to be in the fuel-cladding gap and released into the primary coolant. In the calculation of the activity releases from the failed fuel, the maximum radial peaking factor of 1.7 was applied.

Release Pathway

Activity is released to the environment by way of primary-to-secondary leakage and steaming from the secondary side to the environment. The primary-to-secondary steam generator tube leakage rate was assumed to be at the Technical Specification limit of 150 gpd/steam generator.

The RHRS was assumed to remove all decay heat 30 hours into the accident, with no further releases to the environment after that time.

An iodine-partition factor in the steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) was used. Prior to reactor trip and concurrent LOOP, an iodine-removal factor of 0.01 could have been taken for steam released to the condenser, but this was conservatively ignored.

The retention of particulates in the steam generators was limited by moisture carryover (MCO). The MCO was 0.25 percent, therefore, an alkali metal partition factor in the steam generators of 0.0025 (curies alkali metal/gm steam)/(curies alkali metal/gm water) was used.

All noble gas activity carried over to the secondary side through steam generator tube leakage was assumed to be immediately released to the outside atmosphere.

Control Room Isolation

It was assumed that the control room HVAC System begins in normal-operation mode, and as activity builds up in the control room, a high-radiation signal is generated. It was conservatively assumed that the control room HVAC does not fully enter the emergency mode of operation until 10 minutes after the high radiation signal. For this analysis, this was modeled at 21 minutes.

6.11.9.3.2 Acceptance Criteria

The offsite dose limit for a locked rotor accident is 2.5-rem TEDE per RG 1.183 (Reference 3). This is 10 percent of the guideline value of 10CFR50.67 (Reference 1). The limit for the control room dose is 5-rem TEDE, per 10CFR50.67.

6.11.9.3.3 Results and Conclusions

The calculated doses due to the locked rotor event were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.24	2.5
LPZ	0.54	2.5
Control Room	0.65	5

The acceptance criteria were met.

The SB dose reported was for the worst 2-hour period, determined to be from 6 to 8 hours after event initiation.

6.11.9.4 Rod Ejection Accident

For this analysis, it is assumed that a control rod drive mechanism (CRDM) pressure housing mechanical failure occurs, resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melting (pellet centerline) are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive primary coolant is assumed to leak from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the main condenser, the ARVs, or the safety valves. Also, iodine and alkali metal group activity is contained in the secondary coolant prior to the accident, and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the postulated accident. Finally, radioactive primary coolant is discharged to the containment via spill from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

6.11.9.4.1 Input Parameters and Assumptions

Separate calculations were performed to calculate the dose resulting from the release of activity to containment and subsequent leakage to the environment, and the dose resulting from the leakage of activity to the secondary system and subsequent release to the environment. The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths and nuclides considered.

A summary of input parameters and assumptions is provided in Table 6.11-18.

The analysis of the rod ejection radiological consequences was performed using the analytical methods and assumptions outlined in RG 1.183 (Reference 3), and, therefore, incorporated different gap/fuel activity assumptions than the analysis approved by Reference 4, in addition to the changes made to reflect the increased power. Sedimentation removal of particulates in containment was credited in this rod ejection analysis, although it was only credited for LOCAs in the analysis approved in Reference 4.

Source Term

The assumption is that less than 10 percent of the fuel rods in the core undergo DNB as a result of the rod-ejection accident. In determining the offsite doses following a rod-ejection accident, it was conservatively assumed that 10 percent of the fuel rods in the core suffer sufficient damage such that all of their gap activity is released. Ten percent of the total core activity of iodine and noble gases, and 12 percent of the total core activity for alkali metals were assumed to be in the fuel-cladding gap. In the calculation of activity released from the failed/melted fuel, the maximum radial peaking factor of 1.7 was applied.

A small fraction of the fuel in the failed fuel rods is assumed to melt as a result of the rod-ejection accident. This amounts to 0.25 percent of the core, and the melting took place in the centerline of the affected rods. Of the rods that entered DNB, 50 percent are assumed to experience some fuel melting (5.0 percent of the core). Of the rods that experience melting, 50 percent of the axial length of the rod is assumed to melt (2.50 percent of the core). It is further assumed that only 10 percent of the radial portion of the rod melts (0.25 percent of the total core).

For both the containment leakage release path and the primary-to-secondary leakage release path, it is assumed that all noble gas and alkali metal activity released from the failed fuel (both gap activity and melted fuel activity) is available for release. For the containment leakage release path, it is assumed that all of the iodine released from the gap of failed fuel and 25 percent of the activity released from melted fuel is available for release from containment. And, for the primary-to-secondary leakage release path, it is assumed that, all of the iodine released from the gap of failed fuel and 50 percent of the activity released from melted fuel is available for release from the RCS.

Prior to the postulated accident, the iodine activity concentration of the primary coolant is 1.0 μCi/gm of DE I-131. The noble gas and alkali metal activity concentrations in the RCS when the rod ejection accident is postulated to occur are based on operation with a fuel-defect level of 1 percent. Further, the iodine activity concentration of the secondary coolant is assumed to be equivalent to 0.15 μCi/gm of DE I-131, and the alkali metal activity concentration of the secondary coolant is assumed to be 15 percent of the primary side concentration.

Iodine Chemical Form

Iodine in containment was assumed to be 4.85 percent elemental, 0.15 percent organic, and 95 percent particulate. Iodine released from the secondary system was assumed to be 97 percent elemental and 3 percent organic.

Release Pathways

When determining the offsite doses due to containment leakage, all of the RCS iodine, noble gas, and alkali metal activity (from prior to the accident and resulting from the accident) was assumed to be in the containment.

The containment was assumed to leak at the design leak rate of 0.1 percent per day for the first 24 hours of the accident, and then to leak at half that rate (0.05 percent per day) for the remainder of the 30-day period following the accident considered in the analysis.

When determining the offsite doses due to the primary-to-secondary steam generator tube leakage, all of the RCS iodine, noble gas, and alkali metal activity (from before the accident and resulting from the accident) was assumed to be in the primary coolant.

Primary-to-secondary tube leakage and steaming from the steam generators continues until the RCS pressure drops below the secondary pressure. Bounding times of 1 hour of leakage and 2 hours of steaming were selected for this analysis, although the analysis showed that leakage and releases would stop before then.

The primary-to-secondary steam generator tube leakage rate was assumed to be at the Technical Specification limit of 150 gpd/steam generator. Although the primary-to-secondary pressure differential drops throughout the event, a constant leakage rate was assumed.

Removal Coefficients

An iodine-partition factor in the steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) was used. Prior to reactor trip and concurrent LOOP, an iodine-removal factor of 0.01 could be taken for steam released to the condenser, but this was conservatively ignored.

The retention of particulates in the steam generators was limited by MCO, which was 0.25 percent. Therefore, an alkali metal partition factor in the steam generators of 0.0025 (curies alkali metal/gm steam)/(curies alkali metal/gm water) was used.

All noble gas activity carried over to the secondary side through steam generator tube leakage was assumed to be immediately released to the outside atmosphere.

For the containment leakage pathway, no credit was taken for deposition onto containment surfaces or for containment spray operation that would remove airborne particulates and elemental iodine. Sedimentation of iodine and alkali metal particulates in containment was credited, with a removal coefficient of 0.1 hr^{-1} consistent with that approved for LOCAs in Reference 4. Retention of iodine in the sump solution was ensured by adjusting the solution to a pH greater than or equal to 7.0. The pH adjustment was provided by trisodium phosphate (TSP) stored in the containment sump. The 8000 lbm of TSP, as provided by the Technical Specifications, remains sufficient for the uprating analyses to ensure a post-accident sump solution pH greater than or equal to 7.0.

Control Room Isolation

The low-pressurizer pressure SI setpoint will be reached within 61 seconds from event initiation. The SI signal causes the control room HVAC to switch from the normal-operation mode to the emergency mode of operation. It was conservatively assumed that the control room HVAC would not fully enter the emergency mode of operation until 3 minutes after event initiation.

6.11.9.4.2 Acceptance Criteria

The offsite dose limit for a rod ejection is 6.3-rem TEDE, per RG 1.183 (Reference 3). This is ~25 percent of the guideline value of 10CFR50.67 (Reference 1). The limit for the control room dose is 5-rem TEDE, per 10CFR50.67.

6.11.9.4.3 Results and Conclusions

The calculated doses due to the rod-ejection accident were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	3.1	6.3
LPZ	4.2	6.3
Control Room	1.4	5

The acceptance criteria were met.

The SB dose reported was for the worst 2-hour period, determined to be from 0 to 2 hours.

6.11.9.5 SGTR Accident

The calculation of the thermal-hydraulic results of the SGTR event is given in Section 6.4 of this document.

6.11.9.5.1 Input Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 6.11-19.

The analysis of the SGTR radiological consequences was performed using the analytical methods and assumptions outlined in RG 1.183 (Reference 3) and was consistent with the analysis approved in Reference 4, with changes made to reflect the increased power. The activity available for release to the environment included the iodine assumed to be initially present in the secondary coolant and the activity in the primary coolant (both iodine and noble gases) that could leak into the secondary coolant due to steam generator tube leakage.

The SGTR event was analyzed for two iodine spiking cases: one in which there is a pre-existing iodine spike resulting in elevated primary coolant activity, and the other in which an iodine spike is assumed to be initiated by the accident. For the pre-accident iodine-spike case, it was assumed that a reactor transient occurs prior to the SGTR and raises the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of DE I-131. For the accident-initiated iodine-spike case, it is assumed that the reactor trip associated with the SGTR creates an iodine spike in the RCS, which increases the iodine release rate from the fuel to the RCS to a value 335 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 8 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted within 4 hours, and the accident-initiated spike is terminated at that time.

The noble gas activity concentration in the RCS at the time the SGTR accident occurs is based on operation with a fuel-defect level of 1 percent. The iodine activity concentration of the secondary coolant at that time is assumed to be equivalent to the Technical Specification limit of 0.15 μCi/gm of DE I-131.

Release Pathway

Break-flow flashing fractions and steam-release rates from the intact and ruptured steam generator were calculated. The amount of break flow that flashes to steam was conservatively calculated assuming that all break flow is from the hot-leg side of the break and that the primary temperatures remain constant.

The break flow, flashed break flow, and steam release data presented in Table 6.4-2 of Section 6.4 of this document were used for the offsite and control room dose analysis.

The intact steam generator primary-to-secondary steam generator tube leakage rate was assumed to be at the Technical Specification limit of 150 gpd per steam generator.

An iodine-partition factor in the steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) was used. Prior to reactor trip and concurrent LOOP, an iodine-removal factor of 0.01 was taken for steam released to the condenser.

All noble gas activity carried over to the secondary side through steam generator tube leakage was assumed to be immediately released to the outside atmosphere.

At 30 hours after the accident, the RHRS is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system.

Control Room Isolation

The low-pressurizer pressure SI setpoint will be reached at 4.83 minutes from event initiation. The SI signal causes the control room HVAC to switch from the normal-operation mode to the emergency mode of operation. It was conservatively assumed that the control room HVAC would not fully enter the emergency mode of operation until 5.83 minutes after event initiation.

6.11.9.5.2 Acceptance Criteria

The offsite dose limit for a SGTR with a pre-accident iodine spike is 25-rem TEDE per RG 1.183 (Reference 3), which is also the guideline value of 10CFR50.67 (Reference 1). For an SGTR with an accident-initiated iodine spike, the offsite dose limit is 2.5-rem TEDE per RG 1.183 (Reference 3). This is 10 percent of the guideline value of 10CFR50.67. The limit for the control room dose is 5-rem TEDE per 10CFR50.67.

6.11.9.5.3 Results and Conclusions

The calculated doses due to the SGTR with a pre-existing iodine spike were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Pre-Accident Iodine Spike - SB	3.24	25
Pre-Accident Iodine Spike - LPZ	1.52	25
Pre-Accident Iodine Spike - Control Room	1.36	5

The calculated doses due to the SGTR with an accident-initiated iodine spike were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Accident-Initiated Iodine Spike - SB	1.12	2.5
Accident-Initiated Iodine Spike - LPZ	0.55	2.5
Accident-Initiated Iodine Spike - Control Room	0.48	5

The acceptance criteria were met.

The SB doses reported were for the worst 2-hour period, determined to be from 0 to 2 hours.

6.11.9.6 SBLOCA

An abrupt failure of the primary coolant system was assumed to occur and it was assumed that the break would be small enough that the containment spray system would not be actuated by high containment pressure, but that the core would experience substantial cladding damage such that the fission product gap activity of all fuel rods would be released. Activity that is released to the containment is assumed to be released to the environment due to the containment leaking at its design rate. There is also a release path through the steam generators (primary to secondary) until the primary system becomes depressurized to below the secondary system pressure.

6.11.9.6.1 Input Parameters and Assumptions

Separate calculations were performed to calculate the dose resulting from the release of activity to containment and subsequent leakage to the environment, and the dose resulting from the leakage of activity to the secondary system and subsequent release to the environment. The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths and nuclides considered.

A summary of input parameters and assumptions is provided in Table 6.11-20.

The analysis of the SBLOCA radiological consequences was performed using the analytical methods and assumptions credited in RG 1.183 (Reference 3), and the analysis approved in Reference 4, with changes made to reflect the increased power. Deposition of elemental iodine, although credited in the previous analysis with a coefficient of 1.5 hr^{-1}, was conservatively neglected in this analysis.

Source Term

In determining the offsite doses following a SBLOCA, it is assumed that all of the fuel rods in the core suffer sufficient damage so that all of their gap activity is released and no fuel in the core melts. Five percent of the total core activity of iodines, noble gases, and alkali metals are in the fuel-cladding gap.

It is assumed that for both the containment leakage release path and the primary-to-secondary leakage release path all iodine, noble gas, and alkali metal activity in the failed fuel gap is available for release.

Prior to the accident, it is assumed that the iodine activity concentration of the primary coolant is 1.0 μCi/gm of DE I-131. The noble gas and alkali metal activity concentrations in the RCS when the postulated accident occurs are based on operation with a fuel-defect level of 1 percent. The iodine activity concentration of the secondary coolant when the SBLOCA occurs is assumed to be 0.15 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time of SBLOCA is assumed to be 15 percent of the primary side concentration.

Iodine Chemical Form

Iodine in containment is assumed to be 4.85 percent elemental, 0.15 percent organic, and 95 percent particulate. Iodine released from the secondary system is assumed to be 97 percent elemental and 3 percent organic.

Release Pathways

When determining the offsite doses due to containment leakage, all of the RCS iodine, noble gas, and alkali metal activity (from prior to the accident and resulting from the accident) is assumed to be in the containment.

The containment is assumed to leak at the design-leak rate of 0.1 percent per day for the first 24 hours of the accident, and then to leak at half that rate (0.05 percent per day) for the remainder of the 30-day period following the accident considered in the analysis.

When determining the offsite doses due to the primary-to-secondary steam generator tube leakage, all of the RCS iodine, noble gas, and alkali metal activity (from before the accident and resulting from the accident) is assumed to be in the primary coolant.

Primary-to-secondary tube leakage and steaming from the steam generators continue until the RCS pressure drops below the secondary pressure. Bounding times of 1 hour of leakage and 2 hours of steaming were selected for this analysis, although the analysis shows that leakage and releases would stop before then.

The primary-to-secondary steam generator tube leakage rate was assumed to be at the Technical Specification limit of 150 gpd/steam generator. Although the primary-to-secondary pressure differential drops throughout the event, a constant leakage rate was assumed.

Removal Coefficients

An iodine-partition factor in the steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) was used. Prior to reactor trip and concurrent LOOP, an iodine-removal factor of 0.01 could be taken for steam released to the condenser, but this was conservatively ignored.

The retention of particulates in the steam generators is limited by MCO, which was 0.25 percent. Therefore, an alkali metal partition factor in the steam generators of 0.0025 (curies alkali metal/gm steam)/(curies alkali metal/gm water) could be used. However, a conservative alkali metal partition factor of 0.01 was used instead.

All noble gas activity carried over to the secondary side through steam generator tube leakage was assumed to be immediately released to the outside atmosphere.

For the containment leakage pathway, no credit was taken for plateout onto containment surfaces or for containment spray operation that would remove airborne particulates and elemental iodine. Deposition of elemental iodine was not credited. Sedimentation of iodine and

alkali metal particulates in containment was credited with a removal coefficient of 0.1 hr^{-1}, consistent with that approved in Reference 4. Retention of iodine in the sump solution was ensured by adjusting solution to a pH greater than or equal to 7.0. The pH adjustment was provided by TSP stored in the containment sump. The 8000 lbm of TSP, as provided by the Technical Specifications, remains sufficient for the uprating analyses to ensure a post-accident sump solution pH greater than or equal to 7.0.

Control Room Isolation

The low-pressurizer pressure SI setpoint is reached within 61 seconds from event initiation. The SI signal causes the control room HVAC to switch from the normal-operation mode to the emergency mode of operation. It was conservatively assumed that the control room HVAC would not fully enter the emergency mode of operation until 3 minutes after event initiation.

6.11.9.6.2 Acceptance Criteria

The offsite dose limit for a LOCA is 25-rem TEDE, per RG 1.183 (Reference 3). The limit for the control room dose is 5-rem TEDE, per 10CFR50.67 (Reference 1).

6.11.9.6.3 Results and Conclusions

The calculated doses due to the SBLOCA are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	7.8	25
LPZ	10.8	25
Control Room	3.5	5

The acceptance criteria were met.

The SB dose reported was for the worst 2-hour period, determined to be from 0 to 2 hours.

6.11.9.7 LBLOCA

In this analysis, an abrupt failure of a reactor coolant pipe was assumed to occur, and it was also assumed that the emergency core cooling features would fail to prevent the core from experiencing significant degradation (that is, melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical DBA that considers a single active

failure. Activity from the core is released to the containment and then to the environment by containment leakage or leakage from the Emergency Core Cooling System (ECCS) as it recirculates sump solution outside the containment.

6.11.9.7.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.11-21.

The analysis of the LBLOCA radiological consequences was performed using the analytical methods and assumptions outlined in RG 1.183 (Reference 3) and was consistent with the analysis approved in Reference 4, with changes made to reflect the increased power. Other changes from the Reference 4 approved analysis included reduced containment spray flow rates, longer containment fan cooler operation delay, revised containment spray timing, a slight reduction in the credited sump volume, and a shorter time to switch ECCS from internal recirculation to external recirculation.

The analysis considered the release of activity from the damaged core to the containment via containment leakage. In addition, it was assumed that once external recirculation of the ECCS is established, activity in the sump solution can be released to the environment by means of leakage from ECCS equipment outside containment in the Auxiliary Building. The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths. The following sections address topics of significant interest in the analysis.

Source Term

The reactor coolant activity was assumed to be insignificant compared with the release from the core, and was not included in the analysis.

Of the total core activity provided in Table 6.11-12, the following portions were assumed to be released to the containment atmosphere and available for release to the environment via containment leakage:

- 100 percent of the noble gases (Xe, Kr) (5 percent in the gap and 95 percent in the fuel)
- 40 percent of the iodines (5 percent in the gap and 35 percent in the fuel)
- 30 percent of the alkali metals (Cs, Rb) (5 percent in the gap and 25 percent in the fuel)
- 5 percent of the tellurium metals (Te, Sb)
- 2 percent of the barium and strontium
- 0.25 percent of the noble metals (Ru, Rh, Mo, Tc)
- 0.05 percent of the cerium group (Ce, Pu, Np)
- 0.02 percent of the lanthanides (La, Zr, Nd, Nb, Pr, Y, Cm, Am)

The release of activity to containment is assumed to occur over a 1.8-hour interval. The gap activity is released in the first 30 minutes (starting at 30 seconds), and the fraction of the core activity that is released does so over the next 1.3 hours. A gap fraction of 5 percent was assumed for iodines, noble gases, and alkali metals. Gap activity of the other nuclides was not considered. With the exception of the iodines and noble gases, all activity released to containment was modeled as particulates. The iodine in containment was modeled as 4.85 percent elemental, 0.15 percent organic, and 95 percent particulate. For ECCS leakage considerations, the iodine activity that became airborne after being released by the leakage was modeled as 97 percent elemental and 3 percent organic.

For the containment leakage analysis, all activity released from the fuel was assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay, or leakage from the containment. For the ECCS leakage analysis, all iodine activity released from the fuel was assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS.

Containment Modeling

The containment was modeled as two discrete volumes that considered hold-up, removal, and decay. The two discrete volumes were the sprayed containment, which accounted for 80 percent of the free volume, and the unsprayed containment. Mixing between the two volumes was provided by the fan coolers. The analysis credited 3 fan coolers starting 60 seconds after event initiation.

The containment was assumed to leak at the design-leak rate of 0.1 percent per day for the first 24 hours of the accident, and then to leak at half that rate (0.05 percent per day) for the remainder of the 30-day period following the accident considered in the analysis.

Activity Removal from the Containment Atmosphere

Only containment sprays and radioactive decay remove elemental iodine from the containment atmosphere. Containment sprays, sedimentation, and radioactive decay remove particulates from the containment atmosphere. The noble gases and the organic iodine are subject to removal only by radioactive decay.

One train of the Containment Spray System was assumed to operate following the LOCA. Injection spray was credited with a 60-second startup delay. Earlier spray actuation was conservative since it results in earlier spray termination. There was no benefit from the earlier actuation since there was little activity in the containment at the time the sprays start. When the refueling water storage tank (RWST) drains to a predetermined setpoint level, the operators

switch to sump liquid recirculation to provide a source for the sprays. Injection spray was credited for approximately 40 minutes, after which recirculation spray was modeled. The analysis assumed that the recirculation sprays would operate until 3.4 hours after the accident. Retention of iodine in the sump solution was ensured by adjusting the sump solution to a pH greater than or equal to 7.0. The pH adjustment was provided by TSP stored in the containment sump. The 8000 lbm of TSP, as provided by the Technical Specifications, remains sufficient for the uprating analyses to ensure a post-LOCA sump solution pH greater than or equal to 7.0.

Containment Spray Removal of Elemental Iodine

The *Standard Review Plan* (SRP) (Reference 28) identifies a methodology to determine spray removal of elemental iodine. The removal rate constant is determined by:

$$\lambda_s = 6K_g TF/VD$$

where:

λ_s = Elemental iodine removal rate constant due to spray removal, hr^{-1}
K_g = Gas phase mass transfer coefficient, ft/min
T = Time of fall of the spray drops, min
F = Volume flow rate of sprays, ft^3/hr
V = Containment sprayed volume, ft^3
D = Mass-mean diameter of the spray drops, ft

The upper limit specified for this model is 20 hr^{-1}.

The parameters listed below were chosen to bound the current plant configuration:

K_g = 9.84 ft/min
T = 10 sec
F = 2135 gpm
V = 2.088E6 ft^3
D = 0.111 cm

These parameters and appropriate conversion factors were used to calculate the elemental spray removal coefficients. The upper limit of 20 hr^{-1} specified for this model was applied in the analysis in place of the calculated value of 22 hr^{-1}.

The elemental iodine removal rate during recirculation spray operation can be calculated by multiplying the injection spray removal rate (22 hr^{-1}) by the ratio of the recirculation spray flow rate (1080 gpm) to the injection spray flow rate (2135 gpm). The recirculation spray removal rate is then 11.13 hr^{-1}. However, during recirculation, the spray solution will gradually become loaded with elemental iodine that will limit the capacity of the spray to remove airborne iodine. As the DF approaches its defined limit, the removal coefficient would be only a small fraction of its original value. This was approximated by setting the removal coefficient at approximately one half of the calculated value (5.0 hr^{-1}).

Removal of elemental iodine from the containment atmosphere was assumed to be terminated when the airborne inventory (including both sprayed and unsprayed regions) dropped to 0.5 percent of the total elemental iodine released to the containment (this was a DF of 200). With the RG 1.183 (Reference 3) source term methodology, this was interpreted as being 0.5 percent of the total inventory of elemental iodine that was released to the containment atmosphere over the duration of gap and in-vessel release phases. In the analysis, this occurred at 2.75 hours.

Containment Spray Removal of Particulates

Particulate spray removal was determined using the model described in the SRP (Reference 28).

The first order spray removal rate constant for particulates is written as follows:

$$\lambda_p = 3hFE/2VD$$

where:

λ_p = Particulate removal rate constant due to spray removal, hr^{-1}
h = Drop fall height, ft
F = Spray flow rate, ft^3/hr
V = Volume sprayed, ft^3
E = Single drop collection efficiency
D = Average spray drop diameter, ft

The parameters listed below were chosen to bound the current plant configuration:

h = 118.5 ft
F = 2135 gpm
V = 2.088E6 ft^3

The E/D term depends upon the particle size distribution and spray drop size. It is conservative to use 10 m^{-1} for E/D until the point is reached when the inventory in the atmosphere is reduced to 2 percent of its original (DF of 50). With the RG 1.183 (Reference 3) source term methodology, this is interpreted as being 2 percent of the total inventory particulate iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases.

These parameters and the appropriate conversion factors were used to calculate the particulate spray removal coefficients. A conservative value of 4.4 hr^{-1} was used in the analysis during the spray injection phase. The recirculation spray particulate removal rate used was 2.25 hr^{-1} corresponding with the reduction in the spray flow rate (2135 gpm injection reduced to 1080 gpm for recirculation). Recirculation sprays were terminated at 3.4 hours. The DF of 50 was not reached by 3.4 hours, so no further reduction in particulate removal by sprays was modeled.

Sedimentation Removal of Particulates

During spray operation, no credit was taken for sedimentation removal of particulates in the sprayed region, although it would take place. It was assumed that containment spray operation would be terminated at 3.4 hours. Credit was taken for sedimentation removal of particulates in the sprayed region after spray termination. Sedimentation was credited in the unsprayed region from the start of the event. As approved in Reference 4, the analysis credited a sedimentation coefficient of 0.1 hr^{-1}.

ECCS Leakage

When external ECCS recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment outside containment. The leakage goes into the Auxiliary Building and no filtration or holdup is credited for this release. Initially, the ECCS recirculation is internal to the containment and there is no potential for leakage outside containment. However, the switch to external recirculation occurs at 6.5 hours because of the need to switch from cold leg recirculation mode to hot leg recirculation mode. The ECCS leakage is modeled as 4.0 gallons/hr, which is doubled from the plant allowable leakage value of 2.0 gallons/hr, consistent with RG 1.183 (Reference 3). The leakage continues for the 30-day period following the accident considered in the analysis. The airborne fraction was modeled as time-dependent and the values calculated for the stretch power uprate and used in the analysis are listed in Table 6.11-21.

Control Room Isolation

In the event of a LBLOCA, the low-pressurizer pressure SI setpoint will be reached shortly after event initiation. The SI signal causes the control room HVAC to switch from the normal-

operation mode to the emergency mode of operation. It was conservatively assumed that the control room HVAC would not fully enter the emergency mode of operation until 1 minute after event initiation.

6.11.9.7.2 Acceptance Criteria

The offsite dose limit for a LOCA is 25-rem TEDE per RG 1.183 (Reference 3). This is the guideline value of 10CFR50.67 (Reference 1). The limit for the control room dose is 5-rem TEDE per 10CFR50.67.

6.11.9.7.3 Results and Conclusions

The calculated total offsite and control room doses due to the LBLOCA were reported as:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	17.8	25
LPZ	13.6	25
Control Room	4.9	5

The acceptance criteria were met.

The SB dose reported was for the worst 2-hour period, determined to be from 0.6 to 2.6 hours.

Subsection 6.11.4.6 of this report discusses the calculation of the direct and skyshine control room dose. The calculated dose for the 30-day duration considered in this analysis was 0.041 rem. This dose was included in the control room TEDE dose reported above.

6.11.9.8 GDT Rupture Radiological Consequences

For the GDT rupture analysis, there is assumed to be a failure that results in the release of the contents of one GDT.

6.11.9.8.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.11-22.

The GDT rupture analysis uses the analytical methods and assumptions outlined in RG 1.26 (Reference 29).

Consistent with the UFSAR analysis, the tank contents are assumed to be at the administratively controlled limit of 29,761 Curies of dose equivalent Xe-133. Dose equivalent Xe-133 is the amount of Xe-133 that results in the same gamma radiation dose as a given mixture of noble gases. A failure in the gaseous waste processing system is assumed to result in release of a single tank inventory with release duration of 5 minutes.

Control Room Isolation

It is conservatively assumed that the control room HVAC system remains in normal-operation mode throughout the event.

6.11.9.8.2 Acceptance Criteria

The offsite dose limit for a GDT rupture is 0.5-rem TEDE. This is consistent with the guidance of RG 1.26 (Reference 29), which specifies 0.5 rem whole body or equivalent to any part of the body, and of RG 1.183 (Reference 3), which specifies that doses will be determined as TEDE. The limit for the control room dose is 5-rem TEDE per 10CFR50.67 (Reference 1).

6.11.9.8.3 Results and Conclusions

The calculated doses due to the GDT rupture were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.14	0.5
LPZ	0.07	0.5
Control Room	0.05	5

The acceptance criteria were met.

The SB dose reported was for the worst 2-hour period, determined to be from 0 to 2 hours.

6.11.9.9 VCT Rupture

For the VCT rupture, a failure was assumed that results in the release of the tank contents, plus the noble gases and a fraction of the iodines from the letdown flow until the letdown path is isolated.

6.11.9.9.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.11-23.

The VCT rupture analysis uses the analytical methods and assumptions outlined in RG 1.26 (Reference 29). Changes from the UFSAR analysis include increased letdown flow, lengthened duration of letdown flow and inclusion of iodines.

The inventory of gases in the tank is based on continuous operation with 1.0 percent fuel defects and without any purge of the gas space. The inventory of iodine in the tank is based on operation of the plant with 1.0 percent fuel defects and with 90 percent of the iodine removed by the letdown demineralizer.

As a result of the accident, all of the noble gas in the tank and 1.0 percent of the iodine in the tank liquid are assumed to be released to the atmosphere over a period of 5 minutes.

After event initiation, letdown flow to the VCT continues at the maximum flow rate of 132 gpm (maximum letdown flow plus 10-percent uncertainty) for 30 minutes when the letdown line is assumed to be isolated. The primary coolant noble gas activities used in the VCT rupture dose calculations are based on operation with 1 percent fuel defects. The primary coolant iodine activity is conservatively assumed to be at the pre-existing iodine spike level of 60 μCi/gram dose equivalent (DE) I-131, which is reduced by 90 percent by the letdown demineralizer. All of the noble gas and 1.0 percent of the iodine in the letdown flow are assumed to be released to the environment.

Control Room Isolation

It is conservatively assumed that the control room HVAC system remains in normal-operation mode throughout the event.

6.11.9.9.2 Acceptance Criteria

The offsite dose limit for a VCT rupture is 0.5-rem TEDE. This is consistent with the guidance of RG 1.26 (Reference 29), which specifies 0.5-rem whole body or equivalent to any part of the body and of RG 1.183 (Reference 3), which specifies that doses will be determined as TEDE. The limit for the Control Room dose is 5-rem TEDE per 10CFR50.67 (Reference 1).

6.11.9.9.3 Results and Conclusions

The calculated doses due to the VCT rupture were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.39	0.5
LPZ	0.18	0.5
Control Room	0.12	5

The acceptance criteria were met.

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

6.11.9.10 HT Failure

During normal plant operation water is added to the HTs periodically as the primary coolant is diluted during the fuel cycle to provide reduction in the primary coolant boron concentration. As water enters the HT, gases (the nitrogen cover gas and the noble gas and hydrogen that evolve out of solution from the water entering the tank) are displaced to the gaseous waste system. For the HT failure, a failure is assumed that results in the release of the contents of the tank.

6.11.9.10.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.11-24.

The HT rupture analysis uses the analytical methods and assumptions outlined in RG 1.26 (Reference 29). Other changes from the UFSAR analysis include the inventory based on continued power operation instead of the shutdown condition and the inclusion of iodines

The inventory of gases in the tank is based on continuous operation with 1.0 percent fuel defects and without any purge of the gas space. The inventory of iodine in the tank is based on operation of the plant with 1.0 percent fuel defects and with 90 percent of the iodine removed by the letdown demineralizer.

As a result of the HT failure, all of the noble gas in the tank and 1.0 percent of the iodine in the tank liquid are assumed to be released to the atmosphere over a period of 5 minutes.

Control Room Isolation

It is conservatively assumed that the control room HVAC System remains in normal-operation mode throughout the event.

6.11.9.10.2 Acceptance Criteria

The offsite dose limit for an HT failure is 0.5-rem TEDE. This is consistent with the guidance of RG 1.26 (Reference 29), which specifies 0.5 rem whole body or equivalent to any part of the body and of RG 1.183 (Reference 3), which specifies that doses will be determined as TEDE. The limit for the Control Room dose is 5-rem TEDE per 10CFR50.67 (Reference 1).

6.11.9.10.3 Results and Conclusions

The calculated doses due to the HT failure were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.40	0.5
LPZ	0.19	0.5
Control Room	0.06	5

The acceptance criteria were met.

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

6.11.9.11 FHA

This accident assumes that a fuel assembly is dropped and damaged during refueling. Analysis of the accident was performed with assumptions selected so that the results would be bounding for the accident occurring either inside containment or in the Fuel-Handling Building. Activity released from the damaged assembly was assumed to be released to the outside atmosphere through either the Containment Purge System or the Fuel Pit Ventilation System.

6.11.9.11.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.11-25.

The analysis of the fuel-handling accident (FHA) radiological consequences was performed using the analytical methods and assumptions outlined in RG 1.183 (Reference 3). The analysis was consistent with that approved in Reference 4, with changes made to reflect the increased power. This analysis allowed fuel movement 84 hours after shutdown.

All activity released from the fuel pit was assumed to be released to the atmosphere in 2 hours, using a linear release model. No credit was taken for operating the Spent Fuel Pit Ventilation System in the fuel handling building. No credit was taken for isolating containment for the FHA in containment. Since the assumptions and parameters for an FHA inside containment are identical to those for a FHA in the Fuel-Handling Building, the radiological consequences were the same regardless of the location of the accident.

Source Term

The calculation of the radiological consequences following an FHA used gap fractions of 12 percent for I-131, 30 percent for Kr-85, and 10 percent for all other nuclides. The value for I-131 was taken from NUREG/CR-5009 (Reference 30). The values for Kr-85 and the other iodines and noble gases were taken from RG 1.25 (Reference 31). There are lower values identified in Table 3 of RG 1.183 (Reference 3), but these were not used because the conditions for their use (specified in footnote 11 in RG 1.183) have not been ensured.

As in the existing licensing basis, it was assumed that all of the fuel rods in the equivalent of one fuel assembly would be damaged to the extent that all of their gap activity would be released. The assembly inventory was based on the assumption that the subject fuel assembly had been operated at 1.7 times the core average power. The activity calculated for the third transition cycle was conservatively increased by 4 percent to bound variations in core average enrichment, core mass, and cycle length (Table 6.11-26).

The decay time used in the analysis was 84 hours.

Iodine Chemical Form

Iodine species in the pit were assumed to be 99.85 percent elemental and 0.15 percent organic. This was based on the split leaving the fuel of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine and 0.15 percent organic iodine. It assumed that all CsI was dissociated in the

water and the iodine re-evolved as elemental iodine. This was assumed to occur instantaneously. Thus, 99.85 percent of the iodine released was elemental.

Pit Scrubbing Removal of Activity

The activity released from the damaged fuel rods was assumed to be contained within gas bubbles that rise up through the water pit and are released into the atmosphere above the pit. As the bubbles pass through the water column, there is a significant removal of activity. RG 1.183 (Reference 3) identifies a pit DF of 500 for elemental iodine and no removal for organic iodine and noble gases. The pit DF of 500 for elemental iodine is based on having a water height of 23 feet or more. (Per the Technical Specifications, there are requirements for ≥23 feet of water above the stored spent fuel and above the reactor vessel flange during fuel-handling operations.)

The DF of 500 for elemental iodine is also based on fuel rod pressure of ≤1200 psig. There is the potential for fuel rod pressures to exceed 1200 psig (but remain less than 1500 psig). With this increase in fuel rod pressure, the pit DF is determined to remain above 400. Using a pit DF of 400 for elemental iodine and the defined iodine species split of 99.85 percent elemental and 0.15 percent organic, the overall pit DF would be 250. However, RG 1.183 (Reference 3) also specifies the overall DF for iodine to be 200. The overall DF of 200 has an associated elemental iodine DF of 285, and this value was used in the analysis together with a DF of 1.0 for organic iodine and noble gases.

The cesium released from the damaged fuel rods was assumed to remain in a nonvolatile form and not be released from the pit.

The split between elemental and organic iodine leaving the pit had no effect on the analysis since no filtration was credited.

Filtration of Release Paths

No credit was taken for removing iodine by filters, nor was credit taken for isolating release paths.

Although the containment purge will be automatically isolated on a purge-line high-radiation alarm, isolation was not modeled in the analysis. The activity released from the damaged assembly was assumed to be released to the outside atmosphere over a 2-hour period. Since no filters or containment isolation was modeled, this analysis supported refueling operation with the equipment hatch or personnel air lock remaining open.

Control Room Isolation

It was conservatively assumed that the control room HVAC system would remain in normal operation mode throughout the event.

6.11.9.11.2 Acceptance Criteria

The offsite-dose limit for an FHA is 6.3-rem TEDE per RG 1.183 (Reference 3). This is ~25 percent of the guideline value of 10CFR50.67 (Reference 1). The limit for the control room dose is 5-rem TEDE per 10CFR50.67.

6.11.9.11.3 Results and Conclusions

The calculated doses due to the FHA were:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	4.2	6.3
LPZ	2.0	6.3
Control Room	3.0	5

The acceptance criteria were met.

The SB dose reported was for the worst 2-hour period, determined to be from 0 to 2 hours.

6.11.10 References

1. 10CFR50.67, *Accident Source Term*, 64 FR 72001, December 23, 1999.

2. NUREG-0800, *Standard Review Plan* 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Rev. 0.

3. NRC Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, Rev. 0, July 2000.

4. Letter from P. Milano (NRC) to A. Blind (Consolidated Edison Company of New York), *Indian Point Nuclear Generating Unit No. 2 – RE: Issuance of Amendment Affecting Containment Air Filtration, Control Room Air Filtration, and Containment Integrity During Fuel Handling Operations (TAC NO. MA6955)*, Docket No. 50-247, July 27, 2000.

5. Regulatory Guide 1.49, *Power Levels of Nuclear Power Plants*, Rev. 1.

6. Letter from P. Milano (NRC) to M. Kansler (Entergy Nuclear Operations, Inc.), *Indian Point Nuclear Generating Unit No. 2 – RE: Issuance of Amendment RE: 1.4-Percent Power Uprate (TAC NO. MB6950)*, Docket No. 50-247, May 22, 2003.

7. NUREG-1465, *Accident Source Terms for Light-Water Power Plants*, February 1995.

8. 10CFR20, *Standards for Protection Against Radiation*, May 21, 1991.

9. 10CFR50, Appendix I, *Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion As Low As Reasonably Achievable for Radioactive Material in Light Water Cooled Nuclear Power Reactor Effluents.*

10. Appendix A to Facility Operating License DPR-26 for Entergy Nuclear Indian Point 2, LLC and Entergy Nuclear Operations, Inc., *Indian Point Nuclear Generating Plant Unit No. 2 Docket No. 50-247 Technical Specifications and Bases.*

11. Regulatory Guide 1.26, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*, Rev. 2, June 1975.

12. TID-14844, *Calculation of Distance Factors for Power and Test Reactor Sites*, 1962.

13. SECY-98-154, *Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors*, June 30, 1998.

14. ANSI/ANS-18.1-1999, *Radioactive Source Term for Normal Operation of Light Water Reactors*, The American Nuclear Society Standards Institute, Inc., LaGrange Park, Illinois, September 21, 1999.

15. *Indian Point Unit 2 Environmental Qualification Manual*, Rev. 13, April 1999.

16. *Offsite Dose Calculation Manual.*

17. NUREG-0017, *Calculation of Releases of Radio Active Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors*, Rev. 1, April 1985.

18. *Indian Point 2 Annual Radioactive Effluent Release Reports*, 1997-2001.

19. NUREG-0578,*TMI Lessons Learned Task Force Status Report and Short Term Recommendations*, July 1979.

20. NUREG-0737, *Clarification of TMI Action Plan Requirements*, November 1980.

21. EDS Nuclear Report No. 02-0180-1026, *Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which May be Used in Post Accident Operations*, December 1979.

22. NRC Report 50-247/83-14, *Shielding Design Inspection Report.*

23. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report*, Docket No. 50-247.

24. 10CFR50.49, *Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants*, 66FR64738, December 14, 2001.

25. NRC Regulatory Guide 1.89, *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants*, Rev. 1, 1984.

26. EPA Federal Guidance Report No. 11, EPA-520/1-88-020, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, September 1988.

27. EPA Federal Guidance Report No. 12, EPA 402-R-93-081, *External Exposure to Radionuclides in Air, Water and Soil*, September 1993.

28. NUREG-0800, *Standard Review Plan,* 6.5.2, "Containment Spray as a Fission Product Cleanup System," Rev. 2, December 1988.

29. NRC Regulatory Guide 1.26, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*, Rev. 2 June 1975.

30. NUREG/CR-5009, *Assessment of the Use of Extended Burnup Fuel in Light Water Reactors*, February 1988.

31. NRC Regulatory Guide 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, March 1972.

Table 6.11-1 Input Parameters for Core Inventory Calculations - Cycle 19	
Parameter	**Value**
Core Thermal Power (MWt)	3280.3 (3216*1.02)
Fuel Assembly Type	15 x 15
Uranium Mass (MTU)	86.6
Cycle Length (MWD/MTU)	25430
Loading Pattern	See Table 6.11-2
Uranium Enrichments (wt % U-235)	Region 21B 4.80 Region 21A 4.57 Region 20B 4.79 Region 20A 4.57 Region 19A 4.11

Table 6.11-2 Input Parameters for Loading Pattern - Cycle 19			
Region	**No. of Assemblies**	**EOC Burnup (MWD/MTU)**	**Average Relative Power**
Feed Region 21B	76	30310	1.19
Feed Region 21A	17	32070	1.26
1 x Burned Region 20B	88	52320	0.87
1 x Burned Region 20B	4	44040	0.51
2 x Burned Region 19A	8	45580	0.30

Table 6.11-3

Core Inventory with 1.04 Fuel Management Variation Multiplier (core power = 3280.3 MWt)

Nuclide	Inventory at Shutdown (Ci)	Inventory at 84 Hours after Shutdown (Ci)	Nuclide	Inventory at Shutdown (Ci)	Inventory at 84 Hours after Shutdown (Ci)
I-130	3.80E+06	3.44E+04	Ce-141	1.52E+08	1.42E+08
I-131	9.16E+07	6.94E+07	Ce-143	1.42E+08	2.45E+07
I-132	1.33E+08	6.39E+07	Ce-144	1.20E+08	1.19E+08
I-133	1.88E+08	1.17E+07	Pu-238	4.13E+05	4.15E+05
I-134	2.06E+08	1.16E-20	Pu-239	3.50E+04	3.53E+04
I-135	1.75E+08	2.62E+04	Pu-240	5.23E+04	5.23E+04
			Pu-241	1.18E+07	1.17E+07
Kr-85m	2.43E+07	5.59E+01	Np-239	1.88E+09	6.75E+08
Kr-85	1.10E+06	1.10E+06			
Kr-87	4.66E+07	6.16E-13	Y-90	9.11E+06	8.90E+06
Kr-88	6.56E+07	8.13E-02	Y-91	1.14E+08	1.10E+08
Xe-131m	1.01E+06	9.85E+05	Y-92	1.20E+08	3.64E+01
Xe-133m	5.87E+06	2.91E+06	Y-93	1.39E+08	4.41E+05
Xe-133	1.80E+08	1.36E+08	Nb-95	1.56E+08	1.55E+08
Xe-135m	3.68E+07	4.20E+03	Zr-95	1.54E+08	1.49E+08
Xe-135	4.77E+07	7.83E+05	Zr-97	1.55E+08	4.93E+06
Xe-138	1.55E+08	0.00E+00	La-140	1.73E+08	1.50E+08
			La-141	1.53E+08	6.11E+01
Cs-134	2.06E+07	2.06E+07	La-142	1.48E+08	7.16E-09
Cs-136	6.01E+06	4.99E+06	Nd-147	6.11E+07	4.91E+07
Cs-137	1.19E+07	1.19E+07	Pr-143	1.37E+08	1.25E+08
Cs-138	1.71E+08	0.00E+00	Am-241	1.41E+04	1.43E+04
Rb-86	2.38E+05	2.09E+05	Cm-242	3.52E+06	3.49E+06
			Cm-244	3.82E+05	3.82E+05
Te-127	9.84E+06	6.40E+06			
Te-127m	1.29E+06	1.28E+06			
Te-129	2.92E+07	2.62E+06			
Te-129m	4.30E+06	4.02E+06			
Te-131m	1.33E+07	1.92E+06			
Te-132	1.31E+08	6.20E+07			
Sb-127	9.95E+06	5.38E+06			
Sb-129	2.97E+07	4.21E+01			
Sr-89	8.83E+07	8.41E+07			
Sr-90	8.75E+06	8.75E+06			
Sr-91	1.11E+08	2.41E+05			
Sr-92	1.20E+08	5.59E-02			
Ba-139	1.67E+08	8.54E-11			
Ba-140	1.61E+08	1.33E+08			
Ru-103	1.40E+08	1.31E+08			
Ru-105	9.62E+07	2.00E+02			
Ru-106	4.89E+07	4.86E+07			
Rh-105	8.86E+07	1.98E+07			
Mo-99	1.75E+08	7.23E+07			
Tc-99m	1.53E+08	6.97E+07			

Table 6.11-4 Input Parameters for RCS Activity and Inventory Calculations	
Parameter	**Value**
Core Thermal Power (MWt)	3280.3 (3216*1.02)
Cycle Length (full-power days)	685
Initial Boron Concentration (ppm)	1030
Mixed-Bed Demineralizer Resin Volume (ft^3)	30
Failed Fuel Fraction (%)	1.0
Reactor Coolant Mass (lbm)	4.85×10^5
Purification System Flow Rate, Normal (gpm)	75
Purification System Flow Rate, Maximum (gpm)	120
VCT Liquid Volume (ft^3)	130
VCT Vapor Volume (ft^3)	270
VCT Temperature (°F)	127

Table 6.11-5

Reactor-Coolant-Fission- and Corrosion-Product-Specific Activities (core power = 3280.3 MWt)

Nuclide	Activity µCi/g	Nuclide	Activity µCi/g	Nuclide	Activity µCi/g
Kr-83m	4.67E-01	Mn-54	1.60E-03	Ag-110m	4.89E-03
Kr-85m	1.85E+00	H-3	3.5 (max)	Te-125m	1.15E-03
Kr-85	1.36E+01	Cr-51	5.50E-03	Te-127m	3.83E-03
Kr-87	1.22E+00	Mn-56	2.00E-02	Te-127	1.57E-02
Kr-88	3.49E+00	Fe-55	2.00E-03	Te-129m	1.16E-02
Kr-89	9.90E-02	Fe-59	5.20E-04	Te-129	1.50E-02
Xe-131m	3.18E+00	Co-58	1.56E-02	Te-131m	2.63E-02
Xe-133m	3.61E+00	Co-60	1.98E-03	Te-131	1.42E-02
Xe-133	2.57E+02	Rb-86	4.55E-02	Te-132	3.14E-01
Xe-135m	5.55E-01	Rb-88	4.36E+00	Te-134	3.13E-02
Xe-135	8.94E+00	Rb-89	2.00E-01	Ba-137m	2.48E+00
Xe-137	1.93E-01	Sr-89	4.37E-03	Ba-140	4.36E-03
Xe-138	6.94E-01	Sr-90	2.85E-04	La-140	1.46E-03
Br-83	9.90E-02	Sr-91	5.78E-03	Ce-141	6.56E-04
Br-84	4.86E-02	Sr-92	1.28E-03	Ce-143	5.24E-04
Br-85	5.67E-03	Y-90	8.09E-05	Pr-143	6.37E-04
I-127 (a)	1.53E-10	Y-91m	3.12E-03	Ce-144	4.92E-04
I-129	8.48E-08	Y-91	5.77E-04	Pr-144	4.92E-04
I-130	7.08E-02	Y-92	1.12E-03		
I-131	2.90E+00	Y-93	3.86E-04		
I-132	3.02E+00	Zr-95	6.55E-04		
I-133	4.65E+00	Nb-95	6.56E-04		
I-134	6.52E-01	Mo-99	8.22E-01		
I-135	2.57E+00	Tc-99m	7.62E-01		
Cs-134	5.14E+00	Ru-103	6.42E-04		
Cs-136	5.35E+00	Rh-103m	6.38E-04		
Cs-137	2.62E+00	Ru-106	3.30E-04		
Cs-138	1.06E+00	Rh-106	3.30E-04		

Notes:

1. (a) Grams of I-127 per gram of coolant.
2. Mn-54 is from the ANSI/ANS-18.1-1999. Others are standard Westinghouse values.
3. Calculated specific activities have been multiplied by 1.04.
4. Operation with fuel with defects generating 1% of core power.
5. RCS purification at 75 gpm.
6. No volume control tank purging.
7. RCS mass - 2.20E+08 g.

Table 6.11-6 Nuclide Inventories for Noble Gases and Iodine in the VCT (total of gas and liquid phases)	
VCT Isotope	**Inventory (curies)**
Kr-85m	1.52E+02
Kr-85	2.36E+03
Kr-87	4.64E+01
Kr-88	2.25E+02
Xe-131m	3.77E+02
Xe-133m	4.10E+02
Xe-133	2.99E+04
Xe-135m	7.29E+01
Xe-135	9.08E+02
Xe-138	6.30E+00
I-130	1.74E-02
I-131	5.92E-01
I-132	9.22E-01
I-133	1.47E+00
I-134	2.14E-01
I-135	6.88E-01

Table 6.11-7 CVCS Holdup Tank Inventory	
Nuclide	**Inventory (Ci)**
Kr-85m	2.26E+02
Kr-85	2.65E+03
Kr-87	6.42E+01
Kr-88	3.39E+02
Xe-131m	6.15E+02
Xe-133m	6.74E+02
Xe-133	4.92E+04
Xe-135m	5.52E+00
Xe-135	1.37E+03
Xe-138	6.36E+00
I-130	2.99E-03
I-131	1.44E-01
I-132	6.62E-02
I-133	2.10E-01
I-134	6.18E-03
I-135	9.35E-02

Table 6.11-8 Estimated Effect of the Core Uprate on 10CFR50 Appendix I Doses				
Type of Dose	**Appendix I Design Objectives**	**5-Yr Annual Average Doses (Base Case)(1)**	**Scaled Doses (Uprate Case)(2)**	**Percentage of Appendix I Design Objectives for the Uprate Case**
Liquid Effluents				
Dose to Total Body from all Pathways	3 mrem/yr per unit	1.25E-01 mrem/yr	1.40E-01 mrem/yr	4.67%
Dose to any Organ from all Pathways	10 mrem/yr per unit	2.22E-01 mrem/yr	2.48E-01 mrem/yr	2.48%
Gaseous Effluents				
Gamma Dose in Air	10 mrad/yr per unit	Not reported	Based on the airborne doses below, expected to be well within Appendix I guidelines	N/A
Beta Dose in Air	20 mrad/yr per unit	Not reported	Based on the airborne doses below, expected to be well within Appendix I guidelines	N/A
Dose to Total Body of an Individual	5 mrem/yr per unit	9.19E-03 mrem/yr	1.10E-02 mrem/yr	0.23%
Dose to Skin of an Individual	15 mrem/yr per unit	2.57E-02 mrem/yr	3.09E-02 mrem/yr	0.21%
Radioiodines and Particulates Released to the Atmosphere				
Dose to any Organ from all Pathways	15 mrem/yr per unit	1.63 mrem/yr	1.82 mrem/yr	12.13%

Notes:

1. Average core power level for the base case was 2941.3 MWt.
2. Core power level assumed for uprate analysis is 3280 MWt (includes margin for power uncertainty).

Table 6.11-9 Nuclide Parameters			
Nuclide	Decay Constant (hr^{-1})	CEDE DCF (rem/Ci inhaled)	EDE DCF ($rem \cdot m^3/Ci \cdot sec$)
I-130	5.61E-02	2.642E3	3.848E-01
I-131	3.59E-03	3.289E4	6.734E-02
I-132	3.01E-01	3.811E2	4.144E-01
I-133	3.33E-02	5.846E3	1.088E-01
I-134	7.91E-01	1.314E2	4.810E-01
I-135	1.05E-01	1.228E3	2.953E-01
Kr-85m	1.55E-01	NA	2.768E-02
Kr-85	7.38E-06	NA	4.403E-04
Kr-87	5.45E-01	NA	1.524E-01
Kr-88	2.44E-01	NA	3.774E-01
Xe-131m	2.43E-03	NA	1.439E-03
Xe-133m	1.32E-02	NA	5.069E-03
Xe-133	5.51E-03	NA	5.772E-03
Xe-135m	2.72E+00	NA	7.548E-02
Xe-135	7.63E-02	NA	4.403E-02
Xe-138	2.93E+00	NA	2.135E-01
Cs-134	3.84E-05	4.625E4	2.801E-01
Cs-136	2.20E-03	7.326E3	3.922E-01
Cs-137	2.64E-06	3.193E4	1.066E-01*
Cs-138	1.29E+00	1.014E2	4.477E-01
Rb-86	1.55E-03	6.623E3	1.780E-02
Te-127m	2.65E-04	2.150E4	5.439E-04
Te-127	7.41E-02	3.182E2	8.954E-04
Te-129m	8.60E-04	2.394E4	5.735E-03
Te-129	5.98E-01	8.954E1	1.018E-02
Te-131m	2.31E-02	6.401E3	2.594E-01
Te-132	8.86E-03	9.435E3	3.811E-02
Sb-127	7.50E-03	6.031E3	1.232E-01
Sb-129	1.60E-01	6.438E2	2.642E-01
Sr-89	5.72E-04	4.144E4	2.860E-04
Sr-90	2.72E-06	1.299E6	2.786E-05
Sr-91	7.30E-02	1.661E3	1.277E-01
Sr-92	2.56E-01	8.066E2	2.512E-01
Ba-139	5.03E-01	1.717E2	8.029E-03
Ba-140	2.27E-03	3.737E3	3.175E-02

Table 6.11-9 (Cont.) Nuclide Parameters			
Nuclide	Decay Constant (hr^{-1})	CEDE DCF (rem/Ci inhaled)	EDE DCF (rem· m^3 /Ci · sec)
Ru-103	7.35E-04	8.954E3	8.325E-02
Ru-105	1.56E-01	4.551E2	1.410E-01
Ru-106	7.84E-05	4.773E5	0.00E+00
Rh-105	1.96E-02	9.546E2	1.376E-02
Mo-99	1.05E-02	3.959E3	2.694E-02
Tc-99m	1.15E-01	3.256E1	2.179E-02
Ce-141	8.89E-04	8.954E3	1.269E-02
Ce-143	2.10E-02	3.389E3	4.773E-02
Ce-144	1.02E-04	3.737E5	3.156E-03
Pu-238	9.02E-07	3.922E8	1.806E-05
Pu-239	3.29E-09	4.292E8	1.569E-05
Pu-240	1.21E-08	4.292E8	1.758E-05
Pu-241	5.50E-06	8.251E6	2.683E-07
Np-239	1.23E-02	2.509E3	2.845E-02
Y-90	1.08E-02	8.436E3	7.030E-04
Y-91	4.94E-04	4.884E4	9.620E-04
Y-92	1.96E-01	7.807E2	4.810E-02
Y-93	6.86E-02	2.153E3	1.776E-02
Nb-95	8.22E-04	5.809E3	1.384E-01
Zr-95	4.51E-04	2.364E4	1.332E-01
Zr-97	4.10E-02	4.329E3	3.337E-02
La-140	1.72E-02	4.847E3	4.329E-01
La-141	1.76E-01	5.809E2	8.843E-03
La-142	4.50E-01	2.531E2	5.328E-01
Nd-147	2.63E-03	6.845E3	2.290E-02
Pr-143	2.13E-03	8.103E3	7.770E-05
Am-241	1.83E-07	4.440E8	3.027E-03
Cm-242	1.77E-04	1.728E7	2.105E-05
Cm-244	4.37E-06	2.479E8	1.817E-05

Notes:

CEDE = Committed effective dose equivalent

EDE = Effective dose equivalent

DCF = Dose conversion factor

* This is the DCF for Ba-137m. The DCF for Cs-137 is low; however a significant amount of Ba-137m is produced through decay. This is conservatively addressed by applying the DCF from Ba-137m to Cs-137.

Table 6.11-10 Offsite Breathing Rates and Atmospheric Dispersion Factors	
Time	**Offsite Breathing Rates (m^3/sec)**
0 - 8 hours	3.5E-4
8 - 24 hours	1.8E-4
>24 hours	2.3E-4
	Offsite Atmospheric Dispersion Factors (sec/m^3)
SB[1]	7.5E-4
LPZ	
0 - 8 hours	3.5E-4
8 - 24 hours	1.2E-4
1 - 4 days	4.2E-5
> 4 days	9.3E-6

Note:

1. This SB atmospheric dispersion factor is conservatively applied during all time intervals in the determination of the limiting 2-hour period

Table 6.11-11 Control Room Parameters	
Breathing Rate - Duration of the Event	3.5E-4 m^3/sec
Control Room Volume	102,400 ft^3
Occupancy Factors	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4
Normal Ventilation Flow Rates	
Filtered Makeup Flow Rate	0.0 scfm
Filtered Recirculation Flow Rate	0.0 scfm
Unfiltered Makeup Flow Rate	920 scfm
Unfiltered Recirculation Flow Rate	(not modeled - no effect on analyses)
Unfiltered In-leakage	700 scfm
Emergency Ventilation System Flow Rates	
Filtered Makeup Air Flow Rate	1800.0 scfm
Filtered Recirculation Flow Rate	0 scfm
Unfiltered Makeup Flow Rate	0 scfm
Unfiltered In-leakage	700 scfm
Filter Efficiencies	
Elemental Iodine	95%
Organic Iodine	90%
Particulates	99%
Radiation Monitor Setpoint	0.6 mrem/hr
Delay to Initiate Switchover of HVAC from Normal Operation to Emergency Operation after SI Signal	60 seconds
Delay for Switchover of HVAC from Normal Operation to Emergency Operation after receiving a High Alarm Signal (radiation monitor) Based on Manual Action	10 minutes
Control Room Shielding	0.5 inch steel

Table 6.11-12 Core Total Fission Product Activities Based on 3280.3 MWt (102% of 3216 MWt)	
Isotope	**Activity (Ci)**
I-130	3.80E+06
I-131	9.16E+07
I-132	1.33E+08
I-133	1.88E+08
I-134	2.06E+08
I-135	1.75E+08
Kr-85m	2.43E+07
Kr-85	1.10E+06
Kr-87	4.66E+07
Kr-88	6.56E+07
Xe-131m	1.01E+06
Xe-133m	5.87E+06
Xe-133	1.80E+08
Xe-135m	3.68E+07
Xe-135	4.77E+07
Xe-138	1.55E+08
Cs-134	2.06E+07
Cs-136	6.01E+06
Cs-137	1.19E+07
Cs-138	1.71E+08
Rb-86	2.38E+05
Te-127m	1.29E+06
Te-127	9.84E+06
Te-129m	4.30E+06
Te-129	2.92E+07
Te-131m	1.33E+07
Te-132	1.31E+08
Sb-127	9.95E+06
Sb-129	2.97E+07

Table 6.11-12 (Cont.) Core Total Fission Product Activities Based on 3280.3 MWt (102% of 3216 MWt)	
Isotope	**Activity (Ci)**
Sr-89	8.83E+07
Sr-90	8.75E+06
Sr-91	1.11E+08
Sr-92	1.20E+08
Ba-139	1.67E+08
Ba-140	1.61E+08
Ru-103	1.40E+08
Ru-105	9.62E+07
Ru-106	4.89E+07
Rh-105	8.86E+07
Mo-99	1.75E+08
Tc-99m	1.53E+08
Ce-141	1.52E+08
Ce-143	1.42E+08
Ce-144	1.20E+08
Pu-238	4.13E+05
Pu-239	3.50E+04
Pu-240	5.23E+04
Pu-241	1.18E+07
Np-239	1.88E+09
Y-90	9.11E+06
Y-91	1.14E+08
Y-92	1.20E+08
Y-93	1.39E+08
Nb-95	1.56E+08
Zr-95	1.54E+08
Zr-97	1.55E+08
La-140	1.73E+08
La-141	1.53E+08
La-142	1.48E+08
Nd-147	6.11E+07
Pr-143	1.37E+08
Am-241	1.41E+04
Cm-242	3.52E+06
Cm-244	3.82E+05

Table 6.11-13 RCS Coolant Concentrations Based on 1-% Fuel Defects[1]	
Nuclide	**Activity (μCi/gm)**
I-130	0.0708
I-131	2.90
I-132	3.02
I-133	4.65
I-134	0.652
I-135	2.57
Kr-85m	1.85
Kr-85	13.6
Kr-87	1.22
Kr-88	3.49
Xe-131m	3.18
Xe-133m	3.61
Xe-133	257.0
Xe-135m	0.555
Xe-135	8.94
Xe-138	0.694
Cs-134	5.14
Cs-136	5.35
Cs-137	2.62
Cs-138	1.06
Rb-86	0.0455

Note:

1. Plant Technical Specification limits primary coolant iodine coolant concentration to 1.0 μCi/gm dose equivalent I-131. These coolant concentrations are provided in Table 6.11-14.

Table 6.11-14 Iodine Specific Activities (µCi/gm)			
	Primary Coolant		Secondary Coolant
Nuclide	1 µCi/gm[1]	60 µCi/gm	0.15 µCi/gm
I-130	0.0183	1.10	0.0027
I-131	0.7482	44.89	0.1122
I-132	0.7792	46.75	0.1169
I-133	1.1997	71.98	0.1800
I-134	0.1682	10.09	0.0252
I-135	0.6631	39.79	0.0995

Note:

1. Iodine concentrations are converted to DE I-131 using the CEDE DCFs in Table 6.11-9.

Table 6.11-15 Iodine Spike Appearance Rates (Curies/Minute)[1]						
	I-130	I-131	I-132	I-133	I-134	I-135
335 Times the Equilibrium Rate (SGTR)	4.7	139.3	458.0	271.4	210.2	214.2
500 Times the Equilibrium Rate (MSLB)	7.0	207.9	683.7	405.1	313.8	319.8

Note:

1. Calculated based on the RCS concentration of 1.0 µCi/gm DE I-131, letdown flow of 120 gpm + 10% with perfect cleanup and RCS leakage of 11 gpm.

Table 6.11-16 Assumptions Used for Steamline Break Dose Analysis	
Source Term	
Nuclide Parameters	See Table 6.11-9
Primary Coolant Noble Gas Activity prior to Accident	Based on operation with 1.0% Fuel Defects (See Table 6.11-13)
Primary Coolant Iodine Activity prior to Accident	
Pre-Existing Spike	60 μCi/gm of DE I-131 (See Table 6.11-14)
Accident-Initiated Spike	1.0 μCi/gm of DE I-131 (See Table 6.11-14)
Primary Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate (See Table 6.11-15)
Duration of Accident-Initiated Spike	3.0 hours
Secondary Coolant Iodine Activity prior to Accident	0.15 μCi/gm of DE I-131 (See Table 6.11-14)
Iodine Chemical Form After Release To Atmosphere	
Elemental	97%
Organic	3%
Particulate (cesium iodide)	0%
Release Modeling	
Faulted SG Tube Leak Rate during Accident	150 gpd
Intact SG Tube Leak Rate during Accident	450 gpd
SG Iodine Steam/Water Partition Coefficient	
Intact SG	0.01
Faulted SG	1.0
Time for RHR to take over cooling	30 hours
Time to Cool RCS Below 212°F and Stop Releases from Faulted SG	65 hours
Steam Release from Intact SGs to Environment	
0-2 hours	381,000 lbm
2-8 hours	830,000 lbm
8-30 hours	1,488,000 lbm
Steam Release from Faulted SG to Environment (during first 5 minutes)	134,500 lbm
Primary Coolant Mass	486,000 lbm

Table 6.11-16 (Cont.)	
Assumptions Used for Steamline Break Dose Analysis	
Intact Steam Generator Secondary Mass	83,075 lbm/SG
Faulted Steam Generator Secondary Mass	134,500 lbm
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Time to Start Crediting Emergency Control Room HVAC	1 minute
Control Room Atmospheric Dispersion (χ/Q) Factors	
Intact and Faulted SG Releases:	
0 - 2 hours	1.09E-3 sec/m^3
2 - 8 hours	1.02E-3 sec/m^3
8 - 24 hours	4.99E-4 sec/m^3
24 - 96 hours	3.86E-4 sec/m^3
96 - 720 hours	2.99E-4 sec/m^3

Table 6.11-17 Assumptions Used for Locked Rotor Dose Analysis	
Source Term	
Nuclide Parameters	See Table 6.11-9
Core Activity	See Table 6.11-12
Fraction of Fuel Rods in Core Failing	5% of core
Fission Product Gap Fractions	
I-131	8% of core activity
Kr-85	10% of core activity
Other Iodines and Noble Gases	5% of core activity
Alkali Metals	12% of core activity
Radial peaking factor	1.7
RCS Iodines	60 µCi/gm DE I-131 (See Table 6.11-14)
RCS Noble Gases and Alkali Metals	Based on operation with 1.0% fuel defects (See Table 6.11-13)
Secondary Coolant Iodine Activity at Beginning of Event	0.15 µCi/gm DE I-131 (See Table 6.11-14 values)
Secondary Alkali Metal Activity at Beginning of Event	15% of Table 6.11-13 values
Iodine Chemical Form after Release to Atmosphere	
Elemental	97%
Organic	3%
Particulate (cesium iodide)	0%
Release Modeling	
Primary Coolant Mass	486,000 lbm
Secondary Coolant Mass	315,164 lbm (total)
Primary-to-Secondary Leak Rate	150 gal/day/SG
Steam Released from the Secondary Side	
0 - 2 hr	384,000 lbm
2 - 8 hr	860,000 lbm
8 - 30 hr	1,488,000 lbm
SG Iodine Steam/Water Partition Coefficient	0.01
SG Alkali Metal Steam/Water Partition Coefficient	0.0025
Termination of Releases	30 hours

Table 6.11-17 (Cont.)	
Assumptions Used for Locked Rotor Dose Analysis	
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Time to Start Crediting Emergency Control Room HVAC	21 minutes
Control Room Atmospheric Dispersion (χ/Q) Factors	
Secondary releases:	
0 - 2 hours	9.49E-4 sec/m^3
2 - 8 hours	8.65E-4 sec/m^3
8 - 24 hours	4.17E-4 sec/m^3
24 - 96 hours	3.30E-4 sec/m^3
96 - 720 hours	2.54E-4 sec/m^3

Table 6.11-18 Assumptions Used for Rod Ejection Accident	
Source Term	
Nuclide Parameters	See Table 6.11-9
Core Activity	See Table 6.11-12
Fraction of Fuel Rods in Core that Fail	10 (% of core)
Radial Peaking Factor	1.7
Fission Product Gap Fractions	
Iodines and Noble Gases	10% of core activity
Alkali Metals	12% of core activity
Fraction of Fuel Melting	0.25% of core
Fraction of Activity Released from Failed Fuel (gap activity)	100% for both containment leakage and steam generator steaming release paths
Fraction of Activity Released from Melted Fuel	
Noble Gases and Alkali Metals	100%
Iodines	25% for containment leakage release path 50% for steam generator steaming release path
RCS Iodines	1.0 µCi/gm DE I-131 (See Table 6.11-14)
RCS Noble Gases and Alkali Metals	Based on operation with 1% fuel defects (See Table 6.11-13)
Secondary Coolant Iodine Activity	0.15 µCi/gm DE I-131 (See Table 6.11-14)
Secondary Alkali Metal Activity	15% of Table 6.11-13 values
Containment Leakage Release Path	
Containment Net Free Volume	2.61E6 ft^3
Containment Leak Rates	
0 - 24 hours	0.1% weight %/day
> 24 hours	0.05 weight %/day
Iodine Chemical Form	4.85% elemental, 0.15% organic, and 95% particulate
Spray Removal in Containment	Not credited
Sedimentation Removal in Containment	
Particulate Iodine	0.1 hr^{-1}
Alkali Metals	0.1 hr^{-1}

Table 6.11-18 (Cont.)

Assumptions Used for Rod Ejection Accident

Steam Generator Steaming Release Path	
Primary Coolant Mass	486,000 lbm
Secondary Coolant Mass	315,164 lbm (total)
Primary-to-Secondary Leak Rate	150 gal/day/SG
Duration of Primary-to-Secondary Leakage	1 hr
Steam Released from the Secondary Side	
0 - 2 hours	400,000 lbm
> 2 hours	0 lbm
Iodine Chemical Form after Release to Atmosphere	97% elemental, 3% organic
SG Iodine Steam/Water Partition Coefficient	0.01
SG Alkali Metal Steam/Water Partition Coefficient	0.0025
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Time to Start Crediting Emergency Control Room HVAC	3 minutes
Control Room Atmospheric Dispersion (χ/Q) Factors	
Secondary Releases:	
0 - 2 hours	9.49E-4 sec/m^3
2 - 8 hours	8.65E-4 sec/m^3
8 - 24 hours	4.17E-4 sec/m^3
24 - 96 hours	3.30E-4 sec/m^3
96 - 720 hours	2.54E-4 sec/m^3
Containment Releases:	
0 - 2 hours	3.82E-4 sec/m^3
2 - 8 hours	2.81E-4 sec/m^3
8 - 24 hours	1.05E-4 sec/m^3
24 - 96 hours	8.31E-5 sec/m^3
96 - 720 hours	7.04E-5 sec/m^3

Table 6.11-19 Assumptions Used for SGTR Dose Analysis	
Source Term	
Nuclide Parameters	See Table 6.11-9
Primary Coolant Noble Gas Activity prior to Accident	Based on operation with 1.0% fuel defect (See Table 6.11-13)
Primary Coolant Iodine Activity Prior to Accident	
Pre-Existing Spike	60 µCi/gm of DE I-131 (See Table 6.11-14)
Accident-Initiated Spike	1.0 µCi/gm of DE I-131 (See Table 6.11-14)
Primary Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	335 times equilibrium rate (See Table 6.11-15)
Duration of Accident-Initiated Spike	4.0 hours
Secondary Coolant Iodine Activity Prior to Accident	0.15 µCi/gm of DE I-131 (See Table 6.11-14)
Release Modeling	
Ruptured SG Steam Releases	See Table 6.4-2
Ruptured SG Break Flow Rate	See Table 6.4-2
Break-Flow Flashing Fractions	See Table 6.4-2
Intact SG Tube Leak Rate during Accident	150 gpd
Steam Release from Intact SG to Environment	See Table 6.4-2
SG Iodine Steam/Water Partition Coefficient	
Ruptured and Intact SG Steam Release	0.01
Flashed Break Flow	1.0
Primary Coolant Mass	486,000 lbm
Steam Generator Secondary Mass	67,000 lbm/SG
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Time to Start Crediting Emergency Control Room HVAC	5.83 minutes

Table 6.11-19 (Cont.) Assumptions Used for SGTR Dose Analysis	
Control Room Atmospheric Dispersion (χ/Q) Factors	
Secondary releases:	
0 - 2 hours	9.49E-4 sec/m^3
2 - 8 hours	8.65E-4 sec/m^3
8 - 24 hours	4.17E-4 sec/m^3
24 - 96 hours	3.30E-4 sec/m^3
96 - 720 hours	2.54E-4 sec/m^3

Table 6.11-20 Assumptions Used for SBLOCA Analysis	
Source Term	
Nuclide Parameters	See Table 6.11-9
Core Activity	See Table 6.11-12
Fraction of Fuel Rods in Core that Fail	100% of core
Gap Fractions	
Iodine, Noble Gases and Alkali Metals	5% of core activity
Fraction of Fuel Melting	0% of core
Fraction of Activity Released from Failed Fuel (Gap Activity)	100%
RCS Noble Gas and Alkali Metal Activity Prior to Accident	Based on operation with 1.0% fuel defect (See Table 6.11-13)
RCS Iodine Activity Prior to Accident	1.0 μCi/gm of DE I-131 (See Table 6.11-14)
Secondary Coolant Iodine Activity Prior to Accident	0.15 μCi/gm of DE I-131 (See Table 6.11-14)
Secondary Coolant Alkali Metal Activity Prior to Accident	15% of Table 6.11-13 values
Containment Release Path	
Containment Net-Free Volume	2.61E6 (ft^3)
Containment Leak Rates	
0 - 24 hours	0.1 (weight %/day)
> 24 hours	0.05 (weight %/day)
Iodine Chemical Form	4.85% elemental, 0.15% organic and 95% particulate
Spray Removal in Containment	Not Credited
Sedimentation Removal in Containment	
Particulate Iodine	0.1 hr^{-1}
Alkali Metals	0.1 hr^{-1}
Deposition Removal in Containment	Not credited

Table 6.11-20 (Cont.) Assumptions Used for SBLOCA Analysis	
Steam Generator Steaming Release Path	
Primary Coolant Mass	486,000 lbm
Secondary Coolant Mass	315,164 lbm (total)
Primary-to-Secondary Leak Rate	150 gal/day/SG
Duration of Primary-to-Secondary Leakage	1 hr
Steam Released from the Secondary Side	
0 - 2 hours	400,000 lbm
> 2 hours	0 lbm
SG Iodine Steam/Water Partition Coefficient	0.01
SG Alkali Metal Steam/Water Partition Coefficient	0.01
Iodine Chemical Form after Release to Atmosphere	97% elemental, 3% organic
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Time to Start Crediting Emergency Control Room HVAC	3 minutes
Control Room Atmospheric Dispersion (χ/Q) Factors	
Secondary Releases:	
0 - 2 hours	9.49E-4 sec/m^3
2 - 8 hours	8.65E-4 sec/m^3
8 - 24 hours	4.17E-4 sec/m^3
24 - 96 hours	3.30E-4 sec/m^3
96 - 720 hours	2.54E-4 sec/m^3
Containment Releases:	
0 -2 hours	3.82E-4 sec/m^3
2 - 8 hours	2.81E-4 sec/m^3
8 - 24 hours	1.05E-4 sec/m^3
24 - 96 hours	8.31E-5 sec/m^3
96 - 720 hours	7.04E-5 sec/m^3

Table 6.11-21 Assumptions Used for LBLOCA Analysis	
Source Term	
Nuclide Parameters	See Table 6.11-9
Core Activity	See Table 6.11-12
Activity Release Timing	
Gap Release	Starting at 30 seconds, Ending at 30 minutes
Fuel Release	1.3 hours (ending at 1.8 hours)
Activity Release from the Fuel	
Noble Gases	5% gap, 95% fuel (100% total)
Iodines	5% gap, 35% fuel (40% total)
Alkali Metals	5% gap, 25% fuel (30% total)
Tellurium Metals	0% gap, 5% fuel (5% total)
Barium, Strontium	0% gap, 2% fuel (2% total)
Noble Metals	0% gap, 0.25% fuel (0.25% total)
Cerium Group	0% gap, 0.05% fuel (0.05% total)
Lanthanides	0% gap, 0.02% fuel (0.02% total)
Iodine Chemical Form in Containment	4.85% elemental, 0.15% organic and 95% particulate
Iodine Chemical Form Released to Atmosphere from ECCS Leakage	97% elemental, 3% organic
Containment Release Path	
Containment Net-Free Volume	2.61E6 (ft^3)
Sprayed Fraction	0.8
Containment Leak Rates	
0 - 24 hours	0.1 (weight %/day)
> 24 hours	0.05 (weight %/day)
Fan Cooler Flow Rate	64,500 cfm/unit
Number of Fan Coolers Credited	3
Time to Start Fan Coolers	1 minute

Table 6.11-21 (Cont.) Assumptions Used for LBLOCA Analysis	
Spray Operation	
Time to Initiate Sprays	1 minute
Spray Injection Duration	37.8 minutes
Termination of Spray Recirculation	3.4 hours
Injection Spray Flow Rate	2135 gpm
Recirculation Spray Flow Rate	1080 gpm
Spray Fall Height	118.5 feet
Removal Coefficients	
Elemental Iodine Injection Spray Removal	20.0 hr^{-1}
Particulate Injection Spray Removal	4.4 hr^{-1}
Elemental Iodine Recirculation Spray Removal	5.0 hr^{-1}
Particulate Recirculation Spray Removal	2.25 hr^{-1}
Sedimentation Particulate Removal in Unsprayed Region and in Sprayed Region after Spray Termination	0.1 hr^{-1}
ECCS Leakage Release Path	
Credited Sump Mass	3.12E6 lbm
ECCS Leak Rate to Auxiliary Building	
0 - 6.5 hours	0 gal/hr
> 6.5 hours	4 gal/hr
Iodine Airborne Fraction for ECCS Leakage to Auxiliary Building	
6.5 - 8 hours	0.120
8 - 24 hours	0.0855
24 - 96 hours	0.0523
96 - 720 hours	0.0300
Filtration of Activity Released by ECCS Leakage Outside Containment	Not credited

Table 6.11-21 (Cont.) Assumptions Used for LBLOCA Analysis	
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Time to Start Crediting Emergency Control Room HVAC	1 minute
Control Room Atmospheric Dispersion (χ/Q) Factors	
Containment Releases:	
0 - 2 hours	3.82E-4 sec/m^3
2 - 8 hours	2.81E-4 sec/m^3
8 - 24 hours	1.05E-4 sec/m^3
24 - 96 hours	8.31E-5 sec/m^3
96 - 720 hours	7.04E-5 sec/m^3
ECCS leakage:	
0 - 2 hours	6.44E-4 sec/m^3
2 - 8 hours	4.69E-4 sec/m^3
8 - 24 hours	1.72E-4 sec/m^3
24 - 96 hours	1.37E-4 sec/m^3
96 - 720 hours	1.17E-4 sec/m^3

Table 6.11-22 Assumptions Used for GDT Rupture Dose Analysis	
Nuclide Parameters	See Table 6.11-9
GDT Inventory (Dose Equivalent Xe-133)	29,761 Ci
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Control Room Atmospheric Dispersion (χ/Q) factors	
Containment Vent Releases:	
0 - 2 hours	6.44E-4 sec/m^3
Time to Start Crediting Emergency Control Room HVAC	Control room is not isolated

Table 6.11-23 Assumptions Used for VCT Rupture Dose Analysis	
Nuclide Parameters	See Table 6.11-9
VCT Inventory (Ci)	See Table 6.11-6
Duration of Activity Release from Tank	5 minutes
Iodine Partition Coefficient for VCT Liquid	0.01
Primary Coolant Noble Gas Activity	1.0% fuel defect level (See Table 6.11-13)
Primary Coolant Initial Iodine Activity	60 μCi/gm of DE I-131 (See Table 6.11-14)
Letdown Flow Rate	132 gpm
Iodine Partition Coefficient for Letdown Releases	0.1
Letdown Line Demineralizer DF for Iodine	10
Time to Isolate Letdown Flow	30 minutes
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Control Room Atmospheric Dispersion (χ/Q) factors	
Containment Vent Releases:	
0 - 2 hours	6.44E-4 sec/m^3
Time to Start Crediting Emergency Control Room HVAC	Control room is not isolated

Table 6.11-24 Assumptions Used for HT Failure Dose Analysis	
Nuclide Parameters	See Table 6.11-9
HT Inventory (Ci)	See Table 6.11-7
Duration of Activity Release from Tank	5 minutes
Iodine Partition Coefficient for HT Liquid	0.01
HT Volume	8100 ft^3
HT Full Level	80%
Primary Coolant Noble Gas Activity	1.0% fuel defect level (See Table 6.11-13)
Primary Coolant Initial Iodine Activity	60 μCi/gm of DE I-131 (See Table 6.11-14)
Letdown Flow Rate	132 gpm
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Control Room Atmospheric Dispersion (χ/Q) factors	
Containment Vent Releases:	
0 - 2 hours	6.44E-4 sec/m^3
Time to Start Crediting Emergency Control Room HVAC	Control room is not isolated

Table 6.11-25 Assumptions Used for FHA Analysis	
Source Term	
Nuclide Parameters	See Table 6.11-9
Average Assembly Fission Product Activity	See Table 6.11-26
Radial Peaking Factor	1.70
Fuel Rod Gap Fraction	
I-131	12%
Kr-85	30%
Other Iodines and Noble Gases	10%
Fuel Damaged	One assembly
Time after Shutdown	84 hours
Water Depth	23 feet
Overall Iodine Scrubbing Factor	200
Noble Gases Scrubbing Factor	1
Filter Efficiency	No filtration of releases assumed
Isolation of Release	No isolation of releases assumed
Time to Releases All Activity	2 hours
Offsite Atmospheric Dispersion Factors	See Table 6.11-10
Offsite Breathing Rates	See Table 6.11-10
Control Room Model	See Table 6.11-11
Time to Start Crediting Emergency Control Room HVAC	No isolation of control room assumed
Control Room Atmospheric Dispersion (χ/Q) Factors	
Containment Vent:	
0 - 2 hours	6.44E-4 sec/m^3
2 - 8 hours	4.69E-4 sec/m^3
8 - 24 hours	1.72E-4 sec/m^3
24 - 96 hours	1.37E-4 sec/m^3
96 - 720 hours	1.17E-4 sec/m^3

Table 6.11-26 Average Fuel Assembly Fission Product Inventory 84 Hours after Shutdown Based on 3280.3 MWt (102% of 3216 MWt)	
	Isotopic Inventory, curies
Iodine	
I-130	3.44E4
I-131	6.94E7
I-132	6.39E7
I-133	1.17E7
I-134	0.00E0
I-135	2.62E4
Noble Gases	
Kr-85m	0.00E0
Kr-85	1.10E6
Kr-87	0.00E0
Kr-88	0.00E0
Xe-131m	9.85E5
Xe-133m	2.91E6
Xe-133	1.36E8
Xe-135m	4.20E3
Xe-135	7.83E5
Xe-138	0.00E0

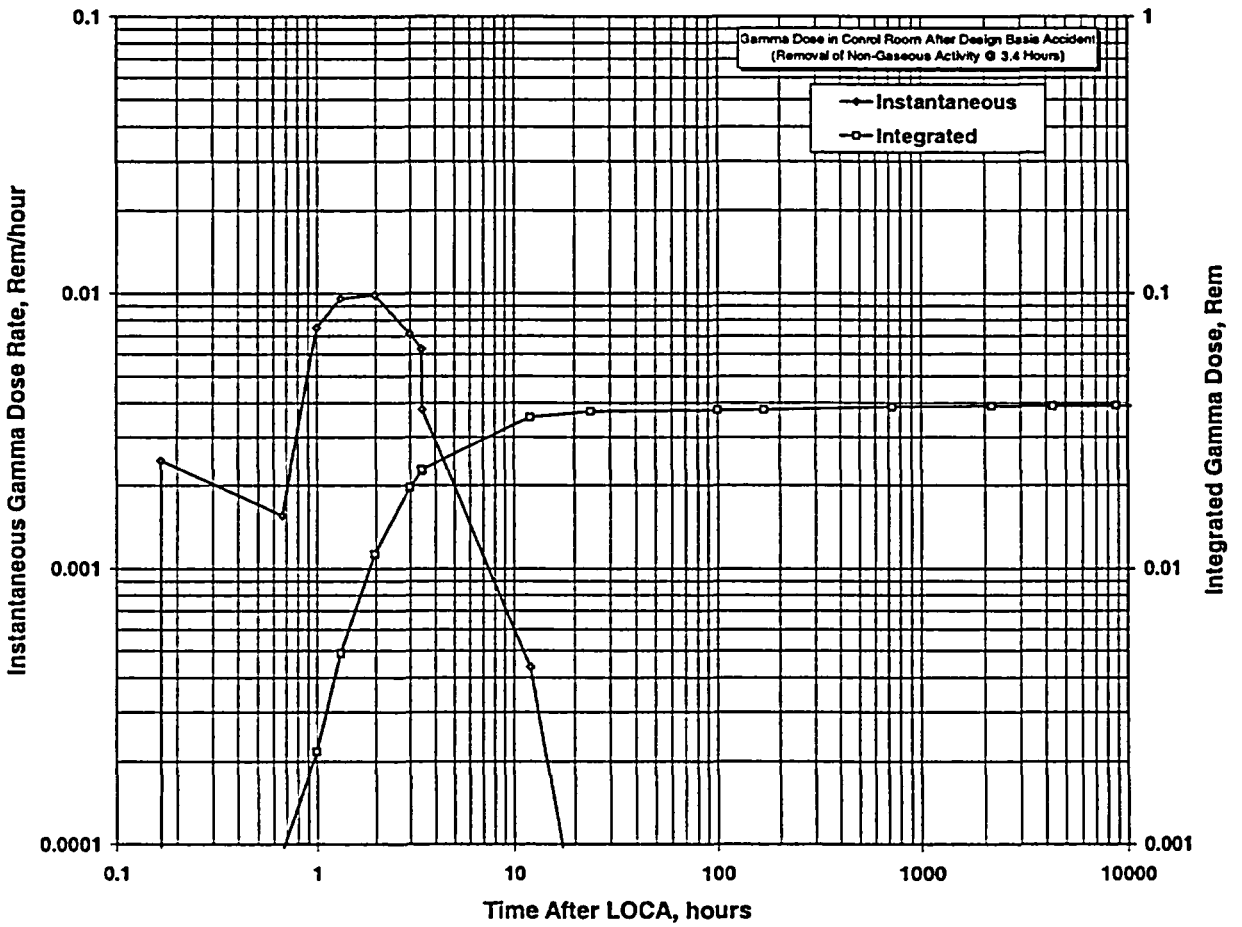


Figure 6.11-1

Direct Gamma Dose Rate and Integrated Dose in the Control Room Following a DBA

6.12 EOPs and EOP Setpoints

As a result of the Indian Point Unit 2 (IP2) Stretch Power Uprate (SPU) Program, the plant operating parameters have changed from the current design parameters. These include parameters which affect analyses and evaluations for plant operations and for plant accident responses. As a result of the parameter revisions, EOP setpoints specified by the IP2 Emergency Operating Procedures (EOPs) were reviewed to determine the potential effect from the changed power uprating parameters. Once this list of EOP setpoints was established, the new EOP setpoint calculations were performed.

To further ensure that the EOP setpoint documentation met the current generic requirements of Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs), all relevant ERG Maintenance Direct Work Items (DWs) approved through August 2003, were reviewed and necessary changes incorporated into the IP2 EOP setpoints and corresponding EOPs.

Based on the identified EOP setpoint changes, the IP2 EOPs were reviewed to identify changes resulting from the changed power uprating parameters and corresponding EOP setpoint changes.

These changes will be incorporated into the IP2 EOPs for use in the operator training program and in plant operations when the SPU is implemented.

7.0 NUCLEAR FUEL

This chapter discusses the analyses performed in support of the Indian Point Unit 2 (IP2) Stretch Power Uprate (SPU) Program in the nuclear fuel and fuel-related areas. Specifically, it addresses fuel thermal-hydraulic design, fuel core design, fuel rod performance, neutron fluence, and heat generation rates. The results and conclusions of each analysis can be found within the applicable subsection.

IP2 is currently operating in Cycle 16 with 15 x 15 VANTAGE+ fuel assemblies. Commencing in Cycle 17, it is planned to refuel with a 15 x 15 upgraded fuel assembly with modified fuel rod support surfaces of the mid-grids and intermediate flow mixing (IFM) grids to enhance margin to grid-to-rod fretting. For the purposes of the SPU analysis, fuel-related safety and design parameters have been chosen to bound the current VANTAGE+ fuel and the upgraded fuel assembly. These bounding parameters have been used in the safety and design analyses discussed in this section and in other sections of this report. If Entergy chooses to implement the upgraded fuel design for Cycle 17, licensing of this upgraded design will occur according to the NRC-approved Westinghouse Fuel Criteria Evaluation Process (FCEP) described in WCAP-12488-P-A. Furthermore applicability of the SPU safety analysis for the 15 x 15 upgraded fuel assembly will be evaluated or re-analyzed during the Cycle 17 reload safety evaluation in accordance with the reload safety evaluation methodology described in WCAP-9272-P-A.

Sections 7.1 through 7.4 discuss the results of analyses and evaluations that have been performed to show that the fuel and core designs as represented by the bounding parameters meet the acceptance criteria. The results of these analyses or evaluations will be reviewed and evaluated for each operating cycle as part of the cycle-specific reload safety evaluation in accordance with the reload safety evaluation methodology described in WCAP-9272-P-A. The cycle-specific reload safety evaluation will provide the technical and licensing bases for operation of the specific cycle at the licensed power level.

7.1 Fuel Design Features and Components

Fuel assemblies are designed to perform satisfactorily throughout their lifetime. The combined effects of the design basis loads are considered in evaluating the capability of fuel assemblies and their components to maintain structural integrity. This is necessary so that fuel assembly functional requirements are met while maintaining the core coolable geometry and the ability for reactor core safe shutdown.

The stretch power uprate (SPU) conditions result in changes to temperatures that affect loss-of-coolant accident (LOCA) forces. LOCA force changes result in changes to core plate motions, the effects of which have been incorporated into the analyses for the fuel assemblies. The SPU core power uprating does not increase operating or transient loads such that they will adversely affect fuel assembly functional requirements. Fuel assembly structural integrity is not affected and the core coolable geometry is maintained for the 15 x 15 VANTAGE+ (Zirlo™ with 0.422 rod and debris mitigating features) fuel assembly design and the 15 x 15 upgraded fuel assembly for Indian Point Unit 2 (IP2).

The lift forces specified for the SPU Program are slightly higher than the lift forces from previous IP2 analyses. Analyses verified the fuel assembly holddown spring capability to maintain contact between the fuel assembly and the lower core plate at normal operating conditions. Thus, fuel assembly structural integrity is not affected by the SPU.

Other areas, such as fuel rod fretting, oxidation and hydriding of thimbles and grids, fuel rod growth gap, and guide thimble wear, were determined to have an acceptable effect. It is concluded that the fuel assemblies are in conformance with all regulatory criteria at the SPU conditions.

7.2 Core Thermal-Hydraulic Design

7.2.1 Introduction

This section describes the core thermal-hydraulic analyses and evaluations performed in support of Indian Point Unit 2 (IP2) operation at an stretch power uprate (SPU) core power level of 3216 MWt over a range of Reactor Coolant System (RCS) temperatures (Table 2.1-2 in Section 2 of this report).

7.2.2 Input Parameters and Assumptions

Table 7.2-1 summarizes the thermal-hydraulic design parameters used in the departure from nucleate boiling ratio (DNBR) analyses. The core inlet temperature used in the DNBR analyses is based on the upper bound of the RCS temperature range for the SPU conditions. Use of the upper bound temperature is conservative for the DNBR analyses. The DNBR analyses also assume that the SPU core designs are composed of 15 x 15 VANTAGE+ and 15 x15 upgraded fuel assemblies.

7.2.3 Description of Analyses and Evaluations

7.2.3.1 Calculation Methods

The thermal-hydraulic design criteria and methods for the SPU remain the same as those presented in the *IP2 Updated Final Safety Analysis Report* (UFSAR) and the 1.4-percent Measurement Uncertainty Recapture (MUR) Report (References 1 and 2). The WRB-1 DNB correlation and the Revised Thermal Design Procedure (RTDP) DNB methodology (Reference 3) continue to be used for the SPU DNB analysis with the 15 x 15 VANTAGE+ and upgraded fuel assemblies. The W-3 DNB correlation is used for events where the conditions fall outside the applicable range of the WRB-1 correlation. The Westinghouse version of the VIPRE-01 (VIPRE) code (Reference 4) is used for DNBR calculations with the WRB-1 and the W-3 DNB correlations. The use of VIPRE for the SPU analysis is in full compliance with the conditions specified in the NRC *Safety Evaluation Report* (SER) in WCAP-14565-P-A (Reference 4).

With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB sensitivity factors. Based on the DNB sensitivity factors, RTDP design limit DNBR values were determined such that there was at least a 95-percent probability at a 95-percent confidence level that DNB would not occur on the most limiting fuel rod during normal operation, operational transients, or transient conditions arising from faults of moderate frequency (Condition I and II events as defined in the IP2 USFAR [Reference 1]).

Uncertainties in plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and RCS flow) are considered in the RTDP DNBR analysis. Only the random portion of each plant operating parameter uncertainty is included in the statistical combination for RTDP. Any adverse instrumentation bias is treated either as a direct DNBR penalty or a direct analysis input.

The RTDP design limit DNBR values specified in the 1.4-percent MUR report (Reference 2) for IP2 were revised for the SPU to 1.22 (for both thimble and typical cells).

In addition to the above considerations for uncertainties, DNBR margin was obtained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. Sufficient DNBR margin was conservatively maintained in the safety analysis DNBR limits to offset the rod bow, transition core, and plant operating parameter bias DNBR penalties. The net remaining DNBR margin, after considering penalties, is available for operating and design flexibility.

As noted in the USFAR and in the 1.4-percent MUR Report (References 1 and 2), the Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

7.2.3.2 DNB Performance

The current DNBR analyses of record for IP2 are primarily those that were performed to support the SPU using VANTAGE+ fuel. All DNBR analyses performed for the SPU for a core power level of 3216 MWt are bounding for operation using both 15x15 VANTAGE+ and upgraded fuels. A comparison of the current thermal-hydraulic parameters and the SPU parameters is shown in Table 7.2-1.

To support the operation of IP2 at SPU conditions, DNBR reanalysis was required to define new core limits, axial offset limits, and Condition II accident acceptability. The accident DNB analyses to support the SPU are addressed below.

7.2.3.2.1 Loss of Flow

The DNB analysis of the loss-of-flow accident was performed for SPU conditions. Three cases, including partial loss of flow (PLOF), complete loss of flow (CLOF), and CLOF-under frequency (CLOF-UF) were checked to ensure the limiting scenario was identified. The effect of updated fuel temperatures was included in the analysis of this event (subsection 7.2.3.3). The CLOF-UF case resulted in the lowest minimum DNBR. The minimum DNBRs calculated for each of the 3

cases were greater than the new safety analysis DNBRs, thereby demonstrating compliance to the DNB design criterion for this event.

7.2.3.2.2 Locked Rotor

The analysis of the locked rotor accident was performed for SPU conditions. The locked rotor accident is classified as a Condition IV event. To calculate the radiation release as a consequence of the accident, DNB calculations were performed to quantify the inventory of rods that would experience DNB and be conservatively presumed to fail. For IP2, the analysis indicates that there would be no rods in DNB due to the locked rotor accident. The radiological consequences analysis conservatively assumed 5 percent of the fuel rods as failed rods and showed that the site dose limits were met (see subsection 6.11.9 of this report).

7.2.3.2.3 Feedwater Malfunction

The core response for the feedwater malfunction event at hot zero power (HZP) was bounded by the steamline break core response. All DNBR design criteria are met for the feedwater malfunction event at zero power. The feedwater malfunction at hot full power conditions is presented in subsection 6.3.9 of this report.

7.2.3.2.4 Dropped Rod

Dropped rod limit lines were calculated to address the acceptability of the plant's response to this accident scenario. The limit lines were calculated based on the reference power shape. The loci of points that would result in the safety-limit DNBR being reached were defined for a wide span of core conditions (inlet temperature, power, and pressure).

The effects on core conditions, including power distribution, are demonstrated to remain within the bounds represented by the dropped rod limit lines. There was no explicit DNBR calculation performed for the dropped rod event. The SPU core design met the limit lines. Calculation of the effects of the accident on the core was checked cycle-by-cycle, inferring compliance to the DNB criterion for each cycle.

7.2.3.2.5 Steamline Break

The DNB analysis of the steamline break event was performed for SPU conditions. Cases were analyzed for both HZP and hot full power (HFP) preconditions. For each of these cases, an appropriate methodology was applied.

For the HFP cases, the RTDP methodology was used. For acceptability, calculated DNBRs must be above the design limit DNBR defined by a convolution of uncertainties on core condition parameters. For the HZP cases, the RDTP methodology was not appropriate, so the mechanistic STDP was applied. For the STDP application, the W-3 DNBR correlation limit for this transient is 1.45. The calculated minimum DNBR is reduced to account for any DNBR penalties applicable at this transient condition.

The minimum DNBRs calculated for both these accident cases were well above the safety analysis DNBRs for their respective bases.

7.2.3.2.6 Rod Withdrawal from Subcritical

The DNB analysis of the rod withdrawal from subcritical accident was performed for SPU conditions.

By nature of the accident, a bottom-skewed power shape was conservatively applied. A power excursion, due to the removed rod bank, would develop more prominently in the lower part of the core. For this calculation, a conservative generic power shape was applied. To preserve applicability of the critical heat flux correlation, two calculations were required for this accident. For fuel assembly spans below the first mixing vane grid, the W-3 correlation was applied. For fuel assembly spans above the mixing grid, the WRB-1 correlation was applied, consistent with other DNBR confirmation calculations. Also, because of the zero power precondition of this event, the methodology that convolutes uncertainty terms to set limits was not appropriate, so the mechanistic STDP was applied. For the STDP application, the DNBR limit applied was the correlation limit DNBR, since uncertainties were mechanistically applied on the calculation input. For the W-3 correlation, this value was 1.30. For the WRB-1 correlation, this value was 1.17.

Calculations have been completed for each span and the results showed that the predicted DNBR remained above the respective correlation limit DNBR, thereby demonstrating compliance to the DNB design criterion for this event.

7.2.3.3 Fuel Temperatures and Rod Internal Pressures

The fuel temperatures and rod internal pressures for the SPU safety analysis for VANTAGE+ and upgraded fuel were based on ZIRLO cladding design. The NRC-approved Westinghouse PAD 4.0 fuel performance models (References 5 and 6) were used in the fuel temperature and rod internal pressure analyses. The integral fuel burnable absorber (IFBA) and non-IFBA fuel temperatures and/or rod internal pressures were used as initial conditions for LOCA and non-LOCA transients. Also, based on the fuel temperature analysis, the linear power limit to preclude fuel centerline melting was determined to be 22.7 kW/ft at the SPU conditions.

7.2.4 Acceptance Criteria

The acceptance criteria are contained in each subsection under subsection 7.2.3.2, of this report.

7.2.5 Results/Conclusions

Core thermal-hydraulic analyses and evaluations were performed in support of IP2 operation at the SPU core power level of 3216 MWt over a range of RCS temperatures. The results showed that the core thermal-hydraulic design criteria listed in the UFSAR (Reference 1) are satisfied.

7.2.6 References

1. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report,* Docket No. 50-247.

2. *Indian Point Nuclear Generating Unit No. 2, 1.4-Percent Measurement Uncertainty Recapture Power Uprate License Amendment Request Package*, Entergy Nuclear Operations, Inc., November 2002.

3. WCAP-11397-A (Nonproprietary) and WCAP-11397-P-A (Proprietary), *Revised Thermal Design Procedure*, A. J. Friedland and S. Ray, April 1989.

4. WCAP-14565-A (Proprietary) and WCAP-15306 (Nonproprietary), *Vipre-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Y. X. Sung, et al., October 1999.

5. WCAP-15063-P-A, *Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)*, Foster, Sidener, and Slagle, Rev. 1 with Errata, July 2000.

6. WCAP-12610-P-A, *VANTAGE+ Fuel Assembly Reference Core Report*, S. L. Davidson and T. L. Ryan, April 1995.

Table 7.2-1 Thermal-Hydraulic Design Parameters for IP2		
Thermal-Hydraulic Design Parameters	**Current**	**SPU**
Reactor Core Heat Output, MWt	3114.4	3216
Reactor Core Heat Output, 10^6 Btu/hr	10,670	10,973
Heat Generated in Fuel, %	97.4	97.4
Pressurizer Pressure, Nominal, psia	2250	2250
$F_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor	1.70	1.70
Part Power Multiplier for $F_{\Delta H}$	[1+0.3(1-P)]	[1+0.3(1-P)]
Minimum DNBR at Nominal Conditions (using RTDP)		
Typical Flow Channel	2.47	2.50[1]
Thimble (cold wall) Flow Channel	2.38	2.40[1]
Design Limit DNBR		
Typical Flow Channel	1.26	1.22
Thimble (cold wall) Flow Channel	1.25	1.22
DNB Correlation[2]	WRB-1	WRB-1
Vessel Inlet Minimum Measured Flow Rate, MMF, (including bypass) gpm	330,000	348,300
Vessel Inlet Thermal Design Flow Rate, TDF, (including bypass) gpm	322,800	322,800
Core Inlet Flow Rate (excluding total bypass, based on TDF) gpm	301,818	301,818
Fuel Assembly Flow Area for Heat Transfer, ft^2	51.54	51.54
Core Inlet Mass Velocity (based on TDF), ft/sec	13.78	13.80
Nominal Vessel/Core Inlet Temperature, °F	546.7	538.2
Vessel Average Temperature, °F	579.2	572.0
Core Average Temperature, °F	583.0	575.9
Vessel Outlet Temperature, °F	611.7	605.8
Average Temperature Rise in Vessel, °F	65.0	67.6
Average Temperature Rise in Core, °F	69.0	71.8

Table 7.2-1 (Cont.) Thermal-Hydraulic Design Parameters for IP2		
Thermal-Hydraulic Design Parameters	**Current**	**SPU**
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	52,100	52,100
Average Heat Flux, Btu/hr-ft^2	198,800	205,243
Average Linear Power, kw/ft	6.43	6.64
Peak Linear Power for Normal Operation, kw/ft	16.11[3]	16.61[3]
Temperature Limit for Prevention of Centerline Melt, °F	4700	4700

Notes:

1. The minimum nominal DNBRs are conservatively listed for both VANTAGE+ and upgraded fuel.
2. See subsection 3.2.2.4 of Reference 1 for the use of the W-3 DNB correlation.
3. This power level is based on a peaking factor (F_Q) of 2.5.

7.3 Fuel Core Design

7.3.1 Introduction

The nuclear design portion of the Indian Point Unit 2 (IP2) Stretch Power Uprate (SPU) Program core analysis determined the effect of the uprate on the key safety parameters. These safety parameters were used as input to the Indian Point Unit 2 *Updated Final Safety Analysis Report* (UFSAR) (Reference 1) Chapter 14 accident analyses.

7.3.2 Input Parameters and Assumptions

The nuclear design analyses demonstrated the acceptability of operation at the SPU core power level of 3216 MWt consistent with parameters in Section 2 of this report.

7.3.3 Description of Analyses and Evaluations

To satisfy these objectives, conceptual models were developed that followed the uprate transition to an equilibrium cycle. Fuel management strategies similar to those used in recent cycles were assumed in developing the models. The SPU assumed a core thermal power level of 3216 MWt during the 3 transition cycles. Key safety parameters were then evaluated to determine the expected ranges of variation in the parameters. The key safety parameters are those described in the standard reload design methodology (Reference 2). Some of these parameters, such as shutdown margin, were sensitive to the fuel management and loading pattern characteristics.

The observed variation in the parameters that were sensitive to loading patterns at SPU conditions were typical of the normal cycle-to-cycle variations for non-transition fuel reloads. Many of the key safety parameters were dependent on the loading patterns.

7.3.3.1 Methodology

All nuclear design analysis in support of the IP2 SPU Program was performed using standard Westinghouse core reload methodology described in WCAP-9272-P-A (Reference 2) with the Westinghouse PHOENIX-P and ANC codes described in WCAP-11596-P-A and WCAP-10965-P-A (References 3 and 4). These licensed methods and models have been used for IP2 and other previous Westinghouse reload fuel designs with and without uprating. No changes to the nuclear design philosophy, methods, or models, are necessary due to the SPU.

The reload design philosophy used by Westinghouse includes an evaluation of the reload core key safety parameters that comprises the nuclear-design-dependent input to the reload fuel

safety evaluation for each reload cycle. This philosophy is described in WCAP-9272-P-A (Reference 2). These key safety parameters will be evaluated for each IP2 reload cycle. If one or more of the key parameters fall outside the bounds assumed in the safety analyses, the affected transients will be reevaluated and the results documented in the Reload Safety Evaluation Report (RSE) for that cycle. The main objective of the uprating core analyses was to determine, prior to the cycle-specific reload design, if the previously used bounds for the key safety parameters remained applicable. The results of these analyses are described below.

7.3.3.2 Physics Characteristics and Key Safety Parameters

Conceptual core loading patterns were constructed to be representative of future IP2 cores. Table 7.3-1 compares the safety parameter ranges considered for the IP2 current designs and for the SPU.

The comparison in Table 7.3-1 shows that the SPU core did not have any marked deviations from the core design at 3114.4 MWt. Of note is a small change in the minimum beta effective to provide margin for the end of life, hot zero power (HZP) rod ejection event. This parameter was confirmed for each cycle to be less than the safety analysis value.

Shutdown margin and maximum boron concentrations are two parameters that are loading-pattern-dependent and the core design must be developed such that these constraints are met. The shutdown margin requirement of 1300 pcm is primarily a function of the power defect from full power to HZP at the time of trip, and the type of fuel that is placed under control rod locations. The power defect is set by the enrichments required to achieve the design cycle length and the operating temperature. The core design can govern the amount of shutdown margin by increasing the amount of fresh fuel in control rod locations. Since the SPU conditions significantly increase the power defect, the required amount of shutdown margin is a loading pattern constraint that must be met in order to consider the loading pattern acceptable. Maximum boron concentration is a function of the feed enrichment needed to achieve the cycle lifetime but also of the fuel management strategy used for the loading pattern. As the maximum boron concentrations are initial or final conditions, they are also a design constraint that must be considered at the time of loading pattern development.

7.3.3.3 Power Distributions and Peaking Factors

Loading patterns were developed and modeled based on the projected energy requirements for the SPU. These models were not intended to represent limiting loading patterns but were developed with the intent to show that enough margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores.

7.3.3.4 Radial Power Distribution Effects

Assembly average powers at beginning of life (BOL), middle of life (MOL), and end of life (EOL) were calculated using the SPU core models for different fuel management techniques. The effect on the radial power distribution due to the SPU conditions was small when compared to loading patterns for similar fuel management practices at nominal power conditions. The effects of these radial power distribution differences on rod worths and on off-nominal condition peaking factors were small and were well within normal cycle-to-cycle variation in these parameters.

7.3.3.5 Axial Power Distribution and FQ(z) Effects

The axial power distribution effect of the SPU conditions shows only a small sensitivity to the uprate.

As part of the reload design process, a cycle-specific final acceptance criteria (FAC) analysis based on constant axial offset control (CAOC) operation (Reference 5) check is performed that implicitly includes the axial effects of the uprating. Load follow simulations were performed through the power range to generate axial power shapes that were typical of Condition I operation. The results of the FAC analysis for this report showed that the total peaking factor (FQ) was acceptable. Therefore, it is expected that all reload cores at SPU conditions will also be acceptable.

7.3.4 Conclusions

In summary, implementing the SPU will not cause changes to the current nuclear design bases given in the UFSAR. The effect of the SPU on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters will be well within normal cycle-to-cycle variation of these values or controlled by the core design, and will be addressed on a cycle-specific basis, consistent with the reload safety evaluation methodology (Reference 2). The ranges of key safety parameters as reported in Table 7.3-1 remain valid and bounding for the SPU.

7.3.5 References

1. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report*, Docket No. 50-247.

2. WCAP-9272-P-A, *Westinghouse Reload Safety Evaluation Methodology*, S. L. Davidson et al., July 1985.

3. WCAP-11596-P-A, *Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores*, T. Q. Nguyen et al., June 1988.

4. WCAP-10965-P-A, ANC: *A Westinghouse Advanced Nodal Computer Code*, Y. S. Liu et al., September 1986.

5. WCAP-8385 (Proprietary), *Power Distribution Control and Load Following Procedures*, T. Morita et al., September 1974.

Table 7.3-1 IP2 SPU Program Key Safety Parameters		
Safety Parameters	**Current Values**	**SPU**
Reactor Core Power (MWt)	3114.4	3216
Vessel T_{avg} HFP (°F)	549.4 to 579.2	549.0 to 572.0
RCS Pressure (psia)	2250	2250
Core Average Linear Heat Rate (Kw/ft)	6.44	6.64
Most Positive MTC (pcm/°F)	0.0	0.0
Most Positive MDC (delta-k/gm/cc)	0.50	0.50
Doppler Temperature Coefficient (pcm/°F)	-0.9 to -3.2	-0.9 to –3.2
Doppler Only Power Coefficients (pcm/% power) Least Negative	-6.36 to –9.55	-6.36 to -9.55
Doppler Only Power Coefficients (pcm/% power) Most Negative	-13.26 to -19.4	-13.26 to -19.4
Beta-Effective	0.0040 to 0.0070	0.0042 to 0.0070
Shutdown Margin (pcm)	1300	1300
Nuclear Design $F_{\Delta H}$	1.574/1.70	1.574/1.70

7.4 Fuel Rod Design and Performance

7.4.1 Introduction

Fuel rod design analyses were performed to assess the potential effects that the SPU operating conditions for Indian Point Unit 2 (IP2) would have on meeting fuel rod design criteria.

7.4.2 Description of Analyses, Acceptance Criteria, and Results

The fuel rod design analyses modeled 15 x 15, 8-inch annular blanket, 1.25X integral fuel burnable absorber (IFBA), ZIRLO™ clad fuel rods irradiated for up to 4 cycles at SPU conditions.

Based on the history of IP2 and the limited margin available, operation at a vessel average temperature of 562±3°F is recommended to avoid potential clad fatigue and rod internal pressure violations for operation at the SPU power level. Representative rod power histories and axial power shapes, generated by the NRC-approved Advanced Nodal Code (ANC) (References 1 and 2) were analyzed. The NRC-approved Westinghouse PAD 4.0 fuel performance models (References 3 and 4) were also used in the analyses. PAD is the main design tool for evaluating fuel rod performance, calculating the inter-related effects of temperature, pressure, clad elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power.

The following sections summarize the effect of the core power uprating on the fuel rod design criteria most affected by the SPU core power. The fuel rod design criteria affected were rod internal pressure, clad corrosion, clad stress, and clad strain criteria. Other fuel rod design criteria were not significantly affected by a core power uprating.

7.4.2.1 Rod Internal Pressure

Design Basis

The fuel system will not be damaged due to excessive fuel rod internal pressure.

Acceptance Limit

The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward clad creep during steady state operation or cause extensive departure from nucleate boiling (DNB) propagation to occur.

Design Evaluation

The analyses showed that meeting the rod internal pressure criterion was most affected by the SPU increase in core power level. The higher power levels resulted in higher fuel operating temperatures with a potential for increased fission gas release. Analysis of the representative rod power histories indicated that the higher duty fuel rods have this potential for increased fission gas release resulting in higher rod internal pressures. The IFBA loading was reduced from 1.5X to 1.25X to meet the rod internal pressure criterion. A vessel average temperature of 572°F showed less than 100-psi margin to the design criteria limit. The rod internal pressure criterion can be met under uprated core conditions with a vessel average temperature of 562±3°F by appropriate cycle-specific core design.

7.4.2.2 Clad Corrosion

Design Basis

The fuel system will not be damaged due to excessive fuel clad oxidation. The fuel system will be operated to prevent significant degradation of mechanical properties of the clad at low temperatures, due to hydrogen embrittlement caused by formation of zirconium hydride platelets.

Acceptance Limit

The calculated fuel clad temperature (metal-oxide interface temperature) will be less than the license limit for ZIRLO clad fuel during steady state operation. For Condition II events, the calculated fuel clad temperature will not exceed the license limit for ZIRLO clad fuel. The hydrogen pickup level in the fuel clad will be less than or equal to the license limit at the end of fuel operation.

Design Evaluation

The SPU conditions result in increased operating temperatures for the fuel clad due to the increased fuel rod average power rating. Since the corrosion process is a strong function of fuel clad temperature, the SPU will affect meeting these criteria. Analysis of the representative rod power histories indicated that the corrosion design criteria will be satisfied for the higher duty fuel rods at the SPU core conditions.

7.4.2.3 Clad Fatigue

Design Basis

The fuel system will not be damaged due to excessive fuel clad fatigue.

Acceptance Limit

The fatigue life usage factor will be less than 1.0 or, for a given strain range, the number of strain fatigue cycles will be less than those required for failure, considering a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is more conservative.

Design Evaluation

The Westinghouse PAD 4.0 fuel performance models (References 3 and 4) were used to evaluate fuel clad fatigue limits. Analysis of the representative rod power histories at the SPU conditions resulted in an increase in the clad fatigue levels. The combinations of long cycle lengths, high burnups, and the presence of cut pin penalties proved clad fatigue to be more limiting than previous reload designs. A vessel average temperature of 549°F resulted in violation of the clad fatigue criterion. The clad fatigue criterion can be met under SPU core conditions with a vessel average temperature of 562±3°F by appropriate cycle-specific core design.

7.4.2.4 Clad Stress and Strain Design Basis

The fuel system will not be damaged due to excessive fuel clad stress and strain.

Acceptance Limit

The volume-average effective stress calculated with the Von Mises equation, considering interference due to uniform cylindrical fuel pellet-clad contact, caused by fuel pellet thermal expansion, fuel pellet swelling, uniform fuel clad creep, and pressure differences, was less than the 0.2-percent offset yield stress with due consideration to temperature and irradiation effects under Condition II events. The acceptance limit for fuel rod clad strain during Condition II events is that the total tensile strain increase, due to uniform cylindrical fuel pellet thermal expansion during a transient, is less than 1 percent of the pre-transient value.

Design Evaluation

The Westinghouse PAD 4.0 fuel performance models (References 3 and 4) were used to evaluate fuel clad stress and strain limits. The local power duty during Condition II events was a key factor in evaluating the margin to fuel clad stress and strain limits. The fuel duty at the SPU conditions was more limiting, resulting in an increase in the cladding stress and strain levels. However, the fuel analyses results showed that the core power uprating will not affect the fuel's capability to meet the clad stress and strain limits.

7.4.3 Cycle-Specific Analyses

The fuel rod design criteria most affected by a change in core power rating have been evaluated. The evaluations indicated that all fuel rod design criteria can be met at the SPU core conditions with the proper cycle-specific core design.

Cycle-specific core designs and fuel performance analyses are performed for each reload cycle. These cycle-specific analyses are performed to ensure that all fuel rod design criteria will be satisfied for the specific operating conditions of that cycle.

Although the SPU analyses described in this section were performed for ZIRLO-clad fuel, the cycle-specific fuel performance analyses considered each specific fuel region (whether ZIRLO-clad fuel design or older fuel designs with different fuel features) in the core during that cycle. These analyses ensure that all fuel rod design criteria are met for each fuel region.

The cycle-specific fuel performance analyses considered any improved fuel performance models and methods licensed and approved by the NRC available at the time of the specific cycle design. These cycle-specific evaluations support the Reload Safety Evaluation (RSE) performed for each cycle of operation.

7.4.4 Conclusions

The fuel rod design criteria most affected by a change in core power rating have been analyzed. The results indicate that all fuel rod design criteria can be met at the SPU core conditions with the proper cycle-specific core design.

7.4.5 References

1. WCAP-11596-P-A, *Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores*, T. Q. Nguyen et al., June 1988.

2. WCAP-10965-P-A, ANC: *A Westinghouse Advanced Nodal Computer Code*, Y. S. Liu et al., September 1986.

3. WCAP-15063-P-A, *Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)*, Foster, Sidener, and Slagle, Rev. 1 with errata July 2000.

4. WCAP-12610-P-A, *VANTAGE+ Fuel Assembly Reference Core Report*, S. L. Davidson and T. L. Ryan, April 1995.

7.5 Neutron Fluence

7.5.1 Introduction

In the assessment of the state of embrittlement of light water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of the materials comprising the beltline region of the vessel is required. This exposure evaluation must, in general, include assessments not only at locations of maximum exposure at the inner radius of the vessel, but also as a function of axial, azimuthal, and radial location throughout the vessel wall.

In order to satisfy the requirements of 10CFR50, Appendix G (Reference 1), for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system (RCS), fast neutron exposure levels must be defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials comprising the beltline region. These locations are commonly referred to as the 1/4t and 3/4t positions in the vessel wall. The 1/4t exposure levels are also used in the determination of upper shelf fracture toughness as specified in 10CFR50, Appendix G. In the determination of values of reference temperature – pressurized thermal shock (RT_{PTS}) for comparison with the applicable PTS screening criterion as defined in 10CFR50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events*, (Reference 2) maximum neutron exposure levels experienced by each of the beltline materials are required. These maximum levels occur at the vessel inner radius.

The methodology used to determine the fast neutron (E > 1.0 MeV) exposure of the IP2 pressure vessel derives from the guidance provided in ASTM Standard E853, *Analysis and Interpretation of Light Water Reactor Surveillance Results*, and Regulatory Guide RG 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001 (Reference 3). The analytical methodology has received regulatory approval as documented in WCAP-14040-NP-A, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves*, January 1996 (Reference 4). The Westinghouse methodology has also been documented in WCAP-15557, Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology, August 2000 (Reference 5).

7.5.2 Description of Analysis/Evaluation and Input Assumptions

A three-dimensional (3-D) assessment of fast neutron exposures for the IP2 reactor geometry was made using discrete ordinates transport techniques. The analysis was based on a two-dimensional/one-dimensional (2D/1D) synthesis of neutron fluxes that were obtained from a series of plant- and cycle-specific forward transport calculations using r-θ, r-z, and r spatial

mesh. These transport calculations were subsequently compared against dosimetry results obtained from the in-vessel surveillance capsules withdrawn to date at IP2 in order to demonstrate that the plant-specific analysis meets the 20-percent uncertainty criterion specified in RG 1.190; however, these comparisons only serve to validate the calculational model and are not used in any way to modify the calculational results.

The generalized equation that was used to assess the fast neutron flux in the reactor pressure vessel, which is described in RG 1.190, is given as:

$$\phi_g(r,\theta,z) = \phi_g(r,\theta) \times \frac{\phi_g(r,z)}{\phi_g(r)}$$

where

$\phi_g(r,\theta)$	=	The group g transport solution in r,θ geometry for a representative axial plane, that is, at the core midplane.
$\phi_g(r)$ and $\phi_g(r,z)$	=	The 1-D and 2-D group g flux solutions whose ratio is used to determine a group-dependent axial shape factor.

The fast neutron exposure calculations were carried out using the DORT (DOORS 3.1 code package, Reference 6) discrete ordinates transport code in the forward mode and the BUGLE-96 cross-section library (Reference 7). This suite of codes has been used to support numerous pressure vessel fluence evaluations and are generally accepted by the Nuclear Regulatory Commission (NRC) for deterministic particle transport calculations, for example, neutron exposure and gamma-ray heating rate evaluations. All calculations were based on an S16 order of angular quadrature and a P5 expansion of the scattering cross-sections.

The core power distributions used in the plant-specific analysis were taken from the nuclear design reports for each of the first 16 operating fuel cycles at IP2. For future projections that support the IP2 Stretch Power Uprate (SPU) Program, core power distributions obtained from Westinghouse Core Engineering fuel management studies for Cycles 17 through 19 were used. The fast neutron transport calculations also account for several changes in core power during plant life. Specifically, reactor power increases from 2758 to 2948.54 MWt near the middle of Cycle 10, to 3071.4 MWt at the onset of Cycle 11, from 3071.4 to 3115 MWt at the middle of Cycle 16, and to 3216 MWt at the onset of Cycle 17, were assumed. Future projections beyond the end of Cycle 19 continued to be based on the Cycle 19 core power distributions; however, the source in the peripheral fuel assemblies was increased by an additional 5 percent, that is, relative to the actual Cycle 19 design, to account for small potential variations in core loading patterns being considered for use at that time.

7.5.3 Acceptance Criteria

Adequacy of the modeling is tested by comparing the calculated results against dosimetry measurements from surveillance capsules withdrawn from the plant. As long as these comparisons fall within the ± 20-percent criterion specified in RG 1.190, the calculational results are validated, that is, no specific acceptance criteria apply to the calculated values. However, these calculated results are used as input to reactor vessel analysis that is described in subsection 5.1.2 of this report.

7.5.4 Results

Comparisons of the measurement results from the in-vessel surveillance capsules withdrawn from the IP2 reactor versus the corresponding calculated predictions obtained at the measurement locations are presented in Table 7.5-1 for the fast neutron sensor reactions. An examination of the measurement/calculation (m/c) ratios of the fast neutron sensor reaction rates obtained from the surveillance capsule irradiations shows consistent behavior for all reactions at all capsule locations within the constraint of the allowable ± 20-percent (1σ) uncertainty in the final calculated results. Specifically, Table 7.5-1 shows that the average M/C ratios range from 1.03 to 1.12 for the individual capsules and that the overall average M/C ratio for the entire 13 foil data set is 1.07 with an associated sample standard deviation of 9.2 percent. Therefore, these comparisons of calculations with the surveillance capsule dosimetry sets withdrawn to date validate the neutron transport calculations performed to support this program and demonstrate that the uncertainty criterion of ± 20 percent (1σ), as specified by RG 1.190, has been satisfied for the IP2 reactor.

Therefore, based on this validation, the maximum calculated fast neutron fluence and displacement of atom (dpa) exposure values for the IP2 pressure vessel are provided in Table 7.5-2. As presented, these data represent the maximum exposure of the pressure vessel clad/base metal interface at azimuthal angles of 0, 15, 30, and 45 degrees relative to the core cardinal axes. The data tabulation includes the plant-specific calculated fluence at the end of Cycle 15 (EOC 15, the last cycle completed at IP2), the end of Cycle 16 (EOC 16, which is the current operating fuel cycle), and projections for future operation to 25.7 (EOC 19), 32, and 48 effective full-power years (EFPYs).

Based on the current NRC position of using the calculated values of neutron fluence to specify the neutron exposure for use in materials damage correlations, the calculated exposure values provided in Table 7.5-2 were provided for use in the materials properties assessments of the IP2 pressure vessel at SPU power conditions.

7.5.5 Conclusions

The calculated maximum pressure vessel neutron exposures that are presented in Table 7.5-2 were used as inputs to the reactor vessel evaluation described in subsection 5.1.2 of this report.

7.5.6 References

1. 10CFR50, Appendix G, *Fracture Toughness Requirements.*

2. 10CFR50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Federal Register*, Volume 60, No. 243, December 19, 1995.

3. NRC Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

4. WCAP-14040-NP-A, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves*, January 1996.

5. WCAP-15557, *Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology*, August 2000

6. RSICC Computer Code Collection CCC-650, DOORS 3.1, *One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System*, August 1996.

7. RSIC Data Library Collection DLC-185, *BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications*, March 1996.

Table 7.5-1

Comparison of Measured and Calculated Sensor Reaction Rate Ratios for the Fast Neutron Threshold Foil Reactions Obtained from In-Vessel Capsules Removed from Service at IP2

Capsule	M/C Ratio				Average	% Std. Dev.
	$^{63}Cu(n,\alpha)^{60}Co$	$^{54}Fe(n,p)^{54}Mn$	$^{58}Ni(n,p)^{58}Co$	$^{238}U(n,f)^{137}Cs$		
T	1.12	1.06	1.01	---	1.06	5.2
Y	1.16	1.01	0.91	---	1.03	12.3
Z	1.22	1.06	1.09	---	1.12	7.6
V	1.14	0.87	1.12	1.12	1.06	12.1
Average	1.16	1.00	1.03	1.12	**1.07**	**9.2**
% Std. Dev.	3.7	9.0	9.1	N/A		

Note:
The average and percent standard deviation values in bold face type represent the entire 13 sample threshold foil data set.

Table 7.5-2 Summary of Calculated Maximum Pressure Vessel Exposure at the Clad/Base Metal Interface for IP2				
Cumulative Operating Time (EFPY)	Neutron Fluence (n/cm^2) (E > 1.0 MeV)			
	0.0 Degrees	15.0 Degrees	30.0 Degrees	45.0 Degrees
18.7 (EOC 15)	2.78e+18	4.47e+18	5.39e+18	8.07e+18
20.6 (EOC 16)	3.00e+18	4.80e+18	5.84e+18	8.75e+18
25.7 (EOC 19)	3.60e+18	5.77e+18	7.08e+18	1.06e+19
32.0	4.45e+18	7.12e+18	8.74e+18	1.30e+19
48.0	6.62e+18	1.06e+19	1.30e+19	1.91e+19
	Iron Atom Displacements (dpa)			
18.7 (EOC 15)	4.49e-03	7.15e-03	8.68e-03	1.30e-02
20.6 (EOC 16)	4.85e-03	7.68e-03	9.40e-03	1.41e-02
25.7 (EOC 19)	5.83e-03	9.22e-03	1.14e-02	1.70e-02
32.0	7.21e-03	1.14e-02	1.41e-02	2.09e-02
48.0	1.07e-02	1.69e-02	2.09e-02	3.07e-02

7.6 Reactor Internals Heat Generation Rates

7.6.1 Introduction

The presence of radiation-induced heat generation in reactor internals components, in conjunction with the various reactor coolant fluid temperatures, results in thermal gradients within and between the components. These thermal gradients cause thermal stress and thermal growth, which must be considered in the design and analysis of the various components. The primary design considerations are to insure that thermal growth is consistent with the functional requirements of the components, and to insure that the applicable ASME Code requirements are satisfied as part of the components evaluation that is described in Section 5.2 of this report. In order to satisfy these requirements, the reactor internals must be analyzed with respect to fatigue and maximum allowable stress considerations.

The reactor internals components subjected to significant radiation-induced heat generation are the upper and lower core plates, lower core support, core baffle plates, former plates, core barrel, thermal shield, baffle-former bolts and barrel-former bolts. However, due to relatively low heat generation rates in the lower core support and the thermal shield, these components experience little, if any, temperature rise relative to the surrounding reactor coolant.

This section provides a description of the methodology that was used to determine the radiation-induced heat generation rates for the axial core components (the upper and lower core plates) and selected radial reactor internals components (the core baffle plates, core barrel and thermal shield) due to the core power uprate to 3216 MWt. Although design-basis neutron exposure data for the reactor internals components are documented in WCAP-9620, Revision 1 (Reference 1), key core power distribution, fuel product, and methodology differences presently exist such that the axial component data reported in WCAP-9620-R1 are non-conservative. However, as demonstrated in the Indian Point Unit 2 (IP2) plant-specific analysis performed to support the Stretch Power Uprate (SPU) Program, the radial component data from WCAP-9620-R1 remains conservative. Key axial components for the IP2 SPU Program were addressed using recently developed baseline upper and lower core plate heating rates applicable to IP2 (that is, four-loop design with 2-inch thick core plates).

7.6.2 Key Input Assumptions

For the core plates, baseline gamma heating rates were determined for both long- and short-term conditions since the WCAP-9620-R1 data was no longer deemed applicable for the reactor internals design calculations of these components. Long-term heat generation rates intended to represent time-averaged behavior are used in component fatigue analyses, whereas the short-term results are intended to provide conservative values for use in calculating

maximum temperatures and thermal stresses of components. For the long-term heat generation rate evaluation of the core plates, a reactor power level of 3950 MWt was used in conjunction with a flat axial core power distribution, since these parameters significantly influence the core plate gamma heating rates and the aforementioned conditions conservatively bound the IP2 SPU Program. (Note: The reactor power level of 3950 MWt was selected since this currently bounds the entire fleet of Westinghouse four-loop plants.) For the short-term heat generation rate evaluation of the upper core plate, the reactor power of 3950 MWt was assumed and a conservative design-basis top-peaked axial power distribution from WCAP-9620 (Reference 1) was used. Analogous conditions were applied in the short-term heating rate evaluation of the lower core plate; however, in this case, the design basis bottom-peaked axial power distribution from Reference 1 was employed for conservatism.

For the radial reactor internals components, only a long-term analysis was performed, since it was anticipated that the current IP2 gamma heating rates would be bounded by the corresponding data reported in WCAP-9620. (This scenario was hypothesized since IP2 has transitioned to low-leakage loading patterns, whereas an out-in loading pattern was assumed in WCAP-9620 (Reference 1). Hence, the long-term case was examined to provide confirmation that the WCAP results remained conservative for the radial components.) Since the long-term radial case of WCAP-9620 was shown to be bounding, the short-term radial case of WCAP-9620 would also remain bounding and, therefore, was not calculated. The long-term heat generation rate evaluation of the core baffle plates, core barrel, and thermal shield was based on the Cycle 19 radial power distribution (which included a 1.05 bias of the peripheral fuel assemblies) forecasted for use by IP2 operating at the reactor power level of 3216 MWt, as reported in Table 2.1-2.

Design basis heat generation rates applicable to the IP2 radial internals were obtained from Appendix J of Reference 1. The core power distributions upon which those calculations were based were derived from statistical studies of 23 independent fuel cycles from 10 four-loop reactors. These power distributions represented an upper tolerance limit for beginning-of-cycle (BOC) and end-of-cycle (EOC) power in the peripheral fuel assemblies, based on a 95-percent probability with a 95-percent confidence level. Most of the evaluated fuel cycles were based on an out-in fuel loading strategy (fresh fuel on the periphery) which, when combined with the statistical processing of the data, resulted in a design basis core power distribution that tended to be biased high on the periphery. This high bias on the core periphery was desired by the reactor internals analysts to ensure conservative, but realistic, design calculations for the critical baffle-barrel region of the reactor internals and explains why the WCAP-9620 radial component heating rate results were expected to bound the corresponding IP2 values.

7.6.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to the reactor internals evaluation that is described in Section 5.2 of this report.

7.6.4 Description of Analysis/Evaluation and Results

The heat generation rate analyses were carried out using the DORT (DOORS 3.1 code package [Reference 2]) two-dimensional (2-D) discrete ordinates transport code in the forward mode and the BUGLE-96 cross-section library (Reference 3). This suite of codes has been used to support numerous pressure vessel fluence evaluations and are generally accepted by the Nuclear Regulatory Commission (NRC) for deterministic particle transport calculations, for example, neutron exposure and gamma-ray heating rate evaluations.

Two different coordinate systems were used in the 2-D heating rate analyses to precisely model the components undergoing evaluation. The core baffle plates were analyzed using a x,y coordinate system, and the core barrel and thermal shield heating rates were determined using a r,θ geometric model.

The results of the radiation-induced heat generation rate calculations were provided as inputs for the reactor internals evaluations described in Section 5.2. The volume-averaged heat generation rates for the core plates and radial reactor internal components that were evaluated as part of this study are summarized in Table 7.6-1. In accordance with WCAP-9620 (Reference 1), this table also segregates the core plate heating rates into two distinct regions. Region A refers to the cylindrical portion of the core plates that are axially adjacent to the active fuel, and Region B refers to the annular portion of the plates that are located radially outboard of the active fuel.

As expected, the revised IP2 zone average gamma heating rates for the core plates tended to be much higher than the corresponding WCAP-9620 (Reference 1) data. As a result, the spatial distributions of long-term and short-term heating rates for the upper and lower core plates that are presented in Tables 7.6-2 through 7.6-5 were also identified for consideration as part of the component evaluation that is described in Section 5.2 of this report.

Table 7.6-1 also shows that the current IP2 zone average gamma heating rates for the core baffle, core barrel, and thermal shield continue to remain bounded by the conservative radial component heating rates that are reported in WCAP-9620 (Reference 1).

7.6.5 Conclusions

The component gamma heating rates that are presented in Tables 7.6-1 through 7.6-5 were provided as inputs to the reactor internals evaluation described in Section 5.2 of this document.

7.6.6 References

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

2. RSICC Computer Code Collection CCC-650, DOORS 3.1, *One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System*, August 1996.

3. RSIC Data Library Collection DLC-185, *BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications*, March 1996.

Table 7.6-1 Reactor Internals Zone Average Gamma Heating Rates		
Location	Region Average Long-Term Heating Rates (Btu/hr-lbm)	
	WCAP-9620-R1 Analysis* (Ref. 1, Appendix J)	New IP2 Analysis
Baffle Plate 18	784	443
Baffle Plate 19	885	554
Baffle Plate 20	821	464
Baffle Plate 21	645	341
Core Barrel	158	83
Thermal Shield	22	12
* Values are scaled down by a factor of 3216/3565 to account for difference in reactor power.		
	Upper and Lower Core Plates Heating Rates (Btu/hr-lbm)	
	WCAP-9620-R1 Analysis (Ref. 1, Appendix E&J)	New Baseline Analysis
Long Term Heating Rates		
Upper Core Plate A	27.4	246
Upper Core Plate B	5.57	29
Lower Core Plate A	249	903
Lower Core Plate B	52.4	88
Short Term Heating Rates		
Upper Core Plate A	64.4	265
Upper Core Plate B	15.0	34
Lower Core Plate A	822	1480
Lower Core Plate B	201	167

Table 7.6-2

Spatial Distribution of Long-Term Gamma Heating Rates (Btu/hr-lbm) in the Upper Core Plate for IP2

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	Top Surface 2.00
0.98	472	426	335	269	219	194
2.95	471	425	334	268	218	194
4.92	470	425	333	267	217	193
6.89	469	423	332	266	217	192
8.86	467	422	331	265	216	192
10.83	466	420	330	264	215	191
12.80	464	419	329	263	215	190
14.76	463	418	328	262	214	190
16.73	462	417	327	262	213	189
18.70	461	416	326	261	213	189
20.67	460	415	325	261	213	189
22.64	459	415	325	260	213	189
24.61	459	415	325	261	213	189
26.57	459	415	325	261	213	189
28.54	459	415	326	261	213	189
30.51	459	415	326	261	213	189
32.48	459	415	325	261	213	189
34.45	458	414	325	260	212	188
36.42	457	412	324	259	211	188
38.39	454	410	322	258	210	186
40.35	449	406	319	255	208	184
42.32	443	400	314	252	205	182
44.29	435	393	309	247	201	178
46.26	424	383	301	241	196	174
48.23	409	369	290	232	189	167
50.20	390	352	277	221	180	160
52.17	366	331	260	208	169	150
54.13	338	306	240	192	156	139
56.10	307	277	218	174	142	126
58.07	273	247	194	155	127	112
60.04	239	216	169	136	110	98
62.01	204	184	144	116	94	83
63.78	172	155	122	97	79	70
64.96	150	135	106	84	69	61
65.65	134	121	95	75	61	54
66.15	121	109	86	68	56	49
66.64	90	82	66	53	44	39
67.20	63	58	48	40	33	30
67.89	54	48	37	30	25	23
68.70	53	46	31	24	20	18
69.52	52	45	29	21	17	15
70.33	50	43	27	19	15	13
71.15	47	40	25	17	13	11
71.96	44	37	23	16	12	10
72.78	39	33	21	14	11	9
73.59	35	29	18	12	9	8
74.00	32	27	17	11	8	7

Table 7.6-3

Spatial Distribution of Short-Term Gamma Heating Rates (Btu/hr-lbm) in the Upper Core Plate for IP2

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	Top Surface 2.00
0.98	517	467	367	295	241	213
2.95	517	466	366	293	240	213
4.92	516	466	365	292	239	212
6.89	514	464	364	291	238	211
8.86	513	463	363	290	237	211
10.83	512	462	362	290	237	210
12.80	510	460	361	289	236	209
14.76	509	459	360	288	235	209
16.73	507	458	359	287	235	208
18.70	506	457	358	287	234	208
20.67	505	456	357	286	234	208
22.64	504	455	357	286	234	208
24.61	504	455	357	286	234	208
26.57	504	455	357	286	234	208
28.54	504	455	357	286	234	208
30.51	504	455	357	286	234	208
32.48	503	454	357	286	233	207
34.45	502	453	356	285	233	207
36.42	500	451	354	284	232	206
38.39	497	448	352	282	230	204
40.35	492	444	348	279	228	202
42.32	485	438	344	275	225	199
44.29	475	429	337	270	220	195
46.26	463	418	328	263	214	190
48.23	446	403	316	253	206	183
50.20	425	384	301	241	197	174
52.17	399	360	283	226	185	164
54.13	369	333	261	209	171	151
56.10	335	302	237	190	155	138
58.07	298	269	212	170	138	123
60.04	261	235	185	148	121	107
62.01	223	201	158	126	103	91
63.78	188	169	133	106	87	77
64.96	163	147	116	92	75	67
65.65	146	131	103	82	67	59
66.15	131	118	93	75	61	54
66.64	99	90	73	59	49	43
67.20	70	65	54	44	37	34
67.89	62	55	42	34	29	26
68.70	61	53	36	28	23	21
69.52	61	52	34	24	20	17
70.33	59	50	32	23	18	15
71.15	56	48	30	21	16	14
71.96	52	44	28	19	15	12
72.78	47	40	25	17	13	11
73.59	42	35	22	15	11	10
74.00	39	33	21	14	11	9

Table 7.6-4

Spatial Distribution of Long-Term Gamma Heating Rates (Btu/hr-lbm) in the Lower Core Plate for IP2

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	Top Surface 2.00
0.98	694	782	958	1196	1518	1679
2.95	693	780	956	1196	1522	1684
4.92	693	781	956	1197	1524	1687
6.89	690	778	953	1193	1519	1683
8.86	686	773	946	1185	1507	1668
10.83	680	766	939	1174	1493	1652
12.80	676	761	932	1165	1482	1641
14.76	672	757	927	1159	1474	1631
16.73	670	755	924	1156	1470	1628
18.70	669	753	922	1153	1467	1624
20.67	667	751	919	1150	1463	1619
22.64	665	749	916	1146	1458	1613
24.61	665	748	915	1144	1455	1611
26.57	667	750	918	1148	1460	1616
28.54	670	755	924	1157	1471	1628
30.51	675	760	932	1166	1484	1642
32.48	677	764	936	1173	1492	1651
34.45	678	765	937	1174	1493	1653
36.42	678	764	936	1172	1491	1650
38.39	677	763	935	1171	1490	1649
40.35	678	764	937	1172	1492	1652
42.32	679	766	941	1178	1500	1660
44.29	681	769	945	1185	1508	1670
46.26	679	768	946	1187	1511	1674
48.23	670	759	937	1177	1499	1660
50.20	650	737	912	1146	1460	1617
52.17	616	700	866	1090	1388	1537
54.13	567	644	798	1004	1279	1417
56.10	505	573	708	890	1134	1256
58.07	434	491	604	758	965	1068
60.04	359	405	496	621	788	872
62.01	286	321	391	488	618	683
63.78	224	251	304	377	476	525
64.96	186	207	249	308	386	425
65.65	163	180	216	266	331	363
66.15	144	160	191	236	292	320
66.64	120	133	158	197	253	280
67.20	98	107	127	161	216	244
67.89	79	87	103	134	185	211
68.70	64	71	84	111	157	180
69.52	54	59	70	93	135	156
70.33	45	49	58	79	117	136
71.15	38	42	49	67	102	120
71.96	32	35	42	58	89	105
72.78	26	29	35	49	76	90
73.59	22	24	28	40	64	76
74.00	19	21	25	36	58	69

Table 7.6-5

Spatial Distribution of Short-Term Gamma Heating Rates (Btu/hr-lbm) in the Lower Core Plate for IP2

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	Top Surface 2.00
0.98	1178	1313	1584	1956	2457	2708
2.95	1174	1310	1581	1957	2464	2717
4.92	1175	1310	1581	1958	2465	2719
6.89	1171	1306	1576	1951	2458	2711
8.86	1163	1297	1565	1938	2440	2690
10.83	1154	1287	1553	1922	2418	2666
12.80	1148	1279	1542	1908	2401	2648
14.76	1142	1273	1534	1898	2388	2633
16.73	1138	1268	1530	1893	2382	2627
18.70	1135	1265	1526	1888	2376	2620
20.67	1132	1262	1522	1883	2370	2613
22.64	1130	1259	1517	1877	2362	2605
24.61	1129	1258	1516	1875	2360	2602
26.57	1133	1262	1521	1881	2367	2610
28.54	1138	1269	1531	1894	2383	2628
30.51	1145	1277	1541	1908	2402	2649
32.48	1149	1282	1549	1918	2414	2662
34.45	1150	1284	1550	1920	2417	2665
36.42	1150	1283	1549	1918	2415	2663
38.39	1149	1282	1548	1917	2414	2662
40.35	1149	1283	1550	1919	2417	2666
42.32	1151	1286	1555	1927	2428	2678
44.29	1151	1288	1561	1935	2439	2690
46.26	1145	1283	1559	1935	2439	2691
48.23	1128	1265	1541	1915	2414	2664
50.20	1092	1227	1497	1863	2348	2591
52.17	1035	1163	1421	1769	2230	2461
54.13	953	1071	1308	1629	2054	2267
56.10	850	954	1162	1446	1824	2012
58.07	732	820	995	1235	1557	1717
60.04	609	680	822	1016	1277	1408
62.01	488	543	652	805	1009	1111
63.78	387	428	511	627	783	861
64.96	324	357	423	517	641	703
65.65	285	313	370	450	554	606
66.15	253	279	330	403	493	539
66.64	214	235	277	342	432	478
67.20	177	193	226	285	377	423
67.89	146	160	188	243	332	376
68.70	122	134	158	208	290	331
69.52	104	114	136	181	257	295
70.33	89	98	117	158	229	265
71.15	76	84	101	138	204	236
71.96	64	72	87	119	179	209
72.78	54	61	73	102	155	181
73.59	45	50	60	83	129	152
74.00	40	44	53	74	116	138

8.0 TURBINE ISLAND ANALYSIS

8.1 Steam Turbine

The currently installed Indian Point Unit 2 (IP2) steam turbine is a Westinghouse nuclear turbo set. The steam turbine is composed of 4 elements—1 double flow partial-arc steam admission high-pressure (HP) turbine BB96 and 3 double-flow, low-pressure (LP) turbines BB81R.

In order to optimize the HP efficiency and to increase the swallowing capacity at the Stretch Power Uprate (SPU) thermal power of 3228.5 MW, the rotor, blading and the inner casing of the HP turbine will be exchanged. The existing LP turbine rotors, turbine valves, and turbine auxiliary systems were found to be acceptable for the full-power uprate pressure, temperature, and flow conditions.

The HP turbine will be replaced by the full-arc steam admission turbine. This all-reaction turbine is designed to provide 2-percent nominal flow margin at the full-uprate power level throttle valve steam conditions. This design also provides improved full-load performance by eliminating the partial admission control stage and applying current blade path technology.

The major changes associated with the new HP turbines are:

- Elimination of the inlet nozzle blocks that will be replaced with full-arc admission and a new inner casing including a diagonal stage.
- Optimized all-reaction blading
- Improved materials for blade rings (stainless steel)
- Monoblock HP rotor with no through-bore
- Full-arc steam admission at all loads

The existing turbine bearings, gland seals, main lube oil system, hydraulic control system, and gland sealing steam system are acceptable for the uprated conditions. The main steam inlet piping, cross-over piping, cross-under piping, stop valves, throttle valves, and control valves are acceptable for the uprated conditions. The HP turbine first-stage instrumentation will be adjusted to the new pressure conditions for the reaction turbine.

The existing LP turbine missile analysis and the generic turbine missile analysis for the original and retrofit HP rotor were also reviewed for the SPU conditions. The uprate steam conditions

were found to have no effect on the validity of the existing turbine missile analyses. Therefore, the existing analysis remains valid.

Based on the above, it is concluded that the new all-reaction HP turbines and the LP turbines are acceptable for operation at the SPU conditions.

8.2 Heat Balances

Heat balances were generated to identify relevant parameters and design inputs to evaluate balance-of-plant systems, structures and auxiliaries at the SPU conditions. Detailed heat balance models were developed and tuned to match plant operational data and extrapolated to SPU conditions. In addition to the guarantee heat balance at full-load conditions at rated condenser pressure, heat balances were also generated for partial-load conditions and for different condenser pressure levels.

These heat balances were used in the BOP evaluations as indicated in Section 9 of this report.

9.0 BOP SYSTEMS

Introduction

To predict the performance of the balance-of-plant (BOP) thermal cycle at the stretch power uprate (SPU) conditions and to determine the corresponding system/equipment operating parameters, heat balances were developed using PEPSE models.

The SPU heat balances define the bounding parameters for evaluating the BOP System performance at the SPU uprate condition.

Method of SPU Heat Balance Development

To accurately predict BOP System performance during SPU operation, it was first necessary to develop a benchmark heat balance model that represented the current plant performance. This benchmark heat balance model was then used as a basis for developing a variety of SPU cases.

Development of the baseline and uprate models was accomplished as follows:

- The existing PEPSE heat balance model that was based on "as-designed" component parameters was reviewed. Physical data in the model were verified as being representative of the current plant design by a detailed review of plant design documents and physical inspection results.

- Actual operating temperatures, pressures, and flows with the plant operating at 100-percent power were obtained. Using these data, the PEPSE model was "tuned" to represent the actual performance characteristics of the plant thermal cycle, including the effects of component degradation or modifications that may change their performance from the as-designed characteristics. This tuned heat balance was then established as the "benchmark heat balance."

- The current operation (benchmark) heat balance was then modified to reflect the performance of the new high-pressure (HP) turbine being installed, as well as any other plant design changes planned prior to SPU. This modified heat balance was then established as the SPU heat balance.

- The SPU was then run for a range of condenser backpressures (that is, circulating water temperature variations), and a margin of 0.5 percent was added to each case to provide conservatism. These heat balance cases, including the 0.5 percent margin, were then used as the basis for evaluating the BOP Systems and components.

9.1 Main Steam System

9.1.1 Introduction

Main Steam System (MSS) piping components and equipment, including specifically the main steam safety valves (MSSVs), atmospheric relief valves (ARVs), main steam isolation valves (MSIVs), and condenser steam dump valves, were evaluated for the Indian Point Unit 2 (IP2) stretch power uprate (SPU) conditions.

The MSS transports saturated steam produced in the steam generators to the main turbine for power generation. The steam dump and bypass piping and valves provide alternate flow paths for the generated steam when the turbine is unavailable, or when a plant operational transient requires a reduction in the main turbine power level.

In addition to supplying saturated steam to the main turbine, the MSS also supplies steam to the following users:

- Main boiler feed pump drive turbines
- Moisture separator reheaters (MSRs)
- High-pressure (HP) turbine cylinders
- HP turbine gland sealing steam system
- Priming and steam jet air ejectors
- Auxiliary feedwater (AFW) pump drive turbine
- Auxiliary steam system (via reducing valve)

An inadvertent opening, with failure to close, of the largest of any single steam dump, relief, or safety valve will not prevent the safe shutdown of the plant. The maximum capacity of any single MSSV, ARV, or main steam dump valve does not exceed 890,000 lb/hr at 1085 psig inlet pressure. This feature limits the potential uncontrolled blowdown flow rate in the event a valve inadvertently fails or sticks in the open position. This maximum value has not changed for SPU.

9.1.2 Input Parameters and Assumptions

The evaluation of the MSS used conditions predicted by SPU thermal cycle heat balances. The SPU heat balances were developed by first establishing a benchmark heat balance model representative of current plant performance, which was then used as a basis for developing heat balances representative of SPU operation. The SPU heat balance parameters were used as the bounding values for evaluating the MSS.

9.1.3 Description of Analysis and Evaluations

The MSS piping, valves, and components were evaluated to verify their ability to operate at SPU conditions. Based on SPU heat balances, operation at the higher SPU power level increases the steam flow required from the steam generators to the HP turbine. Additional steam flow is also necessary for other components, which operate at higher loads and use steam as a motive force.

Specifically, the SPU heat balance parameters were reviewed and compared with original system heat balance parameters as well as the original component design parameters to determine their capability to function adequately at SPU conditions.

The following specific system design features were reviewed and evaluated:

- Main steam pressure drop and flow versus required HP turbine inlet conditions
- Main steam piping pressure/temperature design and flow velocities
- Main steam component pressure/temperature design
- Design closure time for MSIVs
- Setpoints for ARVs
- Setpoints for MSSVs
- Steam supply flow rates/line sizes/velocities to auxiliary components

9.1.4 Acceptance Criteria

Generally, the evaluations must prove that the existing design parameters associated with the MSS piping and components bound the conditions under SPU operation. Specifically, the following criteria must be met:

- Steam pressure and flow must satisfy HP turbine throttle inlet conditions required by the SPU heat balances.

- Main steam piping and component pressure/temperature design must exceed the maximum expected operating pressure and temperature associated with SPU and abnormal and accident conditions.

- MSIVs must be able to close within the required times under power uprate conditions and abnormal and accident conditions.

- Increases in main steam piping velocities due to the power uprate should remain within accepted industry standards for the service conditions and existing pipe material. The expected velocities at SPU flow rates, when considered with the SPU operating temperatures, should not appreciably increase the potential for flow-accelerated corrosion (FAC).

- The MSSV setpoints must consider the added piping pressure drop due to increased SPU flows and must be adequate to ensure that the steam generator pressure does not exceed 110 percent of design pressure.

- Sufficient steam flow and pressure must be provided to auxiliary components using main steam to meet SPU operating requirements for each component.

9.1.5 Results and Conclusions

Based on the system evaluation discussed in the previous sections, it was determined that the IP2 MSS is capable of performing its design function under SPU conditions. The following sections provide a discussion of specific evaluation results/conclusions.

9.1.5.1 Flow Restriction Nozzles

The main steam (MS) header at the outlet of each steam generator contains a venture-type steam flow restriction nozzle. These flow restriction nozzles are designed to limit the blowdown flow from a downstream rupture in the main steam header, and to provide flow measurement of each steam header via differential pressure connections upstream and downstream. As described in Sections 4 and 5 of this report, the flow restriction nozzles are acceptable for use under SPU conditions.

9.1.5.2 Main Steam Safety Valves (MSSVs)

Each of the 4 MS headers, at a location outside of reactor containment, contains 5 MSSVs, which provide overpressure protection for the steam generators and the MSS inside containment. The safety valves are designed to pass a total of 100 percent of MS flow rate while maintaining the steam generators at or below 110 percent of design pressure. Maximum steam flow rate at 100 percent power under SPU conditions is significantly below the MSSV design capacity. As described in Sections 4 and 5 of this report, the IP2 MSSVs are acceptable for overpressure protection under SPU conditions.

Based on the aggregate capacity of the safety valves, the safety valve setpoints were evaluated to confirm that the existing setpoints do not result in a steam generator pressure greater than

110 percent of the design pressure of 1085 psig. There is no change to the steam generator design pressure due to the SPU. The evaluation determined that the steam generator pressure was well below the 110-percent limit when the existing safety valves were passing the required relieving capacity at SPU conditions.

MSSV setpoints are acceptable for operation under SPU conditions and will maintain the steam generators below their design pressure.

9.1.5.3 Atmospheric Steam Relief Valves

The MSS includes 4 ARVs. These relief valves are used for controlling Reactor Coolant System (RCS) temperature to maintain hot standby and to cool the RCS prior to initiating residual heat removal (RHR).

To limit the frequency of safety valve lifts, the setpoints of the ARVs are based on plant no-load conditions and the lowest MSSV setpoint. These 4 valves are designed to pass a total of 10 percent of full-load MS mass flow rate at no-load steam generator outlet pressure. As discussed in subsection 4.2.1 of this document, the IP2 ARVs are adequate to support required steam relief (during a steam generator tube rupture [SGTR] and other cooldown events) under SPU conditions.

Since the no-load steam generator pressure and the lowest set MSSV setpoint are not changed with the implementation of SPU, current setpoints of the ARVs are acceptable and will not change.

9.1.5.4 HP Steam Dump Valves

The HP steam dump valves and associated piping are designed to reduce the transients on the RCS during plant trips and load rejections. Twelve HP steam dump valves, 6 on each MS auxiliary loads header, are provided to discharge MS directly to the main condenser. The valves are required to pass a total of 40 percent of full-load MS mass flow.

The full-load MS flow increases under SPU conditions. As detailed in subsection 4.2.1 of this report, the IP2 HP steam dump valves are adequate for operation under SPU conditions.

9.1.5.5 Low-Pressure Steam Dump Valves

The low-pressure (LP) steam dump valves and associated piping are designed to preclude LP turbine overspeed by diverting a portion of the HP turbine exhaust steam from the crossunder lines directly to the main condensers. Six 10-inch diameter dump valves and piping are

provided, each of which branches from the crossunder line near the MSR to the condenser. Each dump line contains a motor-operated isolation valve and an air-operated dump valve in series.

The LP steam dump valves are required to pass a total of approximately 25 percent of the MS available to prevent overspeed of the turbine following a turbine trip. The full load main steam flow increases under SPU conditions. As detailed in subsection 4.2.1, the IP2 LP steam dump valves are adequate for operation under SPU conditions.

9.1.5.6 MSIVs and Non-Return Valves

The IP2 MSIVs and non-return check valves are located outside of containment (downstream of the MSSVs) and function to prevent uncontrolled blowdown of more than 1 steam generator. The valves are swing-disc check valves. The isolation valves are reverse-mounted on the MS headers, utilizing a spring-loaded air piston to hold the disc out of the steam flow.

Since the steam generator design pressure and the MSSV setpoints are not changing due to SPU operation, the MSIV and non-return valve design pressure and temperature are not affected.

The MSIVs and non-return valves are required to have the capability of closing in 5 seconds or less in the event of a MS line rupture. Because the MSIVs isolation valves and non-return valves are a check valve design, reverse steam flow will assist in closing these valves. Therefore under SPU conditions of increased flow, the valves will continue to meet their design capability, including the capability of closing in 5 seconds or less. The MSIVs and non-return valves are acceptable for SPU operation without modification. Consideration of piping and support loads relating to rapid valve closure is addressed in Section 9.9.

9.1.5.7 AFW Pump Drive Turbine Steam Supply

In the event of an abnormal condition and accident, the MSS must supply motive steam to the AFW pump drive turbine. The AFW pump can operate using MS over the entire range of MS pressures from normal operation to very low pressures at startup or shutdown. Based on the system evaluation, the AFW pump water flow requirement to the steam generators is not affected by SPU operation.

The MS supply line to the turbine drive for the AFW pump is designed to provide steam at a range of pressures from 110 to 1118 psig. The turbine drive is designed to operate at a maximum inlet pressure of 600 psig. A pressure control valve on the steam supply line reduces the supply pressure to 600 psig or less when MS pressure is higher. Based on the evaluation

under SPU conditions at full load, the pressure of the MS supply upstream of the control valve was 750 psig, thus providing sufficient pressure.

9.1.5.8 Main Feedwater Pump Drive Turbine Steam Supply

The MSS supplies motive steam to the main feedwater pump turbine drives (Nos. 21 and 22), during all modes of pump operation. Initially, during plant startup, steam is provided directly from the main steam headers. When sufficient pressure exists in the hot reheat side of the MSR, steam is provided from the "A" MSRs.

Under current conditions, with the plant operating at 20-percent-rated MWt, the MSS supplies 20,465 lb/hr of HP steam to the each of the 2 feedwater pump turbines. Under SPU conditions (that is, higher power level), additional steam is generated. Therefore, the MSS will continue to supply the required 20,465 lb/hr of steam during startup.

Since the full-load main feedwater flow requirements increase relative to SPU operation, the required steam flow for the 2 feedwater pump turbines also increases. A comparison of the required steam flow to the turbine drives during SPU operation with SPU heat balances confirmed adequate steam flow capacity available under SPU operation to provide the required steam flow to the turbine drives (that is, the steam flows available to the main feedwater pump turbines is greater than the steam flow required).

9.1.5.9 Main Steam Piping

Under SPU operating conditions, the steam generator steam outlet mass flow rate will increase approximately 6 percent above the current operating mass flow rate. This increase will impact MS header piping pressure drops and flow velocities.

The MS piping pressure and temperature design bounds power uprate pressure temperature conditions. Piping pressure temperature design is, therefore, acceptable for SPU conditions.

The MS header piping pressure drop at SPU conditions from the steam generators to the HP turbine throttle valve inlet was calculated and compared with original design pressure drop parameters. There was adequate steam flow and pressure to satisfy throttle valve inlet requirements under SPU conditions.

Increased MS piping flow velocities based on SPU conditions in main steam piping to normally operating components were evaluated and found acceptable. Velocities in pipelines to the AFW pump turbines and startup supply line for the main feedwater pump turbines are also acceptable. Existing FAC monitoring activities will ensure that corrosion remains acceptable.

9.2 Extraction Steam System

9.2.1 Introduction

The IP2 Extraction Steam (ES) System was evaluated in conjunction with stretch power uprate (SPU) conditions to determine the extent to which system design parameters bound SPU conditions.

The IP2 ES System is designed to transmit steam from HP and LP main turbines to the shell sides of the feedwater heaters to heat feedwater to improve cycle efficiency.

The ES System has no safety function.

9.2.2 Input Parameters and Assumptions

The Main Steam System (MSS) was evaluated using conditions predicted by SPU thermal cycle heat balances. The SPU heat balances were developed by first establishing a benchmark heat balance model representative of current plant performance, which was then used as a basis for developing heat balances representative of SPU operation. The SPU heat balance parameters were used as the bounding values for evaluating the ES System.

9.2.3 Description of Analysis and Evaluations

The ES System was evaluated to verify its ability to operate at the SPU conditions. SPU heat balances were used to establish the SPU parameters with which the turbine cycle system evaluations were performed. A tuned baseline heat balance was also used in these evaluations.

The following ES System design features were reviewed and evaluated:

- The pressure and temperature design of extraction steam piping and valves was compared with SPU pressure and temperature conditions.

- The FW heater shell pressure and temperature design was compared with SPU pressure and temperature conditions.

- Extraction steam piping velocities at the higher flow rates of SPU operating conditions were compared to industry standard criteria for extraction steam service. These velocities were also evaluated to determine whether the SPU flow rates increase the possibility of flow-accelerated corrosion (FAC).

- Feedwater heater (FWH) extraction steam inlet nozzle velocities at SPU operating conditions were compared with standard industry guidelines (Heat Exchange Institute [HEI]) to size FWH nozzles to determine the potential for increased wear and FAC.

- The results of past FWH inspections were reviewed to determine the current physical condition of the heaters.

- The SPU extraction steam flow rates into the FWHs were evaluated to determine the effects on tube vibration and erosion of internal subcomponents and support structures.

- Extraction steam piping flow regimes were evaluated relative to moisture carryover (MCO) capability.

9.2.4 Acceptance Criteria

Generally, the evaluations must demonstrate that design parameters of the existing ES System piping and valves bound the corresponding parameters at SPU conditions. Specifically, the following criteria must be met:

- Extraction steam piping and valves pressure and temperature design should envelope the pressure and temperature conditions expected under SPU operation.

- FWH shell pressure and temperature design should envelope the pressure and temperature conditions expected under SPU operation.

- FWH extraction steam inlet nozzle velocities at SPU conditions should not appreciably increase the potential for wear and FAC.

- Extraction steam piping flow velocities due to SPU are within the industry standard values for extraction steam piping of this size, material, and service. The expected velocities at SPU flow rates, when considered with the SPU operating temperatures, should not appreciably increase the potential for FAC.

- The SPU extraction steam flow rates into the FWHs should not cause destructive tube vibration or the erosion of internal parts such that their function is impaired.

- Relative to extraction steam line flow regimes, system piping flow must exhibit effective MCO and should not exhibit slug flow characteristics.

9.2.5 Results and Conclusions

Based on the system evaluation as discussed in the above sections, it was determined that the IP2 ES System is capable of performing its design function under SPU conditions. The following sections provide discussion of specific evaluation results and conclusions.

9.2.5.1 Extraction Steam Piping and Components

ES System pressure/temperature conditions predicted under SPU are bounded by system component and piping design parameters.

Calculated pipeline velocities under SPU conditions are bounded by industry standard velocity limits and are acceptable for SPU operation. FAC associated with these lines under SPU conditions will not significantly increase. FAC Program activities for the extraction lines should be continued during SPU operation.

9.2.5.2 FWHs

Design Pressure and Temperature

With the exception of IP2 FWHs 24A, B, and C and 26A, B, and C, the FWH shells pressure and temperature design envelops the SPU pressure/temperature conditions. For FWH 24A, B, and C, the maximum shell-side inlet temperature during SPU exceeds the shell design temperature by 10°F. For FWH 26A, B, and C, the SPU temperature is 28.5°F above design. The shell materials of these heaters is carbon steel SA 516 Grade 70. Therefore, the shell design can accept the higher SPU temperatures since the maximum allowable stress value of material SA 516 Grade 70 in tension does not change in the temperature range of –20 to 650°F. The manufacturer of these heaters has been contracted to change the supporting stress analysis.

Extraction Steam FWH Nozzle Velocities

With the exception of IP2 FWHs 21A, B, and C and 22A, B, and C, extraction steam inlet nozzle velocities are bound by the HEI standard industry guidelines for FWH nozzles. The nozzles of FWHs 21A, B, and C and 22A, B, and C will be included in the plant FAC Program to monitor future wear.

Extraction Steam Flow – Effect on FWH Internals

The SPU extraction steam flows are below the existing design values for FWHs 23A, B, and C; 24A, B, and C; and 25A, B, and C. For FWHs 21A, B, and C; 22A, B, and C; and 26A, B, and C, the SPU flows are above the design values. These heaters will be maintained and inspected to ensure that excessive tube vibration or significant erosion of internal parts will not occur at the higher SPU flow rates.

9.2.6 Flow Regimes

Horizontal portions of the ES System piping are expected to develop either a semi-annular pattern, or to contain a liquid phase portion that is small enough to be carried over. Vertical upward flows are expected to develop annular or mist flow patterns so that effective MCO will occur. Void coefficients associated with the vertical downward flowing portions of the system exceed minimum acceptance criteria with enough margin that slug patterns are not expected.

9.3 Heater Drains System

9.3.1 Introduction

The Heater Drains System was evaluated in conjunction with stretch power uprate (SPU) conditions to determine the extent to which system design parameters bound SPU conditions.

The turbine cycle has 6 stages of feedwater heaters (FWHs). Each stage consists of 3 strings of heaters.

The drains from the heaters 25 and 26 are collected in the heater drain tank and then pumped by 2 half-size heater drain pumps to the suction of the main feedwater pumps. The drains from heaters 24, 23, 22, and 21 flow cascade from higher pressure to lower pressure heaters. The combined drains in heaters 21 flow to the condenser. Bypass drain lines to the condenser are also provided for each heater and the heater drain tank to dump drains directly to the condenser on high level.

As part of the plant's turbine water induction prevention features for events such as a heater tube rupture, a second emergency drain is required for condenser neck heaters 22 and 21 since a non-return valve cannot be provided in the extraction steam lines. On high-high level in these heaters, the emergency lines will open to drain additional flow to the condenser. Simultaneously, the level control valves on the bypass drain line to condenser will remain open and the cascading drain flow from the preceding heaters will be isolated.

Moisture Separator, Reheater, and Moisture Pre-Separator Drain Systems

Each moisture separator drains to its associated moisture separator drain tank. The moisture separator drain tanks flow to the heater drain tank during normal plant operation and to the drain collecting tank during startup, shutdown, or high water level conditions. The drain collecting tank drains to the condenser.

Each reheater drains to its associated reheater drain tank. The reheater drain tanks flow to heater 26. In the abnormal situation of high water level, the reheater drains are diverted to the condensers.

The moisture pre-separator drains are collected in separating tanks, which, in turn, drain to the heater drain tank.

The moisture pre-separator, separator, and reheater drains are returned to the thermal cycle by pumping the heater drain tank into the suction of the feedwater pumps.

Normal Operating Vent Lines of Heaters to Condensers

The normally operating vent lines of heaters are directed to the condenser through piping provided with globe valves for isolation or throttling of the flow.

Scavenging Steam to Reheaters

The additional heating steam supplied to the reheater, called scavenging steam, ensures that all reheater tubes are flowing clearly and a vapor space exists over condensed steam. This scavenging steam is directed to FWHs 26 during normal operation and to the condensers during start up.

Heaters Relief Valves

All heaters, with the exception of condenser neck heaters 21 and 22, are equipped with shell-side relief valves for overpressure protection of heater shells in the event of rupture of heater tube. These heaters, 21 and 22, have no isolation valves in their extraction lines from the low-pressure (LP) turbine and, therefore, have no relief valves for the shell.

9.3.2 Input Parameters and Assumptions

A current operating (benchmark) heat balance, tuned to the current plant operating characteristics and SPU heat balances at 1.0-, 1.5-, and 3.0-inch HgA condenser pressures, were used in the evaluation of the system. Additionally, these heat balances included a margin of 0.5 percent as a conservatism for evaluation purposes. Each of these heat balances and the corresponding parameters were reviewed and the most conservative case was chosen for the specific evaluation being performed.

Plant design basis documents, system descriptions, equipment and piping specification, drawings, and calculations provided system and component design parameters.

9.3.3 Description of Analysis and Evaluation

Hydraulic Analysis of Operation of Heater Drain Pumps and Associated Suction and Discharge Piping System

Refer to subsection 9.4 of this report for the analysis of heater drain pumps and associated suction and discharge piping at SPU conditions.

FWHs and Drain Tanks Level Control Valves

The change in the flow coefficient (% C_V difference) of the level control valves (LCVs) at the current operating and SPU conditions has been determined based on the heat balance parameters.

A generic flow characteristic curve has been used to determine the expected change of valve position at SPU conditions based on current valve position and % C_V difference. The expected change of valve position at SPU conditions has been added to the current opening position to predict the opening position of the normally operated drain valves after SPU.

If the opening position of LCVs exceeds 75 percent at SPU conditions or the change in valve opening position at current and SPU conditions is significantly different, a detailed pressure drop analysis has been performed to determine the change of valve position from current and SPU conditions.

The LCVs on bypass lines from heaters to the condenser are closed during normal operation and are opened when the normal drain line is not in service. The design of these bypass lines and valves are the same as the normally operated level control valve for the subject heater. These bypass lines and valves are adequately sized for SPU conditions.

The required C_V at SPU conditions is expressed in a percentage of maximum C_V as 100 x [$C_{V\ UPRATE}/C_{V\ MAX}$]. The expected opening positions of heater drain tank bypass and condenser neck heaters emergency dump to condenser LCVs at SPU conditions have been determined from the generic flow characteristic curve.

FWH, Moisture Separator, and Reheater Gravity Drain Lines

The capability of gravity drain lines has been analyzed by using the Froude Number criteria, $F_N < 0.3$, for self-venting flow. The gravity drain lines include the following drain flows:

- FWHs 25 to the heater drain tank
- Moisture separators to moisture separator drain tanks 21, 22, and 23
- Moisture separator drain tanks 21, 22, and 23 to heater drain tank
- Reheaters to reheater drain tanks 21, 22, and 23

Flow Regimes of Fluid Flow in Piping Downstream of Reheater Drain Tanks LCVs

The reheater drain piping and valves have a lower flow rate during SPU operation than current conditions. Downstream of reheater drain tank level control valves, an unstable flow regime,

such as slug flow, could develop in long horizontal runs. Flow regime is evaluated by computing Baker parameters B_X and B_Y and applying them to Baker's map for two-phase flow regimes.

Scavenging Steam Vent Chamber Discharge Lines

Each moisture separator reheater has a vent chamber discharge line designed to accept approximately 26 percent of the reheater steam flow. These discharge lines are directed to heaters 26 during normal operation and to the condenser during start up. The flow in this line is dictated by a control station. To evaluate two-phase choked flow in the control section, critical flow analysis software has been used that is based on the Henry-Fauske model.

Heater Shell-Side Normal Operating Vent System

The heater shell-side normal operating vent system has been analyzed to confirm that the Heat Exchange Institute (HEI) recommended flow at SPU conditions can be vented (0.5 percent of the steam entering the FWHs at SPU conditions).

Heater Shell-Side and Heater Drain Tank Relief Valves for Overpressure Protection

The FWHs 23, 24, 25, and 26 and the heater drain tank have relief valves for overpressure protection. The set pressure of the heater relief valves should be equal to or less than the design pressure of its shell side. The set pressure of the heater drain tank relief valve should be equal to or less than the heater drain tank design pressure.

The heater shell-side relief valves were evaluated for compliance with the HEI requirement that the relief valve should be capable of passing the larger of the following flows with 10 percent accumulation:

- Minimum of 10 percent of the maximum overload feedwater flow through the heater based on average tube-side temperature

- Flow based on the rupture of 1 heater tube resulting in 2 open ends discharging as orifices

The heater drain tank accepts drain flows from heaters 25 and 26, moisture separator drain tanks, and moisture pre-separator drain tank. The heater drain tank relief valves were evaluated by comparing the total of all drain flows into the tank with the rated capacity of the 2 valves.

Piping and Component Design Pressures and Temperatures

The maximum sustained operating pressures and temperatures of heaters, moisture separators, reheaters and separating tanks drain/vent system at SPU conditions are compared with the piping, tank and heater shell design pressures and temperatures.

Flow-Accelerated Corrosion of Drain and Vent Lines

The piping velocities were calculated at SPU conditions and compared to standard industry velocity criteria as a measure of whether there was a greater potential for flow-accelerated corrosion (FAC). The potentially contributing factors to FAC, such as piping velocities, operating temperatures, piping velocities, and flashing service conditions, were evaluated to determine if a particular pipe needed to be added to the current FAC Program.

Drain Inlet and Outlet Nozzle Velocities of Heaters

The drain inlet and outlet nozzle velocities of heaters at SPU conditions were compared with HEI standard industry guidelines for prevention of undue wear of the nozzles and to determine whether any nozzles needed to be added to the present scope of the FAC Program.

Inlet and Outlet Drain Flows – Effects on FWHs Internals

The SPU drain inlet and drain outlet flow rates of the FWHs were evaluated to determine the effects on tube vibration and erosion of internal subcomponents and support structures.

9.3.4 Acceptance Criteria

Heater, Moisture Separator, Reheater, Separating Tank, and Heater Drain Tank LCVs

The acceptance criteria for the LCV position is that the valve should be open below or near 75 percent at full-load SPU operation to provide adequate assurance of long-term control margin and operability.

Heater, Moisture Separator, and Reheater Gravity Drain Lines

The drain lines with gravity flow should be self-venting if the liquid Froude Number is less than approximately 0.3 at SPU conditions.

Flow Regimes of Fluid Flow in Piping Downstream of Reheater Drain Tanks LCVs

The piping downstream of reheater drain tanks level control valves should not have any unstable flow regime such as slug flow at SPU conditions

Scavenging Steam Vent Chamber Discharge Lines

The scavenging steam vent chamber discharge line should be adequately sized to pass the required flow at SPU conditions.

Heater Shell-Side Normal Operating Vent System

The existing piping design should be capable of removing the expected non-condensable gases per HEI requirement at SPU conditions.

Heater Shell-Side and Heater Drain Tank Relief Valves for Overpressure Protection of Heater Shells and Drain Tank

The set pressure of the heater shell-side relief valve should be equal to or less than the associated heater shell design pressure. The set pressure of the heater drain tank relief valve should be equal to or less than the heater drain tank design pressure.

The HEI-required flow capacity for heater shell-side relief valves should be bounded by design flow capacity of heater shell-side relief valves.

The total incoming flow to the heater drain tank should be bounded by the total design flow capacity of heater drain tank relief valves.

Piping and Component Design Pressures and Temperatures

The acceptance criteria is that the maximum sustained system operating pressures and temperatures at SPU conditions be bounded by design or rated pressures, and temperatures of piping and components.

FAC of Drain and Vent Lines

The piping velocities at SPU conditions associated with the single-phase flow of drain lines for the heater, moisture separator, and reheater drain system should be bounded by the standard industry velocity criteria. Other potential FAC influences, such as operating temperature greater than 200°F, piping material, and flashing service are also considered.

Drain Inlet and Outlet Nozzle Velocities of Heaters

The drain inlet and outlet nozzle velocities of heaters at SPU conditions should not appreciably increase the potential for wear and FAC.

Inlet and Outlet Drain Flows – Effects on FWHs Internals

The SPU drain inlet and drain outlet flow rates of the FWHs should not cause destructive tube vibration or the erosion of internal subcomponents and support structures such that their function is impaired.

9.3.5 Results and Conclusions

The heater, moisture separator, reheater, and pre-separator drain systems are capable of accomplishing their design functions during SPU conditions as discussed in the following paragraphs.

FWH, Moisture Separator, Reheater, Separating Tank, and Heater Drain Tank LCVs

All the drain line LCVs are capable of transporting the required flows at SPU conditions with the open position below or near 75 percent.

LCV 1104 and LCV 1104A and B, and LCV 1105 and LCV 1105 A and B are currently operating successfully over a range of flows and positions. After SPU, the flow rate through all these valves will be reduced, placing less of a burden on the valves.

All level control valves that bypass drains to the condenser are adequately sized, based on the sizing of the corresponding normally operated drain valves.

Heaters 21 emergency dump valves to the condenser are approximately 58 percent open when draining the maximum emergency dump requirement of 10 percent of feedwater flow to the heaters, in the event of heater tube rupture.

Heaters 22 emergency dump valves to the condenser are 99 percent open when draining the maximum emergency dump requirement of 10 percent of feedwater flow, in the event of heater tube rupture. However, the bypass to condenser LCVs on these same heaters are open at 65 to 70 percent and can accommodate part of the heater tube rupture flow without opening more than 75 percent. The corresponding open position of emergency dump valves to the condenser will be less than 75 percent.

There are 6 heater drain tank level control valves (3 large and 3 small) that gradually open to discharge to the heater drains to the condenser in the event of heater drain pump failure. Two large valves out of the 6 total in service are capable of draining all incoming water to the heater drain tank with sufficient margin.

Heater, Moisture Separator, and Reheater Gravity Drain Lines

The Froud Numbers of heater, moisture separator, and reheater gravity drain lines at SPU conditions are less than maximum allowed 0.3 for properly designed gravity drain lines.

Flow Regimes of Fluid Flow in the Piping Downstream of Reheater Drain Tanks LCVs

The flow downstream of the reheater drain control valves is annular at SPU conditions. Hence, the flow in this line should not be restricted due to the anticipated slug flow conditions. A small portion of the piping is in close proximity to slug flow. Since this piping does not currently have problems, such as excessive vibration, it is not expected to have any problems at SPU plant operation.

Scavenging Steam Vent Chamber Discharge Lines

The mass flow in the scavenging steam vent chamber discharge line at SPU conditions is slightly lower than current condition. This discharge line is adequately sized for SPU conditions.

Heater Shell-Side Normal Operating Vent System

The evaluation concluded that the existing piping design of the operating vent system from heaters to condenser is capable of removing the expected non-condensable gases per HEI requirements, at SPU conditions.

Heater Shell-Side and Heater Drain Tank Relief Valves for Overpressure Protection of Heater Shells and Drain Tank

The shells of heaters are not overpressurized in the event of tube or tubesheet failure because:

- The set pressures of relief valves are equal to the design pressures of associated FWHs heaters.
- The HEI-required maximum flow capacity for relief valves for heaters 23, 24 and 25 is bounded by design flow capacity of the relief valves.

- The design flow capacity of relief valves for heaters 26 exceeds the HEI requirement of 1 double-ended tube rupture flow, but is slightly below the HEI requirement of 10-percent feedwater flow. In the event of heater tube rupture, the heater drain LCVs should remain open and have adequate margin to drain the additional small amount of flow to heater drain tank.

The total incoming flow to the heater drain tank (that is, moisture separators drain, moisture pre-separators drain, and heaters 25 and 26 drains) is less than the total relieving flow capacity of heater drain tank relief valves. The set pressures of relief valves are equal to the design pressures of the heater drain tank. Hence, the heater drain tank is not overpressurized in the extreme event of complete loss of both heater drain pumps and the heater tank emergency drain system to condenser.

Piping and Valve Design Pressures and Temperatures

The maximum sustained operating pressures and temperatures of the FWHs and moisture separators, reheaters, and separating tanks drain and vent piping systems at SPU conditions are enveloped by the currently operating piping design pressures and temperatures, except for the following:

- The maximum normal sustained temperature of heater drain tank drain line from last stop valve to condenser exceeds the design temperature by 88°F.

- The maximum normal sustained temperatures of reheaters drains and vents from last stop valve to condenser exceeds the design temperature by 179°F.

The piping is made of carbon steel. The pipe walls are acceptable since the allowable tensile stress value of this carbon steel material remains unchanged in the temperature range of –20 to 650°F. Also, these sections of piping do not have valves and flanges.

The maximum sustained operating pressures and temperatures of heater shells at SPU conditions are enveloped by heater shell design pressures and temperatures except for heaters 24A, B, and C and 26A, B, and C. The maximum normal sustained temperatures of heaters 24A, B, and C exceed the heater design temperature by 10°F. For FWHs 26A, B, and C, the SPU temperature is 28.5°F above design. The shell material of these heaters is carbon steel SA 516 Grade 70. Therefore, the shell design can accept the higher SPU temperatures since the maximum allowable stress value of material SA 516 Grade 70 in tension does not change in the temperature range of –20 to 650°F.

The maximum sustained operating pressures and temperatures of heater, moisture separator, reheater, and separating drain tanks at SPU conditions are enveloped by the tank design pressures and temperatures

FAC of Drain and Vent Lines

The majority of the piping experiences velocities below the industry standard pipe velocity limit. All the carbon steel piping with temperatures exceeding 200°F and flashing service are presently in the FAC Program. The limited number of piping with velocities above the applicable limit are presently in the FAC Program except piping from moisture pre-separators to separating tanks. The temperature of piping from moisture pre-separators to separating tanks exceeds 200°F. This piping was originally included in the FAC Program, however, it has been removed from the Program since the material has been changed to stainless steel clad piping.

Heater Drain Inlet and Outlet Nozzle Velocities of FWHs

The drain inlet and outlet nozzle velocities of the FWHs at SPU conditions are below the HEI standard industry guidelines except for the drain outlet nozzles of heaters 26A, B, and C; 25A, B, and C; 23A, B, and C; and 21A, B, and C and drain inlet nozzles of heaters 22A, B, and C. All of these nozzles are currently included in the FAC Program.

Inlet and Outlet Drain Flows – Effects on FWHs Internals

The SPU drain inlet and outlet flow rates are below the existing design values for FWHs 22A, B, and C; 23A, B, and C; 24A, B, and C; and 25A, B, and C. For FWHs 21A, B, and C, the SPU drain inlet flows are below design values; however, the SPU drain outlet flow rates are above design value. For FWHs 26A, B, and C, the SPU drain inlet and outlet flows are above the design value. FWHs 21A, B, and C and FWHs 26A, B, and C will be monitored to determine whether destructive tube vibration or the significant erosion of internal parts will occur at the higher SPU flow rates.

9.4 Main Feedwater and Condensate System

9.4.1 Introduction

The Main Feedwater and Condensate System was evaluated in conjunction with stretch power uprate (SPU) conditions to determine the extent to which system design parameters bound SPU conditions.

The Condensate System was designed to transport condensate and low-pressure (LP) heater drains from the condenser hotwell through 5 stages of feedwater heating to the suctions of the main feedwater pumps (MFPs). Three one-third capacity condensate pumps are provided.

Two half-size heater drain pumps are designed to transport the high-pressure (HP) heater drains from the heater drain tank into the condensate header upstream of the MFPs.

The Feedwater System increases the pressure of the condensate/heater drains for delivery to the steam generators. The Feedwater System also provides the final stage of feedwater heating and controls the feedwater flow via the regulating valves and feedwater pump turbine speed control system. This system has 2 half-size steam turbine-driven MFPs.

9.4.2 Input Parameters and Assumptions

SPU heat balances were developed to define the thermal plant performance at the current operating conditions and at SPU conditions. A current operating (benchmark) heat balance, tuned to the current plant operating characteristics and SPU heat balances at 1.0- , 1.5-, and 3.0-inch HgA condenser pressures, was used in the evaluation of the system. For evaluation purposes, these heat balances included a margin of approximately 0.5 percent. Each of these heat balances and the corresponding parameters were reviewed and the most conservative case was chosen for the specific evaluation of Main Feedwater and Condensate System.

Plant design basis documents, system descriptions, equipment and piping specifications, calculations, and drawings provided system and component design parameters.

9.4.3 Description of Analysis and Evaluation

Operation at SPU conditions affects a variety of system parameters, such as flow rates and velocities, temperatures and pressures, and the thermal performance of the feedwater heaters (FWHs). The Main Feedwater and Condensate System was evaluated to confirm their ability to operate successfully at the SPU conditions. The following subsections describe the specific evaluation.

Hydraulic Analysis of Condensate, Feedwater, and Heater Drain Pump Systems

The hydraulic model of the Main Feedwater and Condensate System operation under SPU conditions (including associated portions heater drain pumps suction and discharge system) under SPU conditions was developed, and included the following scenario cases:

Case 1: Flow analysis for 3 condensate pumps, 2 feedwater pumps and 2 heater drain pumps in operation at 100-percent power level for the SPU

Case 2: Flow analysis for 2 condensate pumps, 2 feedwater pumps and 2 heater drain pumps in operation at 90-percent power level for the SPU

Case 3: Flow analysis for 3 condensate pumps, 2 feedwater pumps and 2 heater drain pumps in operation to provide 103 percent of full-power feedwater flow for the SPU with steam generator pressure 75 psi above SPU full-power steam generator pressure

Component and Piping Design Pressures and Temperatures

The maximum sustained SPU system operating pressures and temperatures were compared with the piping design/rated pressures and temperatures of piping, valves, flanges, FWH tubes, and pump casings to verify that the design bounds SPU sustained operating conditions.

Flow-Accelerated Corrosion of Condensate, Feedwater, and Heater Drain Pump Piping

The piping velocities were calculated at SPU conditions and compared to standard industry velocity criteria as a measure of whether there was a greater potential for flow-accelerated corrosion (FAC). The potentially contributing factors to FAC, such as piping velocities, operating temperatures, and service conditions, were evaluated to determine if a particular pipe needed to be added to the current FAC Program.

FWHs - Nozzle Velocities, Tube Velocities, and Past Inspection Results

The feedwater inlet and outlet nozzle velocities at SPU conditions were compared to Heat Exchange Institute (HEI) standard industry guidelines to determine the potential for increased wear and FAC. The FWH tube velocities at SPU conditions were compared with HEI-recommended velocities for FWH tubes.

The FWHs inspection results were evaluated to determine whether the actual operating condition of these components, including any existing degradation in performance or component materials, affected their ability to perform under SPU conditions.

Condensate and Heater Drain Pumps Brake Horsepower

The condensate and heater drain pumps' brake horsepower (bhp) at SPU conditions were compared with condensate and heater drain pump motors rated horsepower for acceptability of operation under SPU conditions.

Condenser Operation with Main Steam Dump Resulting from 50-Percent Turbine Load Reduction

The probability of excessive condenser tube vibration and a condenser pressure increase (that is, loss of vacuum) during a main steam dump following a 50-percent turbine load reduction at SPU conditions was evaluated to ensure that the condenser HP alarm and turbine trip setpoint was not exceeded.

Condenser Hotwell Volume

The volume of the condenser hotwell was evaluated to confirm that there would be sufficient volume to accept the condensate flow at SPU full load.

Motor-Operated Valve Program Review

The feedwater pumps discharge motor-operated valves (MOVs) BFD 2-21 and BFD 2-22 are included in the Generic Letter 89-10 MOV Program. The differential pressure calculation for BFD 2-21 and BFD 2-22 were reviewed to evaluate the effects of SPU on the operating parameters for these valves (for example, maximum opening/closing differential pressure, flow rate, and fluid temperature).

9.4.4 Acceptance Criteria

The Feedwater and Condensate Systems are considered acceptable under SPU conditions provided the criteria in the following paragraphs are met.

Hydraulic Analysis of Condensate, Feedwater, and Heater Drain Pump Systems

The feedwater pumps, operating in conjunction with condensate pumps and heater drain pumps, must be capable of providing the required heat balance flow rate and pressure to steam generators at 100-percent SPU power level and transient conditions.

The condensate, feedwater, and heater drain pumps should have sufficient net pump suction (net positive suction head, actual [NPSHA]) with sufficient margin over NPSH, required (NPSHR) at all modes of system operation.

Component and Piping Design Pressures and Temperatures

Maximum sustained system operating pressures and temperatures at SPU conditions should be bounded by the piping design and the component rated (or design) pressure/temperature.

FAC of Condensate, Feedwater, and Heater Drain Pump Piping

The piping velocities and other potential FAC influences, such as operating temperature greater than 200°F, at SPU conditions in the condensate, feedwater, and heater drain pump systems should not cause the potential for increased FAC.

Feedwater Heaters - Nozzle Velocities, Tube Velocities, and Past Inspection Results

The feedwater inlet and outlet nozzle velocities should not significantly increase the wear and FAC of the nozzles. The tube velocities at SPU conditions should be bounded by HEI-recommended velocities.

Condensate and Heater Drain Pumps BHP

The condensate and heater drain pumps' bhp at SPU conditions should be bounded by condensate and heater drain pump motor-rated horsepower.

Condenser Operation with Main Steam Dump Resulting from 50-Percent Turbine Load Reduction

The condenser HP alarm and turbine trip set point should not be exceeded with main steam dump resulting from a 50-percent turbine load reduction.

The tube support spacing recommended by HEI for prevention of tube vibration at SPU conditions should exceed the existing tube support spacing.

Condenser Hotwell Volume

The condenser hotwell should contain sufficient volume to accept full-condensate flow at SPU conditions for a minimum of 5 minutes.

9.4.5 Results and Conclusions

Specific results of each evaluation are discussed below.

Hydraulic Analysis of Condensate/Feedwater/Associated Heater Drain Pump System

The feedwater/condensate/associated heater drain pump system is capable of providing the required heat balance flow rate and pressure to steam generators at 100-percent SPU power level and transient conditions with sufficient margin in control valve open position and feedwater pump turbine speed.

The analysis also confirmed that the condensate, feedwater, and heater drain pumps will have sufficient NPSHA with margin over NPSHR in all modes of system operation.

The results confirmed that the feedwater pump suction header pressures are higher than the pump speed runback set pressures with sufficient margin.

The operating flows of the heater drain pumps during the SPU operating modes indicated that the drain pumps will operate near their runout capability. Although pump operation at runout is not desirable, no changes are recommended at this time. Cold water from the Condensate System is presently injected into the pump suctions to improve the fluid conditions at the pump suctions and avoid cavitation problems.

Component and Piping Design Pressures and Temperatures

The maximum sustained operating pressures and temperatures for piping at SPU conditions are enveloped by the existing piping design pressures and temperatures, except for the maximum normal sustained temperature that exceeds design temperature of condensate pumps suction piping from the condenser. The piping is made of carbon steel. The pipe walls of condensate pumps suction piping from condenser are acceptable at SPU since the allowable tensile stress values of the carbon steel materials remain unchanged in the temperature range of –20 to 650°F. Also, the rated temperature of the valves and flanges (that is, –20 to 150°F) bounds the maximum normal sustained temperature.

The maximum sustained operating pressures and temperatures at SPU conditions are enveloped by the rated/design pressures and temperatures of valves, flanges, FWH tubes, and pump casings.

FAC of Condensate, Feedwater, and Heater Drain Pump Piping

The majority of the piping experiences velocities below the standard industry pipe velocity limit. Most of the piping with temperatures exceeding 200°F is presently in the FAC Program. The limited number of pipes with velocities above the applicable limit and/or temperatures over 200°F (condensate piping between heaters 22A, B, and C and 23A, B, and C) are considered susceptible to FAC and will be added to the FAC Program.

FWHs - Nozzle Velocities, Tube Velocities, and Past Inspection Results

The inspection results of FWHs in November 2002 determined that the tubes, tubesheets, and tube-to-tubesheet joints of FWHs are in good condition. Additionally, very few tubes are plugged or exhibit wall loss. Thus, the thermal performance of the heaters will be very similar to that predicted by the SPU heat balances.

The FWH tube velocities at SPU conditions meet the HEI standard industry guidelines.

The FWH inlet and outlet nozzle velocities at SPU conditions exceed the HEI standard industry guidelines for velocity. The current FAC Program includes the outlet nozzles of feedwater heaters 23A, B, and C and inlet and outlet nozzles of feedwater heaters 24A, B, and C; 25A, B, and C; and 26A, B, and C. The inlet and outlet nozzles of feedwater heaters considered susceptible to FAC will be added to the FAC Program.

Condensate and Heater Drain Pumps BHP

The condensate and heater drain pumps bhp at SPU conditions are enveloped by condensate and heater drain pump motor-rated horsepower.

Condenser Operation with Main Steam Dump Resulting from 50-Percent Turbine Load Reduction

The reduced condenser vacuum is above the condenser low-vacuum alarm set point and the turbine trip set point. Therefore, the turbine will not trip in the event of main steam dump resulting from 50-percent turbine load reduction.

The existing tube support spacing is less than the HEI requirement and, therefore, more structurally adequate. Excessive tube vibration will not occur.

NRC Generic Letter 89-10 MOV Program – Feedwater Pump Discharge MOVs BFD 2-21 and BFD 2-22

The operating parameters in the calculation for maximum opening and closing differential pressure of these MOVs are not affected by the SPU conditions, except for the flow rate. The effect of the increased SPU flow rate on related Generic Letter 89-10 parameters (that is, open and close dynamic thrust values) has been determined to be acceptable.

Condenser Hotwell Volume

The condenser hotwell is of sufficient size to contain the required volume of condensate flow at SPU conditions.

9.5 Steam Generator Blowdown System

9.5.1 Introduction

The Steam Generator Blowdown System (SGBS) is designed to extract blowdown water from the secondary side of the steam generators as a means of removing particulates and dissolved solids to control water chemistry in the steam generators. By maintaining the proper water chemistry, steam generator tube corrosion is reduced, thereby minimizing the likelihood and magnitude of tube leaks. Steam generator blowdown is collected from the steam generator and piped to the blowdown tank, which is vented to the atmosphere and drains to the Service Water System (SWS).

The SGBS also provides samples of the secondary side water in the steam generator. These samples are used for monitoring water chemistry and for detecting the amount of radioactive primary coolant leakage through the steam generator tubes. In the event of a high-radiation signal, both isolation valves in the blowdown lines close automatically. The valves also shut on a Phase A containment isolation signal, an automatic start signal for the motor-driven auxiliary feedwater pumps (MDAFWPs) and also fail shut on loss of air or electrical power.

The portion of the SGBS from the steam generators' connections inside containment, up to and including the containment isolation valves outside containment, are considered a part of the containment boundary and are safety-related.

9.5.2 Input Parameters and Assumptions

The SGBS was originally designed for a blowdown flow of 29,000 lbm/hr total for all 4 steam generators. The system is currently operating with a blowdown flow of 13,550 lbm/hr per steam generator for a total of 54,200 lbm/hr.

The maximum limits for blowdown flow from the IP2 steam generators are:

- Continuous normal flow at 66,300 lbm/hr per steam generator (33,150 lbm/hr from each of the 2 steam generator nozzles).

- Flow at 198,900 lbm/hr per steam generator (99,450 lbm/hr per nozzle) for short periods of operation, not to exceed 1 year cumulative over the life of the steam generator.

9.5.3 Description of Analysis and Evaluation

The SGBS was evaluated to verify that the required blowdown flow could be processed during SPU conditions. The system design pressure, design temperature, pipe sizing, and flow velocities were reviewed against the SPU operating conditions.

For the purposes of evaluating the SGBS, the current normal continuously operating blowdown flow was increased in proportion to the increase in feedwater flow during SPU operation.

The maximum blowdown flow rate to correct water chemistry problems will not be affected by SPU operation since it is chosen by the operator for short periods of operation depending on the severity of the situation and the length of time required for cleanup.

The piping velocities, operating temperatures, piping material, and service time during SPU operation were evaluated for their potential to accelerate pipe corrosion and the need to include these lines in the plant Flow-Accelerated Corrosion (FAC) Program.

9.5.4 Acceptance Criteria for Analysis

The SGBS is considered acceptable under SPU conditions by satisfying the following:

- The piping system can pass the increased blowdown flow rate at SPU conditions.
- The SPU maximum pressure and temperature conditions are bound by the piping and valve design pressures and temperatures.
- Increased flow velocities and/or operating temperatures above 200°F due to SPU conditions will not increase the potential for FAC.

9.5.5 Results

The plant is currently operating with a blowdown flow of 13,550 lbm/hr from each steam generator for a total flow of 54,200 lbm/hr. Since feedwater flow increases at SPU conditions, the blowdown flow must be increased to achieve the same chemistry control in proportion to the SPU increase in feedwater flow to the steam generators. Based on the current operating flows and SPU heat balances, feedwater flow increases approximately 6 percent and, therefore, the SPU normal continuous blowdown flow increases to 14,365 lbm/hr from each steam generator, and a total of 57,455 lbm/hr.

The higher SPU blowdown flow can be accommodated by the existing design since the SGBS is sized for the higher flow rates limits, shown earlier in subsection 9.5.2, used to correct steam generator chemistry. The only impact of SPU operation is the need to reposition the throttle valve on the blowdown line from each steam generator for the slightly higher SPU flow rate.

The velocity in the blowdown lines increases from 3.8 ft/sec at the current operating conditions and 4.0 ft/sec at SPU conditions. Considering the low-flow velocities involved, this increase is not significant.

The maximum velocities in the blowdown lines occur when correcting steam generator chemistry and, although not affected by SPU operation, can be as high as 18.6 ft/sec at 66,300 lb/hr at the normal blowdown limitation, and 55 ft/sec at the absolute upper limit of 198,900 lb/hr.

In regard to FAC, although the normal operating velocities are slow and the maximum velocities only occur for short durations, the blowdown lines are monitored as part of the IP2 FAC Program, since the temperature is 513°F and the lines can experience flashing flow due to the saturated conditions in the steam generators.

The steam generator steam outlet temperature and pressure increases from the original design values of 512.7°F/762.7 psia to 513.1°F/765 psia at SPU conditions. This slight increase (+0.4°F and +2.3 psia) does not affect the main steam safety valve (MSSV) setpoints nor the design pressure and temperature of the steam generators. Therefore, the SGBS design pressure and temperature is not affected.

9.5.6 Conclusions

The design and operation of the IP2 SGBS is acceptable for the SPU conditions. Aside from the need to reposition the throttle valves controlling blowdown flow rate, no changes are required for SPU operation.

Due to their operating temperature and saturated, potentially flashing process conditions, the blowdown lines, inside and outside containment, will remain in the existing FAC Program.

9.6 Essential and Non-Essential Service Water System

9.6.1 Introduction

The Indian Point 2 (IP2) Essential and Non-Essential Service Water System (SWS) is a safety-related system that provides cooling water from the Hudson River to essential (loads that would require cooling water immediately after a loss of power or an accident) and non-essential (loads that do not require cooling water immediately after a loss of power or an accident) components on both the nuclear and conventional sides of the plant. The cooling water removes waste heat from the equipment for all plant operating modes and rejects the waste heat to the Hudson River through a discharge canal. One set of 3 pumps provides water to the essential header and the other set of 3 pumps supplies the non-essential header. During periods of high river water temperature, a cross-connection to Indian Point Unit 1 (IP1) can be aligned to supply the conventional plant loads normally fed from the non-essential header.

9.6.2 Input Parameters and Assumptions

The latest system hydraulic analysis provided the basis for the system alignments, valve/equipment controls and operation evaluated. Inputs for cooling flow rates and heat load requirements were provided by Westinghouse or developed based on the latest plant design specifications, drawings, licensing documents, design basis documents, test data and inspection reports and previous uprate experience and confirmed with equipment suppliers.

9.6.3 Description of Analysis and Evaluations

The stretch power uprate (SPU) will increase the heat rejection to the SWS.

The latest system hydraulic analysis was modified to incorporate the requirements of SPU operation.

The following were evaluated at SPU conditions:

- Heat load removal capability
- Flow adequacy to system components and SWS pump capacity and head
- Effects of higher outlet temperatures versus the existing piping design
- System stress analysis and environmental conditions
- Design pressure and temperature of system piping and components

Operation of SWS motor-operated valves (MOV) in the Generic Letter 89-10 MOV Program and adequacy of the SW system instrumentation and controls was evaluated. Outlet service water temperatures were established for SPU operation and evaluated at SPU conditions.

9.6.4 Acceptance Criteria

The Essential and Non-Essential SWS is considered acceptable under SPU conditions provided the following conditions are met:

- The SWS remains capable of providing the required flow rate for each of its design functions (safety and non-safety) under SPU operating conditions.
- SWS pump operation at SPU flow conditions is within the acceptable margins of pump design parameters (for example, net positive suction head [NPSH], flow and total discharge head [TDH]) for all applicable operating modes.
- The SWS remains capable of performing its heat removal functions (safety and non-safety) specified for each component for all applicable operating modes.
- The SWS piping and components design pressure and temperature bound the SPU pressure and temperature conditions. The existing SWS pipe stress bound SPU conditions and outlet SWS conditions are bound by existing plant environmental conditions.

9.6.5 Results and Conclusions

Adequate SWS and equipment performance was verified under SPU conditions, including pump NPSH requirements, system flashing, strainer backwash capability, etc. SWS MOVs in the Generic Letter 89-10 MOV Program were determined to be adequate. Increased heat loads from the equipment were found to be bounded by the original equipment and system design with additional service water flow required to some components. The increased flow requirements were verified to be within the SWS capability.

SWS instrumentation and controls were found to be adequate at SPU conditions.

Outlet service water temperatures were confirmed to be within the system and equipment design specifications, piping design system stress analysis, and plant environmental limits.

9.7 Circulating Water System and Main Condenser

9.7.1 Introduction

The Indian Point 2 (IP2) Circulating Water System (CWS) is a non-safety-related system that provides cooling water for the main condenser of the turbine generator unit. The CWS is a once-through system that uses 6 CWS pumps to supply water from the Hudson River and circulate it through the main condenser to condense the exhaust steam from the main turbine and other steam/water drains and return heated water back to the Hudson River.

The main condenser is a conventional triple-shell, single-pass, divided waterbox, radial flow surface condenser used to condense and deaerate exhaust steam from each of the 3 low-pressure (LP) turbines, 2 boiler feedwater pump turbine exhausts, the steam dump system, and other miscellaneous drains. Heat is removed by the CWS where it is ultimately rejected to the Hudson River.

The Main Condenser Air Removal System is a non-safety-related system that removes non-condensable gasses from the main condenser to help maintain condenser vacuum. The Condenser Air Removal System consists of 3 steam jet air ejectors (SJAEs), with each SJAE serving 1 condenser shell. Two Nash dual-stage vacuum pumps are provided for startup.

9.7.2 Input Parameters and Assumptions

Thermal cycle heat balances were developed to define the thermal plant performance at the current operating conditions and at SPU conditions. The CWS pumps ratings, main condenser data, and the assumptions made for the Main Condenser Air Removal System were used in the evaluation of the CWS and main condenser.

The SPU evaluation assumed the existing CWS pumps and air removal equipment were not modified and would continue to operate at the same flow rates.

9.7.3 Description of Analyses and Evaluations

Plant operation at the SPU conditions will increase the exhaust steam flow and duty of the main condenser and, therefore, increase the heat load rejected by the CWS to the Hudson River. The existing CWS pumps were not modified for SPU and continue to operate at the same flow rates. Since the CWS inlet temperatures from the Hudson River were not affected by the SPU, the CWS discharge temperature to the Hudson River increased. The SPU resulted in a higher flow of turbine exhaust steam to the condenser, which, in turn, increased the amount of air and non-condensables that needed to be removed from the condenser during plant operation.

The State Pollutant Discharge Elimination System (SPDES) Permit places restrictions on discharge temperatures and discharge flow rates.

The maximum pressure rise in the main condenser was found to result from a main steam dump following a 50-percent load rejection at the turbine while operating at the SPU power level. This abnormal operating condition maximized the incoming steam flow and heat load to the condenser. The potential for excessive vibration of the main condenser tubes and tube supports due to the worst case incoming steam was evaluated in accordance with the requirements and methods of the Heat Exchange Institute (HEI). The existing tube supports and their spacing were confirmed to be acceptable.

9.7.4 Acceptance Criteria

The CWS design is considered acceptable to support SPU conditions provided the following criteria are met:

- CWS pressure is bounded by system piping and component design.
- CWS temperature is bounded by system piping and component design and is within the SPDES limitations.
- CWS pumps provide the required flow to ensure condenser duty requirements are met.
- CWS discharge flows are within the SPDES limitations at SPU conditions.
- Remaining CWS equipment is adequate to support SPU conditions.

The main condenser design is considered acceptable to support SPU conditions, provided the following criteria are met:

- Main condenser thermal performance meets the increased heat loads and power output during SPU operation as required by the SPU heat balances.
- Main condenser pressure with the maximum incoming steam flow and heat load at SPU conditions remains below the main turbine trip setpoint.
- Main condenser tubes and tube support design is adequate to prevent excessive tube vibration with the maximum incoming steam flow at SPU conditions.
- Main condenser auxiliary equipment is adequate to support SPU conditions.

The Condenser Air Removal System must be capable of removing all non-condensables, including air leakages and associated water vapor from the condenser shell, by maintaining a minimum steam condensing pressure. The Condenser Air Removal System is considered acceptable if the SPU requirements are bounded by the system and equipment design capability.

9.7.5 Results and Conclusions

The SPU evaluation confirmed that the existing CWS pumps provided sufficient flow for SPU heat removal and that the discharge temperature was within the SPDES limits. Main condenser duty, corresponding CWS discharge temperatures, steam flows, and condenser pressure increase due to SPU conditions were found to be within original design specifications.

The CWS pressure was not affected by operation at SPU conditions. No physical changes are being made to the CWS pumps, main condenser, piping or auxiliary equipment. Therefore, none of the parameters that affect CWS pressure or inlet operating or design temperatures are affected by operation at SPU conditions.

The main condenser can accept the worst-case steam dump flow without exceeding the turbine trip setpoint and without experiencing excessive tube vibration.

Other main condenser design factors including deaerating effects, tube cleanliness, tube-side velocity, and tube-side friction losses, will not be affected by SPU conditions.

The original sizing of the SJAEs was based on the 1970 version of the HEI Standard. The SJAE capacity envelops the current plant recorded air and non-condensable gas in-leakage to the condenser with sufficient margin such that the SJAEs are considered acceptable for operation under SPU conditions. The capability of the 2 air removal vacuum pumps is not affected by the SPU since these pumps operate only during plant startup.

9.8 Electrical Systems

9.8.1 AC and DC Plant Electrical Systems

An upgrade of reactor thermal power from the existing level of 3071.4 to 3216 MWt will occur as part of the stretch power uprate (SPU) at Indian Point Unit 2 (IP2). The alternating current (AC) and direct-current (DC) electrical distribution system and associated equipment were reviewed to evaluate the impact of the SPU on system and equipment performance, capacity, and capability. Specifically, the following items were evaluated:

- Main generator
- Iso-phase bus (IPB) duct
- Main transformers (MTs)
- Unit auxiliary transformer (UAT)
- Station auxiliary transformer (SAT)
- 6900-V power distribution system (including loads and cables)
- Protective relay schemes
- Miscellaneous systems (480-VAC emergency diesel generators [EDGs], 118-VAC instrument supply systems, and 125-VDC systems)
- Grid stability

The system review also included evaluating the station load flow analysis, the station fault analysis, and grid stability studies. The purpose of the review was to determine if the electrical systems and equipment would operate satisfactorily and continue to perform their intended functions under SPU power levels. The results of the evaluation are described in the following sections.

9.8.1.1 Main Generator

9.8.1.1.1 Input Parameters and Assumptions

The main generator is a turbine-driven, hydrogen cooled, four-pole machine rated 1439.2 MVA, 22 kV, 0.91 power factor at 75-psig hydrogen pressure. The output of the main generator is delivered to the low-voltage windings of the main transformers (MTs) (MT21 and MT22) via the iso-phase bus (IPB) duct. An IPB tap bus connects the main generator output to the unit auxiliary transformer (UAT). Unit operation at SPU conditions will result in increased power output from the unit.

The scope of this review includes an evaluation of the main generator electrical parameters relevant to assessing equipment adequacy at SPU conditions. The review includes an evaluation of the generator operating at 75-psig hydrogen pressure, since this reflects the maximum capability of the machine, as well an evaluation of the generator operating at 60-psig hydrogen pressure, since this reflects the normal operating hydrogen pressure of the machine. (The evaluation of the existing main generator protective relay schemes and grid system stability studies relative to SPU is discussed elsewhere in this report.)

The evaluation of the main generator is based upon the following inputs and assumptions:

- The main generator gross real power output at the reactor thermal power level of 3216 MWt is assumed to be 1100 MWe. This is a bounding value based on the nominal value of 1080 MWe from the heat balance.

- The main generator can provide rated output (1439.2 MVA) when operated from 0.91-power factor, lagging, up to and including unity (1.0) power factor, at 75-psig hydrogen pressure.

- The main generator was assumed to operate within the constraints of the applicable generator capability curve.

- The generator real and reactive power output capacity at SPU conditions will be determined from the generator capability curve.

- The generator currently operates at 60-psig hydrogen pressure and was assumed to operate at 60-psig hydrogen pressure at SPU conditions.

- The main generator can provide approximately 1325 MVA when operated from 0.92 power factor lagging, up to and including unity (1.0) power factor, at 60-psig hydrogen pressure.

- The generator reactive power requirements for normal power operation were assumed to be 600 MVAR lagging and 100 MVAR leading.

9.8.1.1.2 Description of Analysis and Evaluation

The nameplate rating of the main generator is 1439.2 MVA (based on 75-psig hydrogen), 22 kV, 0.91 power factor, three-phase, 60 Hz, 1800 rpm.

The main generator evaluation was based upon a comparison between the generator capability curve and the anticipated operating requirements when the machine operates at SPU conditions. Unit operation at leading and lagging power factor was considered.

9.8.1.1.3 Acceptance Criteria

- The generator real power output capability (MW) does not limit turbine output capability at SPU conditions.

- The generator reactive power requirements will not exceed 600 MVAR lagging, and 485 MVAR leading, when the unit is operating at SPU conditions and 60-psig hydrogen pressure.

9.8.1.1.4 Results and Conclusions

The real power output (MW) capability of the main generator meets the requirements for operation at SPU conditions. The generator capability curve shows that the machine is capable of continuous operation at an output of 1219 MW (0.92-lagging power factor) up to and including 1325 MW (unity power factor) at 60-psig hydrogen pressure. Maximum required unit output at SPU conditions is assumed to be 1100 MW. Therefore, the real power output (MW) capability of the main generator is significantly higher than the real power output required at SPU conditions.

The reactive power capability (MVAR) of the main generator from the generator capability curve is 600 MVAR lagging, at 60-psig hydrogen pressure when the unit operates at SPU conditions. Machine operation at the specified values corresponds to a generator lagging power factor of 0.878 at SPU conditions and 60-psig hydrogen pressure. Machine leading reactive power capability is 485 MVAR at 60-psig hydrogen pressure when the unit operates at SPU conditions.

Machine operation at the specified values corresponds to a generator leading power factor of 0.915 at SPU conditions and 60-psig hydrogen pressure.

The reactive capability of the main generator meets or exceeds the normal power requirement of 600 MVAR lagging and 100 MVAR leading, and the IP2 reactive power commitments.

A review of the generator capability curve confirms that the main generator is adequate to support unit operation at SPU load conditions. Additionally, the main generator is adequate to support contractual agreements regarding machine leading and lagging reactive power requirements (MVAR).

9.8.1.2 Iso-Phase Bus Duct

9.8.1.2.1 Input Parameters and Assumptions

The output of the main generator is delivered to the low-voltage windings of the main transformers (MT21 and MT22) via the IPB duct. An IPB tap bus connects the main generator output to the UAT. Unit operation at uprate conditions will result in increased power output from the unit and an attendant increase in MT and UAT loading. Accordingly, the IPB main and tap bus conductor current will also increase.

The scope of this review includes an evaluation of IPB electrical parameters relevant to assessing equipment adequacy at SPU conditions.

Evaluation of the IPB is based upon the following inputs and assumptions:

- The Iso-Phase Bus System is organized into segments. The first segment runs from the generator terminals to the point where the main bus splits into the 2 segments that run to the 2 MTs. This first segment has a forced air-cooled rating of 32 kA at 22 kV, 65°C. The second segment of the main bus runs from the split to each MT. These segments have a forced air-cooled rating of 16 kA at 22 kV, 65°C. The third segment runs from 1 split bus tap to the UAT. This segment has a self-cooled rating of 1.5 kA at 23 kV. This segment does not have a forced-cooled rating.

- The transformer test report shows that the 2 MTs have identical MVA ratings and impedances. Since the current splits evenly between the transformers in proportion to the impedance, current to each MT primary winding is approximately the same.

- The highest IPB loading will occur when the house loads are fed from the UAT. The 16 kA portion of the bus between the split and UAT tap is the most limiting since it carries the generator output to one MT plus the UAT load.

- The highest IPB loading will occur when the generator is operating at minimum voltage and maximum generator output. Industry standards allow generator operation within a voltage range of ± 5 percent.

- The generator is assumed to be operating at 60-psig hydrogen pressure.

Fault current at the IPB is a function of equipment parameters associated with the main generator, MT, auxiliary transformer, etc. Since the SPU does not change any relevant equipment parameters, unit operation at SPU conditions will not adversely affect IPB fault duty.

9.8.1.2.2 Description of Analysis and Evaluation

The evaluation of the IPB main and tap buses was based upon a comparison between the maximum anticipated full-load current and the design ratings of the main and tap bus conductor with the generator operating at both lagging and leading power factor. This evaluation was based on house loads being fed from the UAT, since this results in the worst-case IPB loading. Since the IPB main and tap bus short circuit design ratings were adequate prior to the SPU and the SPU did not adversely affect IPB fault current levels, the IPB main and tap bus short circuit design ratings are adequate for SPU.

9.8.1.2.3 Unit Operation at Lagging Power Factor

The generator capability curve was reviewed to identify gross generator output when the unit operates at SPU conditions with main generator operation at lagging power factor.

Table 9.8-1 shows the IPB loading with the generator operating at the SPU level lagging power factor (60-psig hydrogen).

9.8.1.2.3.1 Unit Operation at Leading Power Factor

Review of the generator reactive capability curve confirmed that the machine is capable of importing reactive power at SPU load conditions.

Table 9.8-2 shows the IPB loading with the generator operating at the power uprate level, leading power factor (60-psig hydrogen)

9.8.1.2.3.2 Tap Bus

Maximum anticipated full-load current for the tap buses results when the connected UAT operates at maximum output load conditions. Based on a review of calculated transformer loading included in the applicable station load flow analysis, this occurs when the unit is operating at maximum full-load conditions. The calculated UAT loading is identified in Table 9.8-3. The resulting tap bus current is shown in Tables 9.8-1 and 9.8-2.

9.8.1.2.4 Acceptance Criteria

- The continuous current rating of the IPB main bus is equal to or greater than the required IPB bus ampacity at maximum generator output (MVA).
- The continuous current rating of the IPB tap bus is equal to or greater than the required bus ampacity at maximum UAT loading.
- Short circuit current ratings of the IPB main and tap buses are equal to or greater than the calculated available fault current at SPU conditions.

9.8.1.2.5 Results and Conclusions

IPB Main Bus

The anticipated worst-case bus loading at SPU conditions (see Tables 9.8-1 and 9.8-2) exceeds the continuous current rating of the IPB main bus as originally designed. Based upon the results of an evaluation performed by the IPB vendor, the existing coolers will be upgraded to provide the additional main bus ampacity required to support unit operation at SPU conditions.

Since the IPB main bus short circuit design ratings were adequate prior to the SPU and the SPU does not adversely affect available fault current levels, the IPB main bus short circuit design ratings are considered adequate for the SPU.

IPB Tap Bus

The continuous current rating of the IPB tap bus exceeds the anticipated worst-case bus loading at SPU conditions with substantial margin (that is, 1500 amps versus 1210 amps, see Tables 9.8-1 and 9.8-2). Accordingly, the IPB tap bus is considered adequate to support the SPU.

Since the IPB tap bus short circuit design ratings were adequate prior to SPU, and SPU does not adversely affect available fault current levels, the IPB tap bus short circuit design ratings are considered adequate for the SPU.

9.8.1.3 Main Transformers

9.8.1.3.1 Input Parameters and Assumptions

The MTs provide the interface between the main generator and the power system grid. Main generator output power is delivered to the primary windings of the MT at 22 kV. The MT steps up generator output to 345 kV and delivers the output to the 345-kV switchyard. Unit operation at SPU conditions will result in an attendant increase in MT output loading.

The scope of this review included an evaluation of MT capacity based upon a comparison between transformer nameplate rating and the maximum transformer loading at SPU conditions. The review also included an evaluation of remaining transformer life expectancy and existing MT cooler capacity.

Evaluation of the MT is based upon the following inputs and assumptions:

- The main generator real power output will be 1100 MWe.
- The generator is assumed to be operating at 60-psig hydrogen pressure.
- The MT will be evaluated using generator reactive power requirements of 600 MVAR lagging and 100 MVAR leading. These are assumed reactive power requirements for normal operation.
- The unit auxiliary system (house) load will be supplied from the main generator via the UAT, and from offsite via the station auxiliary transformer (SAT). This is consistent with the normal plant configuration when the unit is operating at full power.
- The load to the UAT is assumed to be 40 MVA at 0.76 lagging power factor based on a review of the normal load flow runs. This assumed UAT load is lower than the calculated load (42.88 MVA, see Table 9.8-3), and therefore is conservative for sizing the MT, since it results in a higher MT output.

Description of Analysis and Evaluation

The MT consists of 2 half-size generator step-up transformers MT21 and MT22. The nameplate rating for each transformer is 542/607 MVA (fans, oil, and air) at 55°C/65°C rise, 20.3 kV primary, 345 kV secondary, three-phase, 60 HZ. The transformers, manufactured by Westinghouse, have been in service for approximately 29 years.

The MT evaluation was based upon a comparison between the applicable transformer design ratings and the anticipated operating requirements when the unit operates at SPU conditions. Unit operation at leading and lagging power factor conditions was considered assuming the main generator operates within the limits identified previously in subsection 9.8.1.3.1.

9.8.1.3.2.1 MT Loading

MT loading at SPU conditions is determined in Table 9.8-4 assuming house loads are supplied from the UAT and the main generator real and reactive power requirements are as defined previously in subsection 9.8.1.3.1. This is considered bounding for the purpose of determining maximum MT output loading.

MT loading determined above assumes house loads are supplied from the main generator via the UAT, consistent with the normal plant configuration when the unit is operating at full power. Plant operating scenarios evaluated in the load flow analysis assume the house loads are supplied from the main generator via the UAT and from offsite via the SAT, consistent with the assumption included in subsection 9.8.1.3.1. However, based upon a review of the calculated MT loading, each MT has adequate capacity to support unit operation at SPU conditions even if the house loads are supplied entirely from the SAT. If the house load were removed from the UAT, the MT output load would increase to approximately 1175.9-MVA lagging power factor and 1140.5-MVA leading power factor. The calculated load is still less than the maximum MT rating of 1214 MVA at 65°C rise.

9.8.1.3.2.2 MT Cooler Evaluation

The transformer life expectancy study will include an evaluation to determine if the existing coolers are adequate as installed, or if modifications are required to permit continuous transformer operation near the 65°C rise rating (1214 MVA) to support unit operation at SPU conditions.

9.8.1.3.3 Acceptance Criteria

MT design rating will not limit unit operation at uprate conditions.

9.8.1.3.4 Results and Conclusions

The maximum calculated load for the MT is 1137.6 MVA with house load from UAT (see Table 9.8-4). The transformers maximum rating is 1214 MVA at 65°C rise over ambient.

The preceding evaluation confirms that the existing MT nameplate rating is adequate to support unit operation at SPU conditions when the main generator is operated in accordance with the assumptions included previously in subsection 9.8.1.3.1. It is also reasonable to conclude that the MT is adequately sized to support unit operation at SPU even if the SAT supplies the entire house load.

9.8.1.4 UATs

9.8.1.4.1 Input Parameters and Assumptions

Power required for station auxiliaries during normal operation is split between the UAT and the SAT. Power to the auxiliaries (house loads) on 6900-V buses 1 through 4 is supplied by the UAT that is connected to the main generator via the IPB. Unit operation at SPU conditions will result in an increase in UAT output loading because the brake horsepower (bhp) required by several large-pump motor drives supplied from 6900-V buses 1 through 4 will increase due to the SPU.

The scope of this review included an evaluation of UAT design capacity based upon a comparison between transformer nameplate rating and the maximum transformer loading at SPU conditions.

The evaluation of the UAT is based upon the following inputs and assumptions:

- The house load is shared between the UAT and the SAT. This is consistent with the normal plant configuration when the unit is operating at full power.

- IP2 load flow analysis of the Electrical Distribution System will be used to evaluate the effect of the SPU on the Electrical Distribution System.

9.8.1.4.2 Description of Analysis and Evaluation

The nameplate rating for the UAT is 43.00 MVA FOA at 55°C rise and 48.16 MVA FOA at 65°C rise, 22 kV primary, 6900 V secondary winding, three-phase, 60 Hz. The secondary winding supplies power directly to downstream 6900-V buses 1 through 4. The UAT secondary is

equipped with an automatic load tap changer that regulates voltage to a preset value at the downstream 6900-V normal buses.

9.8.1.4.2.1 UAT Loading

IP2 load flow analysis of the Electrical Distribution System was used to evaluate the effect of SPU on the UAT. A baseline for transformer loading was developed using the values in the load flow. The incremental changes in loading due to the SPU were added to the baseline and the resulting values were compared to the UAT rating to determine the equipment adequacy for the SPU.

The station load flow analysis was reviewed to identify maximum calculated UAT loading. The analysis determined UAT loading during normal operation, unit trip, hot shutdown, plant shutdown using 13.8kV and gas turbines, SI, and cold shutdown conditions. Review of the calculated results confirmed that worst-case UAT loading occurs during normal operation. To determine UAT loading at SPU conditions, the incremental loading is combined with the existing loading. The resulting UAT loading was summarized in Table 9.8-5.

9.8.1.4.3 Acceptance Criteria

UAT nameplate rating of 43.00 MVA FOA at 55°C rise and 48.16 MVA FOA at 65°C rise will not limit unit operation at SPU conditions.

9.8.1.4.4 Results and Conclusions

The worst-case total secondary load on the UAT is 38.73 MVA (see Table 9.8-5), which is less than the UAT maximum nameplate rating of 43.00 MVA FOA at 55°C rise and 48.16 MVA FOA at 65°C rise. Accordingly, it is reasonable to conclude that the transformer temperature rise will be within the design ratings and the transformer coolers will be operating within their design capacity when the unit operates at SPU conditions.

The UAT has adequate capacity to support unit operation at SPU conditions, and based on the transformer condition, the UAT will require no modifications to the cooling system to meet SPU conditions.

9.8.1.5 SATs

9.8.1.5.1 Input Parameters and Assumptions

A single SAT serves IP2. Offsite power from the 138-kV switchyard is supplied to the 6900-V buses via the SAT during normal operation, plant start-up, outage, and design bases accident (DBA) conditions. Power required for station auxiliaries (house loads) during normal operation is split between the UAT and the SAT, with house loads on 6900-V buses 5 and 6 supplied by the SAT, which is connected to the 138-kV switchyard via overhead lines. During normal operation, on a generator trip other than a generator over-frequency trip, a "deadfast" transfer scheme ties buses 1 and 2 to bus 5, and buses 3 and 4 to bus 6.

Unit operation at SPU conditions will result in an increase in SAT output loading when the house loads are transferred, because the bhp required by several large pump motor drives supplied from 6900-V buses 1 through 4 will increase due to power uprate.

The scope of this review included an evaluation of SAT design capacity based upon a comparison between transformer nameplate rating and the maximum transformer loading at SPU conditions.

Evaluation of the SAT is based upon the following inputs and assumptions:

- The normal source of auxiliary power for 6900-V buses 5 and 6, and standby power required during plant startup, shutdown, and after reactor trip is the SAT.

- IP2 load flow analysis of the Electrical Distribution System will be used to evaluate the effect of uprate on the Electrical Distribution System.

9.8.1.5.2 Description of Analysis and Evaluation

The nameplate rating for the SAT is 43.00 MVA OA (oil and air)/FOA at 55°C rise and 48.16 MVA OA/FOA at 65°C rise, 138 kV primary, 6900-V secondary winding, three-phase, 60 Hz. The secondary winding supplies power directly to downstream 6900-V buses 5 and 6. The SAT secondary is equipped with an automatic load tap changer that regulates voltage to a preset value at the downstream 6900-V buses.

9.8.1.5.2.1 SAT Loading

IP2 load flow analysis of the Electrical Distribution System was used to evaluate the effect of SPU on the SAT. A baseline for transformer loading was developed using the values in the load

flow. The incremental changes in loading due to SPU were added to the baseline and the resulting values were compared to the SAT rating to determine the equipment adequacy for the SPU.

The station load flow analysis was reviewed to identify maximum calculated SAT loading. The analysis determined SAT loading during normal operation, unit trip, hot shutdown, plant shutdown using 13.8 kV and gas turbines, SI, and cold shutdown conditions. Review of the calculated results confirmed that worst-case steady-state SAT loading occurs during SI. To determine SAT loading at SPU conditions, the incremental loading is combined with the existing loading. The resulting SAT loading is summarized in Table 9.8-6.

9.8.1.5.3 Acceptance Criteria

SAT nameplate rating of 43.00 MVA FOA at 55°C rise and 48.16 MVA FOA at 65°C rise will not limit unit operation at the SPU conditions.

9.8.1.5.4 Results and Conclusions

The worst-case total secondary load on the SAT is 41.38 MVA (see Table 9.8-6), which is less than the SAT maximum nameplate rating of 43.00 MVA OA/FOA at 55°C rise, and 48.16 MVA OA/FOA at 65°C rise. Accordingly, it is reasonable to conclude that the transformer temperature rise will be within the design ratings and the transformer coolers will be operating within their design capacity when the unit operates at SPU conditions.

The SAT has adequate capacity to support unit operation at SPU conditions, and based on the transformer condition, the SAT will require no modifications to the cooling system to meet the SPU conditions.

9.8.1.6 Medium-Voltage, 6900-V System

9.8.1.6.1 Input Parameters and Assumptions

The 6900-V system supplies power for the majority of the safety- and non-safety-related AC loads. Large station loads are supplied directly from the system. Smaller low voltage AC loads (480 V and below) are also supplied from the system via appropriately rated step-down transformers. During normal full-load operation the system is supplied power from the UAT and the SAT, with buses 1 through 4 supplied by the UAT, and buses 5 and 6 supplied by the SAT. During plant start-up, shutdown, outage, and plant accident conditions the system is supplied from the SAT.

Unit operation at SPU conditions will result in increased fluid system flow requirements that will, in turn, increase the bhp load on several medium-voltage pump motor drives supplied from the 6900-V buses. The 6900-V non-segregated phase bus, switchgear, medium-voltage motors, and associated feeder cables affected by SPU are discussed in this section.

The scope of this review included an evaluation of the 6900-V system to confirm adequacy of the applicable switchgear ratings and to confirm that bus voltage levels are adequate to support equipment operation and function when the unit operates at SPU conditions. The review also included an evaluation of motor load requirements at SPU conditions to verify that the affected pump motor drives and associated feeder cables will operate within their rated capability.

The evaluation of the 6900-V system was based on the following inputs and assumptions:

- Loading from the station load flow analysis, together with incremental changes in loading due to the SPU, determine the 6900-V system loading requirements at SPU conditions.
- Revised load bhp data for balance-of-plant (BOP) 6900-V motor driven pumps is as determined in the bhp for condensate and heater drain pumps at 3282.5 MWt analysis, based upon unit operation at a reactor thermal power level of 3216 MWt.
- Revised bhp data for the reactor coolant pumps (RCP) is as specified in Sections 4 and 5 of this report.
- Cable ampacities are assumed values based on protective relay settings.

9.8.1.6.2 Description of Analysis and Evaluation

The effect of the power uprate on the 6900-V non-segregated phase bus, switchgear buses and breakers, station bus voltage levels, and the 6600-V pump motor drives and associated feeder cables is discussed separately below.

9.8.1.6.2.1 6900-V Switchgear

The evaluation of the 6900-V non-segregated phase bus, switch-gear buses, and breakers was based upon a comparison between the applicable equipment ratings and the anticipated operating requirements at SPU conditions, as determined in the station load flow and following analysis. The continuous current design ratings of the 6900-V non-segregated phase bus, switchgear, and switchgear breakers are potentially affected by unit operation at the SPU because of the attendant increase in electrical load flow throughout the system. Conversely, 6900-V equipment short circuit duty is not expected to be adversely affected because unit

operation at SPU conditions does not require any equipment changes, replacements, and/or new installations that could increase the fault current duty at the 6900-V level. Since no new 6900-V loads were added as a part of the SPU, the existing switchgear physical arrangement remains unchanged.

6900-V Bus Incoming Supply Breaker and Motor Feeder Breaker Continuous Current Ratings

The station load flow analysis was reviewed to identify the maximum calculated steady-state loading for the 6900-V incoming supply breakers. The analysis determined equipment loading during maximum normal full load, design bases accident, and outage load conditions with either the UAT and/or the SAT, as applicable, supplying power for the house loads. A review of the calculated results confirmed that worst-case loading occurs on buses 1 through 4 during normal operation and that worst-case steady-state loading occurs on buses 5 and 6 during SI. The incremental uprate loading is combined with the existing loading, and the resulting bus loading is summarized in Table 9.8-7. A comparison between the expected uprate load conditions, as determined in this evaluation, and the continuous current ratings of the switchgear incoming supply breakers is also provided in Table 9.8-7. Note that the non-segregated bus comprises segments with ratings of 4000A, 2000A, and 1200A as shown on the 6900-V one line diagram. The non-segregated bus loading is also provided in Table 9.8-7.

A comparison between the maximum continuous current load at uprate and the design rating of the affected motor feeder breakers is shown in Table 9.8-8.

6900-V Equipment Fault Current Ratings

Short circuit duty is not adversely affected by equipment load changes associated with unit operation at SPU conditions. This is because the load changes did not require replacement of, or changes to, existing electrical components and equipment (for example, motor drives, power transformers, and feeder cables) that could result in increased equipment fault current duty at the 6900-V buses.

9.8.1.6.2.2 Medium-Voltage (6600 V) Motors and Motor Feeder Cables

6600-V Motors

Unit operation at SPU conditions will result in a pump load change on several 6600-V motor-driven pumps. Specifically, the condensate, heater drain, and RCP motors, all of which are supplied from the 6900-V switchgear buses, will each experience a load change. The evaluation of each affected motor drive was based upon a comparison between the motor

nameplate rating (HP) and the required motor bhp at uprate flow conditions. It should be noted that the condensate and heater drain pump motors actually operate at a bhp that is less than the respective bhp shown in the current load flow analysis. A summary of the comparison data is shown in Table 9.8-9.

All affected BOP motor drives will be operating at less than nameplate rating when the unit operates at SPU conditions. RCP motors were evaluated with information provided by Westinghouse.

6600-V Motor Feeder Cables Including the RCP Electrical Penetrations

The existing motor feeder cables were evaluated to confirm that cable ampacity was equal to or greater than motor full-load current when the associated motor operates at SPU load conditions. The comparison between feeder cable ampacity and motor full-load current was developed as follows:

- Motor load flow (MW and MVAR) for each motor was calculated based on the motor bhp requirements at SPU conditions and motor efficiency and power factor data taken from the applicable motor data sheets, where:

  Motor MW = (bhp x 0.746) / (efficiency x 1000)

  Motor MVAR = (MW / pf) * [sin (arcos(pf))].

- Motor full-load current was calculated using the calculated motor load flow data and assuming a motor terminal voltage equal to 90 percent of motor-rated voltage for conservatism, where:

  $$\text{Motor Full Load Current} = \frac{\sqrt{MW^2 + MVAR^2}}{\sqrt{3} \times 6.6kV \times 0.9} \times 1000$$

- Cable ampacity is assumed to be equal to or greater than the overload protective relay pickup (1.25 x BL-1 relay setting). It should be noted that in the case of the RCP motor feeder cables, cable ampacity is assumed to be equal to or greater than the BL-1 relay pick up point (768 amps), shown in the 6.9kV coordination study, rather than 1.25 x BL-1 relay setting.

The results are shown in Table 9.8-10.

Evaluation of the electrical penetrations associated with the RCP motor feeders was based upon a comparison between motor full-load current during cold loop operation and penetration rated ampacity. Cold loop values bound normal-operating full-load current. The penetrations associated with the RCP motor feeders consist of 2 feed-through conductors per phase each rated 351 amps, continuous for a total ampacity of 702 amps. RCP motor full-load current under cold loop conditions and maximum steam generator tube plugging adjusted for motor operation at 90-percent rated voltage, results in motor full-load current of 636 amps (318 amps/ penetration conductor). Since the required ampacity is less than the rated penetration ampacity, the penetrations are considered adequate. Additionally, if the penetration feed-through conductor current were 318 amps, heat dissipated within the penetration would also be less than the specified acceptance value of 22 watts/foot identified in the electrical penetration ampacity, short circuit, and heat generation calculation.

9.8.1.6.2.3 System Voltage Levels

The existing load flow calculation analyzes a number of load flow (steady-state) and motor start scenarios. The estimated bus and motor terminal voltage for normal-power operation at SPU conditions and the estimated bus and motor terminal voltage for SI conditions at uprate were evaluated to determine the extent of impact on the 6900-V system. The evaluations show that the worst-case reduction of 6900-V system voltage levels as a result of the SPU is less than 0.50 percent.

9.8.1.6.3 Acceptance Criteria

Unit operation at SPU conditions will not result in:

- Continuous current or fault current requirements that exceed the applicable design ratings of the 6900-V switchgear or circuit breakers
- Operation of 6600-V motors at loads greater than rated motor horsepower
- Full-load current requirements that exceed motor feeder cable ampacity or result in excessive cable voltage drop
- Minimum-voltage levels at the 480-V buses that are less than the voltage required to reset the degraded voltage relays
- Protective relay requirements that exceed the capability of the applied electrical protection schemes

9.8.1.6.4 Results and Conclusions

9.8.1.6.4.1 6900-V Switchgear

The continuous current design ratings of the 6900-V switchgear incoming supply breakers and buses, and the affected motor feeder breakers exceed the SPU load requirement. For example, the worst-case load current for the incoming supply breakers is 1838 amps, whereas the breakers and buses are rated 2000 amps. Also, the worst-case load current for the motor feeder breakers is 636 amps, whereas the breakers are rated 1200 amps. Short circuit duty is not adversely affected by equipment load changes associated with unit operation at SPU conditions, because the load changes did not require replacement of, or changes to, existing electrical components and equipment.

9.8.1.6.4.2 6600-V Motors

All affected 6600-V motor drives will operate at less than nameplate rating when the associated systems are operating at SPU flow conditions during normal full-load operation.

9.8.1.6.4.3 6600-V Motor Feeder Cables Including RCP Electrical Penetrations

The ampacity of the existing 6600-V motor feeder cables exceeds the motor full-load current required when the associated motors operate at SPU load conditions. The minimum available margin between the calculated cable ampacity and motor full-load current at SPU conditions is approximately 21 percent (RCP motor feeders during cold loop operating conditions).

Similarly, the containment electrical penetrations associated with the RCP motor feeders are considered adequate for use at SPU load conditions, based upon the discussion included earlier in subsection 9.8.1.6.2.2.

9.8.1.6.4.4 6900-V System Voltage

The estimated bus and motor terminal voltage for normal power operation at SPU conditions, and the estimated bus and motor terminal voltage for SI conditions at SPU were evaluated and results showed that the worst-case reduction of 6900-V system voltage levels as a result of the SPU is less than 0.50 percent.

9.8.1.6.4.5 6600-V Motor Protective Relays

The review concluded that the existing 6900-V switchgear and breakers are adequate as installed to support unit operation at SPU conditions. The review also estimated the effect that

the SPU will have on 6900-V system voltage levels for normal operation and SI conditions. The review showed that the effect on 6900-V system voltage levels as a result of the SPU is minimal.

9.8.1.7 Protective Relay Schemes

Plant electrical equipment is provided with protective relay schemes that are designed to prevent or minimize equipment damage and to limit equipment outages to the immediately involved equipment or component during system disturbances. Protective relay schemes associated with equipment affected by the SPU may, in turn, be affected because of the change in the protected equipment operating point. Accordingly, the scope of this review is limited to an evaluation of affected equipment protection schemes.

9.8.1.7.1 Input Parameters and Assumptions

The evaluation of the protective relay schemes was based, in part, on the following inputs and assumptions:

- Existing protective relay schemes are in accordance with the IP2 Electrical System description, Overall Unit Protection System Description, Overall Unit Protection System design basis document (DBD), and are as shown on the one-line diagrams.

9.8.1.7.2 Description of Analysis and Evaluation

Protective relay schemes for the main generator, main transformer, UATs and SATs and those medium-voltage motors affected by the SPU were reviewed to evaluate the effects of unit operation at SPU conditions.

9.8.1.7.2.1 Unit Equipment Protection

Entergy maintains the existing unit equipment protection schemes and associated setpoints. The review confirmed that the schemes and the associated setpoints were unaffected. The following paragraphs summarize of the protective relay scheme review.

Main Generator Protection

The applied main generator protection schemes are intended to limit machine damage for internal fault conditions and prevent machine damage during abnormal operating or external fault conditions. A review of one-line diagrams and the protection system description confirms that the applied schemes are dependent upon machine ratings and design parameters, and the

design of the connected system. They are not affected by machine operation at SPU conditions. For example, overlapping differential schemes provide machine protection for both internal (generator differential and unit differential schemes) and external (unit differential scheme) phase fault conditions. The schemes are not affected by load changes within the rated operating range of the machine. Ground fault protection schemes, provided by ground over-voltage relays, are designed and set based upon the system grounding design and are independent of main generator output. Loss of excitation and negative sequence protection schemes that are included among the remaining main generator protection schemes are similarly unaffected by unit operation at SPU conditions because the machine will be operated within its rated capability.

MT, UAT, and Auxiliary SAT

A review of one-line diagrams, electrical system description, protection system description, and the protection system design basis documentation (DBD) indicates that transformer protection essentially consists of high-speed phase fault protection and ground fault protection.

The MT is protected by a dedicated differential relay scheme. Back-up protection is provided by the unit differential relay and the MT ground overcurrent relay.

The UAT is protected by a dedicated differential relay scheme and overcurrent protection relay scheme for internal phase and ground faults, as well as faults within the 6900-V bus sections fed, by the transformer. A neutral time overcurrent relay provides ground fault protection to the low-voltage winding. Back-up protection is also provided by these UAT protective relay schemes.

The SAT is protected by a differential relay scheme. Back-up protection is provided by single-phase overcurrent relays and a ground overcurrent relay.

Since the existing transformers will continue in service and operate within their nameplate ratings, the existing electrical protection schemes are unaffected when the units operate at the SPU conditions.

9.8.1.7.2.2 Medium-Voltage Motor Protection

The essence of the medium-voltage (6600-V) motor and motor feeder protection scheme is to provide electrical protection against the damaging effects of sustained overload, locked rotor, and phase and ground fault conditions. For example, instantaneous overcurrent and time overcurrent relays provide phase and ground fault protection and motor overload protection, respectively. The protection scheme also incorporates thermal overload relays.

The applied motor protective relay schemes design is based upon motor application, ratings, and design parameters and feeder ratings. Since the BOP motors affected by the SPU will be operated within their respective rated capabilities, and because none of the affected motor drives will be replaced, operation at SPU conditions will not affect the existing BOP medium-voltage motor protection schemes.

9.8.1.7.2.3 EDG

Loading associated with the EDGs is bounded by operation at SPU conditions, as described in subsection 9.8.1.8.2.2, below. Since no new loads or EDG changes have been identified, the existing EDG electrical protection schemes are similarly unaffected.

9.8.1.7.3 Acceptance Criteria

Protective relay schemes and associated setpoints will not constrain equipment operation at SPU load conditions.

9.8.1.7.4 Results and Conclusions

Protective relay schemes associated with the 6900-V equipment provide adequate margin to permit unit operation at SPU conditions. The review also determined that unit equipment protection schemes (that is, main generator, MTs, UATs, and the SATs) are adequate as installed. Results of the evaluation of the RCP motor protection scheme will be available later.

The review concluded that the protection schemes are considered adequate for use at SPU conditions.

9.8.1.8 Miscellaneous Systems

This section includes evaluations of various systems that are either not impacted, or are marginally impacted by the power uprate. The systems include:

- 480-V System
- EDG System
- 118 VAC Instrument Supply Systems
- 125 VDC Systems

9.8.1.8.1 Input Parameters and Assumptions

Relevant inputs and assumptions are identified within the evaluation discussion for the applicable system.

9.8.1.8.2 Description of Analysis and Evaluation

9.8.1.8.2.1 480-V System

The 480-V system is arranged as 4 engineered safety features (ESFs) buses. The system supplies 480 V, three-phase power to various essential and non-essential electrical loads. The system consists of 2 switchgear assemblies and a distribution network comprising a number of motor control centers and distribution panels for power distribution throughout the station. Each switchgear assembly houses 2 independent buses. The switchgear assembly contains the buses, bus supply breakers, bus tie breakers, load feeder breakers, station service transformers, and potential transformers. Each switchgear bus receives its normal source of power from a dedicated 6900-V/480-V AC station service transformer via normal supply breakers. Each transformer is supplied from a separate 6900-V bus. Each switchgear bus receives its emergency power from 480-V EDGs via emergency supply breakers.

The 480-V system loading is slightly affected by operation at SPU conditions when the switchgear bus receives its power from the EDGs, based on the Westinghouse assessment of the peak EDG loads. This assessment was done by evaluating the change in the peak load from WCAP-12655, Rev. 2. The peak load occurs after recirc switch 4 (right after the recirc pump starts) on EDG 21, with failure of EDG 23. Previously the peak load was 2268 kW (Table 5.5-2a, page 5-39 of the WCAP). As a result of the stretch power uprate, the containment recirculation fan power requirement changes from 223 to 227 kW per fan due to the increased density of the containment atmosphere. However, based upon containment analysis at SPU conditions, fan power requirements drop to 216 kW after recirc switch 4 actuates, which is a reduction of 7 kW per fan, when compared to the previous Westinghouse analysis. Since EDG 21 supplies 2 fan coolers, the net decrease is 14 kW. The resulting peak load, therefore, is reduced from 2268 to 2254 kW. This is less than the 2300-kW half-hour limit of the EDG, and therefore will be adequate.

The CS pump, which is still running at this time in recirc, operates against a slightly higher containment pressure. Since the containment back-pressure was conservatively evaluated at 20 psig for large loss-of-coolant accidents (LOCAs) (higher pressure requires less power and a smaller bhp at the reduced flow), there would be no incremental increase in the power requirement for the CS pump (there would actually be a slight decrease). Based on a review of the other loads on the EDG, no other loads increase. Therefore, the load remains adequate.

Since the loading on the EDGs resulting from uprate is bounded by the existing EDG load study, the loading on the 480-V switchgear bus is similarly bounded.

The 480-V system loading is unaffected by operation at uprate when the switchgear bus receives its power from the normal source based on the Westinghouse assessment of the EDG loads.

The 480-V bus voltage levels are potentially affected because of the increased load flow in the 6900-V system. Therefore, voltages have been estimated to show the effect on 480-V switchgear voltage levels resulting from unit operation at SPU conditions.

System Voltage Levels and Equipment Fault Duty

Two scenarios were evaluated, normal full-load operation with the main generator online, and SI conditions. The results are summarized in Tables 9.8-11 and 9.8-12, respectively.

Short circuit duty at the 480-V buses is not adversely affected by equipment load changes associated with unit operation at SPU conditions, because the load changes did not require replacement of, or changes to, existing electrical components and equipment.

440-/460-V Motors, Motor Feeder Cables, Associated Electrical Penetrations, and Overcurrent Protection

Based on the Westinghouse assessment of the peak EDG loads affected by the SPU, containment recirculation (CR) fan motor power requirements change from 223 to 227 kW for CR fans 21 and 22. This peak load occurs after recirculation switch 4 (right after the recirculation pump starts) on EDG 21, with failure of EDG 23. This is less than the peak load of 243 kW shown earlier in the time sequence, or the max load of 250 kW in the existing EDG load study. Since CR fans 21 and 22 motors will each operate at less than the existing peak or maximum kW, the existing motors are considered adequate to support the SPU. Since these are the only low-voltage motors affected by the SPU, the remaining low-voltage motors are also adequate.

Since the loading on CR fans 21 and 22 motors resulting from the SPU is bounded by the existing EDG load study, it can be concluded that the feeder cable ampacity, electrical penetrations, and overcurrent trip setpoint associated with CR fans 21 and 22 motors are adequate to support unit operation at SPU conditions.

9.8.1.8.2.2 EDG System

Three independent EDGs supply emergency power to the engineered safety feature (ESF) buses in the event of a loss of AC auxiliary power. Each EDG is started automatically on a SI signal, or upon the occurrence of an undervoltage signal on any vital 480-V switchgear bus. The system is sufficiently redundant such that any 2 diesels have adequate capacity to supply the ESFs for the DBA concurrent with a loss-of-offsite power (LOOP). One diesel is adequate to provide power for a safe and orderly shutdown in the event of a LOOP. The EDG System includes the bus duct connections up to the 480-V switchgear circuit breaker generator-side stabs. The 480-V switchgear buses and associated circuit breakers are included in the 480-V system.

EDG System Loading

Westinghouse performed an assessment of the peak EDG loads. This assessment was done by evaluating the increase in the peak load from WCAP-12655, Rev. 2. This peak load occurs after recirculation switch 4 (right after the recirculation pump starts) on EDG 21, with failure of EDG 23. Previously this load was 2268 kW (Table 5.5-2a, page 5-39 of WCAP-12655). However, the peak load is reduced to 2254 kW based upon containment analysis performed by Westinghouse at SPU conditions (see subsection 9.8.1.8.2.1 of this report). This is less than the 2300-kW half-hour limit, and therefore, will be adequate.

The containment spray (CS) pump, which is still running at this time in recirculation, operates against a slightly higher containment pressure. Since the containment back-pressure was conservatively evaluated at 20 psig for large LOCAs (higher pressure requires less power, and a smaller bhp at the reduced flow), there would be no incremental increase in the power requirement for the CS pump (there would actually be a slight decrease). Based on a review of the other loads on the EDG, no other loads increase; therefore, the load remains adequate.

Since the loading on the EDGs resulting from uprate is bounded by the existing EDG load study, the loading on the EDG bus ducts is similarly bounded.

The EDG generator ratings analysis concluded that the generators and exciters have the capability of continuous operation at 2300 kW, 480 VAC at 0.8 power factor. This conclusion and the conclusion of an EDG generator dynamic analysis is still valid for the SPU, based on the preceding load change description. However, it should be noted that the EDG steady-state output current should not exceed 3300 amps, based upon results of EDG uprate and field testing. The restriction results from temperature limitations of the associated 480-V switchgear buses. Based upon a review of load flow data at the 480-V buses included in IP2 load flow analysis of the Electrical Distribution System, the minimum power factor at the buses is

approximately 87 percent. Assuming 85-percent power factor for conservatism and a peak load of 2300 kW, EDG output current would be approximately 3255 amps. Accordingly, the 480-V buses are considered adequate to support EDG operation at SPU conditions.

The review confirmed that the EDGs and the associated 480-V switchgear buses have adequate capacity to support unit operation at SPU conditions.

9.8.1.8.2.3 118 VAC Instrument Supply Systems

There are 4 independent safety-related 118-VAC instrument supply systems serving the various instrumentation and control systems throughout the station.

Existing 118-VAC power and control schemes supplied from the system are unaffected by SPU. Similarly, no new equipment requiring 118-VAC motive or control power is expected to be added to support SPU. Consequently, operation at SPU conditions will not result in load or equipment changes in the 118-VAC system.

9.8.1.8.2.4 125-VDC Systems

There are 4 separate safety-related 125-VDC systems serving the various DC loads throughout the station.

Existing 125-VDC power and control schemes are unaffected by SPU. Similarly, no new equipment requiring 125 VDC motive or control power is expected to be added to support the SPU.

Two level control valves (LCVs) in the heater drains system (LCV-1104 and LCV-1125) will be replaced as a result of the SPU (refer to Section 3.4 of this report for a detailed description of this change). Valve replacement may result in a minor load change the 125-VDC system. Load changes, if any, will be evaluated as part of the implementing Design Change Package (DCP) Process.

Accordingly, unit operation at SPU conditions will not adversely affect the 125-VDC system.

9.8.1.8.3 Acceptance Criteria

The objective of this section is to demonstrate that the systems included herein are adequately designed to operate at the SPU power level. The systems fall into one of two categories:

- Systems that are not affected by any parameter changes associated with SPU, or are bounded by existing analysis and, therefore, are adequate for uprate operation.

- Systems that have small or reduced operating parameter changes and can be easily demonstrated as adequate.

9.8.1.8.4 Results and Conclusions

The system evaluations included in this section concluded that the following systems are not affected by the SPU or are bounded by existing analysis, and are considered adequate as installed for operation at SPU conditions:

- EDG system
- 118 VAC instrument supply systems
- 125 VDC system

The system evaluation included in this section also concluded that the 480-V system voltage levels are only marginally affected by the SPU (see Tables 9.8-11 and 9.8-12). Accordingly, the 480-V system is considered adequate to support unit operation at SPU conditions.

9.8.1.9 Grid Stability

Grid stability was reviewed to assess the transmission system impact resulting from the power uprate at IP2. The purpose of the review was to verify that the transmission system would remain stable under uprate conditions, and to determine stability issues or modifications, if any, that require resolution to support power uprate. The evaluation was based upon system studies performed by PowerGem. The studies were conducted to assess the system reliability impact of a power uprate by Entergy of the IP2 and Indian Point Unit 3 (IP3) nuclear power plants. The studies followed the New York Independent System Operator (NYISO) System Reliability Impact Study (SRIS) Criteria and Procedures. The studies evaluated two independent stretch power uprate projects—the IP2 power uprate from 1042 MW (gross) to 1078 MW and the IP3 power uprate from 1042 to 1080 MW. (Both units had previously been assessed to a conservatively high value of 1042 MW.)

9.8.1.9.1 Input Parameters and Assumptions

The review was based upon the inputs and assumptions in the following paragraphs.

Interconnection Plan

No changes to the connection of IP2 and IP3 to the bulk power system, or the impedances of the generators or generator step-up transformers, are planned as part of the uprates.

Study Period

The study period was summer 2005 and winter 2005/2006. Plant gross MW outputs, which are maximum winter values, were assumed to be the same for both seasons.

Study Area

The study analyzed the impact of the IP2 and IP3 uprates on the New York State Bulk Power Transmission System. The study concentrated on the impact on the local transmission system and the major transmission interfaces in eastern New York: Total East, Central-East, Upper New York to southeast New York (UPNY-SENY), upper New York to Con Edison (UPNY-ConEd), New York to New England (NY-NE), and New York to PJM (NY-PJM).

9.8.1.9.2 Description of Analysis

The following analyses were conducted:

- Evaluation of impact on transfer limits and transfer capability

  Analyses determined the incremental impact of the uprates on the normal and emergency transfer limits of transmission interfaces within the study area considering thermal, voltage, and stability limitations. The interfaces considered were: Central-East, Total-East, UPNY-SENY, UPNY-Con Edison, NYC Cable interface, NYISO-PJM, PJM-NYISO, NYISO-ISONE, and ISONE-NYISO. Summer and winter peak load conditions were analyzed.

- Thermal Analysis

  Thermal analyses were conducted to evaluate the impact of the uprates on the thermal transfer limits of the above interfaces, and on the Con Edison Bulk Power Transmission System in the Buchanan area, in accordance with the Con Edison design criteria. The effect of the uprates on the phase-shifted regulating lines controlling the 1000 MW wheeling contract between PSE&G and Con Edison were also evaluated.

- Voltage Analysis

  Voltage analyses were conducted to evaluate the impact of the uprates on the New York bulk power system transmission system, the Con Edison system (emphasis on the Buchanan area), in accordance with NYISO Transmission Planning Guideline No. 2, Con Edison Engineering Specification EP-7000, and the Con Edison design criteria.

- Stability Analysis

  Stability analyses were conducted to assess the stability impact of the uprates on the bulk transmission system in accordance with NYISO Transmission Planning Guideline No. 3. The stability analyses evaluated the transient stability performance of the system for normal criteria and extreme contingencies in accordance with NPCC, NYSRC, and NYISO criteria and standards. In addition, the impact of the uprates on critical clearing times of Con Edison's Substations in the area was determined.

  The system stability studies identify no new normal and extreme contingency stability performance issues.

- Short Circuit Analysis

  No changes to IP2 or IP3 generator impedances, generator step-up transformer impedances, or interconnections to the bulk power system are anticipated. Thus, there would be no effect of the uprates on short circuit contributions when calculated in accordance with the NYISO Guideline for Fault Current Assessment.

9.8.1.9.3 Acceptance Criteria

Operation under SPU conditions will not adversely affect transmission system stability or existing power system performance.

9.8.1.9.4 Results and Conclusions

The results of the analyses described above demonstrated that the IPEC uprate has no significant impact on the New York Transmission System and all applicable NYISO criteria are satisfied, and that no significant modifications are required as a result of the uprates.

Table 9.8-1 IP2 IPB Duct Loading Generator Lagging Power Factor (exporting MVARs) House Loads from UAT									
MVA	MWe	MVAR	Gen. Voltage (p.u.)	32kA BusLoad (kA)	16kA Bus to MT Load (kA)	16kA Bus to UAT Tap Load (kA)	UAT Load (MVA)	1.5kA Tap Bus Load (kA)	Notes
1253	1100	600	1.050	31.32	15.12	16.19	43.8	1.09	1, 2, 3
1253	1100	600	1.000	32.88	15.88	17.00	43.8	1.15	1, 2, 3
1253	1100	600	0.950	34.61	16.72	17.90	43.8	1.21	1, 2, 3

Notes:

1. MVA, MVAR based on review of generator capability curve at SPU.
2. UAT load for normal plant operation from Table 9.8-3.
3. Loads shown in table are approximations since they are not the result of a load flow calculation.

Table 9.8-2 IP2 IPB Duct Loading Generator Leading Power Factor, (importing MVARs) House Loads from UAT									
MVA	MWe	MVAR	Gen. Voltage (p.u.)	32kA Bus Load (kA)	16kA Bus to MT Load (kA)	16kA Bus to UAT Tap Load (kA)	UAT Load (MVA)	1.5kA Tap Bus Load (kA)	Notes
1202.2	1100	485	1.050	30.05	14.79	15.27	43.8	1.09	1, 2, 3
1202.2	1100	485	1.000	31.55	15.53	16.03	43.8	1.15	1, 2, 3
1202.2	1100	485	0.950	33.21	16.35	16.88	43.8	1.21	1, 2, 3

Notes:

1. MVA, MVAR based on review of generator capability curve at SPU.
2. UAT load for normal plant operation from Table 9.8-3.
3. Loads shown in table are approximations since they are not the result of a load flow calculation.

Table 9.8-3 UAT Load, Primary Winding - Maximum Normal Full-load Conditions			
Transformer UAT	Transformer Primary		
	MW	MVAR	MVA[1]
Existing[2]	32.48	27.99	42.88
Increase for Uprate	0.81	0.43	0.92
Total at Uprate	33.29	28.42	43.80

Notes:
1. $MVA = (MW^2 + MVAR^2)^{1/2}$
2. Existing UAT primary loading taken from IP2 load flow analysis of the Electrical Distribution System.

Table 9.8-4 MT Output Loading							
Main Generator Output[1]				MT Output Load[2]			MT Max Rated Output[3]
MW	MVAR	MVA[4]	PF, %[5]	MW	MVAR	MVA[4]	MVA
Unit Operating at Lagging Power Factor (exporting VARs)							
1100	600	1253	87.8	1065.7	399.7	1138.2	1214
Unit Operating at Leading Power Factor (importing VARs)							
1100	-100	1105	99.6	1064.7	344.6	1119.1	1214

Notes:
1. Main generator output based on generator real power at uprate and assumed normal plant reactive power requirements.
2. MT Output Load = Main Generator Output – (UAT load + Estimated Main Transformer Losses)
3. MT rated output taken from the Main Transformer Nameplate, at 65°C rise.
4. $MVA = (MW2 + MVAR2)^{1/2}$.
5. Power Factor (PF), % = $\frac{MW}{MVA} \times 100$

Table 9.8-5 UAT Output Loading				
Secondary Winding	Output Loading[1]			Maximum Nameplate Rating[3], MVA
	MW	MVAR	MVA[2]	
Existing	32.00	20.14	37.81	43.00
Incremental	0.81	0.43	0.92	
Total Secondary Load	32.81	20.57	38.73	43.00

Notes:

1. Existing UAT output loading taken from IP2 load flow analysis of the Electrical Distribution System.
2. $MVA = (MW^2 + MVAR^2)^{1/2}$
3. UAT rated output @ 55°C taken from UAT nameplate.

Table 9.8-6 SAT Output Loading				
Secondary Winding	Output Loading[1]			Maximum Nameplate Rating[3], MVA
	MW	MVAR	MVA[2]	
Existing	33.46	22.52	40.33	43.00
Incremental	0.93	0.49	1.05	
Total Secondary Load	34.39	23.01	41.38	43.00

Notes:

1. Existing UAT output loading taken from IP2 load flow analysis of the Electrical Distribution System.
2. $MVA = (MW^2 + MVAR^2)^{1/2}$
3. SAT rated output taken from SAT nameplate.

Table 9.8-7 6900-V Bus Loading							
	Bus Loading[1]				Bus Voltage V[2]	Bus Amps[4]	Breaker Rating Amps[5]
Bus	Loading	MW	MVAR	MVA[3]			
1	Existing	7.130	4.532				
	Incremental	0.205	0.107				
	Total	7.335	4.639	8.679	5989	837	1200
2	Existing	8.881	5.662				
	Incremental	0.205	0.107				
	Total	9.086	5.769	10.763	5991	1037	1200
3	Existing	8.094	4.938				
	Incremental	0.228	0.125				
	Total	8.322	5.063	9.741	5991	938	1200
4	Existing	7.883	4.883				
	Incremental	0.173	0.092				
	Total	8.056	4.975	9.468	5993	912	1200
5	Existing	16.439	10.809				
	Incremental	0.465	0.247				
	Total[6]	16.904	11.056	20.199	6547	1781	2000
6	Existing	16.995	11.303				
	Incremental	0.465	0.247				
	Total[7]	17.460	11.550	20.935	6575	1838	2000

Notes:

1. Existing bus loading taken from IP2 load flow analysis of the Electrical Distribution System.
2. Bus voltage at uprate extrapolated from IP2 Load Flow Analysis of the Electrical Distribution System based on existing load flow data and results plus incremental load changes due to uprate.
3. $MVA = (MW^2 + MVA^2)^{1/2}$
4. Bus current (amps) calculated as follows: $\frac{MVA}{\sqrt{3} \times \text{Bus Voltage}} \times 10^6$.
5. The 6900-V incoming supply breaker ratings are taken from one-line diagrams.

Table 9.8-8 6900-V Motor Feeder Breaker Loading at Uprate Conditions		
Description	**Full Load, Amps[1]**	**Breaker Rating, Amps[2]**
RCPs		
RCP21, 22, 23, 24	502/636	1200
Condensate Pumps		
CP21, 22, 23	238	1200
Heater Drain Pumps		
HD21, 22	78	1200

Notes:

1. Motor full-load amps taken from Table 9.8-10 of this report.
2. Feeder breaker ratings taken from IP2 one-line diagrams.

Table 9.8-9 6600-V Motors Affected by Uprate Conditions		
Affected Pump Motor Load	**Nameplate Rating (HP)[1]**	**Uprate Load (BHP)[2,3]**
Condensate Pumps		
CP21	3000	2660
CP22	3000	2660
Heater Drain Pumps		
HD21	1000	910
HD22	1000	910
Reactor Coolant Pumps		
RCP21	6000	5600 (Hot) 7100 (Cold)
RCP22	6000	5600 (Hot) 7100 (Cold)
RCP23	6000	5600 (Hot) 7100 (Cold)
RCP24	6000	5600 (Hot) 7100 (Cold)

Notes:

1. Nameplate rating (HP) taken from motor data sheets included in load flow calculation.
2. Uprate load bhp for BOP motors taken from bhp for condensate and heater drain pumps at 3282.5 MWt.
3. RCP motor load for hot and cold loop operation taken from the Westinghouse Engineering Report. The displayed data represents motor operation at the maximum allowed steam generator tube plug (SGTP) level of 10%. Motor evaluation included in the report concludes that the motors are adequate for use at uprate conditions while at the maximum SGTP level.

Table 9.8-10 Motor Load Current and Feeder Cable Ampacity at Uprate Conditions								
Affected Pump Motor Load	Rated HP[1]	Uprate Load BHP[2]	Efficiency[1]	Power Factor[1]	Load Flow[3]		Load Current at Uprate, Amps[4]	Cable Ampacity[5]
					MW	MVAR		
Condensate Pumps CP21 CP22 CP23	3000	2660	0.9500	0.8520	2.089	1.284	238	300
Heater Drain Pumps HD21 HD22	1000	910	0.9340	0.9050	0.727	0.342	78	125
RCPs RCP21 RCP22 RCP23 RCP24	6000	5600 (Hot)	0.9200	0.8800	4.541	2.451	502	768
RCPs RCP21 RCP22 RCP23 RCP24	6000	7100 (Cold)	0.9200	0.8800	5.757	3.107	636	768

Notes:

1. Motor-rated HP, efficiency, power factor, and full-load current data taken from motor data sheets included in load flow calculation.
2. Uprate load bhp for BOP motors taken from bhp for condensate and heater drain pumps at 3282.5 MWt analysis. RCP motor bhp data obtained from Westinghouse.
3. Load flow calculated as follows:

   MW = BHP*0.746/(efficiency*1000); MVAR = (MW/pf)*[(sin (arcos(pf))]
4. Motor full load current (amps) calculated at 90% rated voltage, as follows:

$$\frac{\sqrt{MW^2+MVAR^2}}{\sqrt{3}\times 6.6kV\times 0.9}\times 1000$$

5. Minimum feeder cable ampacity based on overload protective relay set points or pick up values.

Table 9.8-11 Estimated Voltage at 480-V Switchgear Buses (normal operation)					
	Voltage (V)				
			Delta		
Equipment	Existing	Uprate	(V)	(%)	Ref. Note
Bus 2A	393	391	2	0.50	1
Bus 3A	395	393	2	0.50	1
Bus 5A	469	469	0	0.00	1
Bus 6A	471	471	0	0.00	1

Note:

1. Existing bus voltage taken from IP2 load flow analysis of the Electrical Distribution System. Estimated voltages at uprate are extrapolated from load flow data and results included in the IP2 load flow analysis of the Electrical Distribution System, plus incremental load changes resulting from uprate.

Table 9.8-12 Estimated Voltage at 480-V Switchgear Buses (SI operation)					
	Voltage (V)				
			Delta		
Equipment	Existing	Uprate	(V)	(%)	Ref. Note
Bus 2A	438	436	2	0.46	1
Bus 3A	441	439	2	0.45	1
Bus 5A	431	429	2	0.46	1
Bus 6A	425	423	2	0.47	1

Note:

1. Existing bus voltage taken from IP2 load flow analysis of the Electrical Distribution System. Estimated voltages at uprate are extrapolated from load flow data and results included in the IP2 load flow analysis of the Electrical Distribution System, plus incremental load changes resulting from uprate.

9.9 Piping and Supports

9.9.1 Introduction

The purpose of the piping review is to evaluate piping systems for the effects resulting from stretch power uprate (SPU) conditions to demonstrate design basis compliance in accordance with ASA B31.1-1955, Code for Pressure Piping. System operation at SPU conditions may increase piping stress levels, pipe support loads, equipment nozzle loads, etc., due to slightly higher operating temperatures, pressures, and/or flow rates.

The scope of the Indian Point 2 (IP2) piping that was evaluated for SPU conditions included the following piping systems.

Steam and Power Conversion Systems

- Main steam
- Extraction steam
- Condensate
- Feedwater
- Heater drains
- Moisture separator and reheater drains
- Steam generator blowdown
- Circulating water

Auxiliary Systems

- Auxiliary feedwater
- Fuel pit cooling
- Service water

Miscellaneous Balance-of-Plant (BOP) Systems

- Auxiliary steam

9.9.2 Inputs and Assumptions

BOP Plant Walkdown Summary

To further support the acceptable evaluations completed, a plant walkdown was performed on portions of the power cycle (that is, condensate, feedwater, extraction steam, feedwater

heaters (FWHs) vents and drains, and moisture separator and reheater drains) BOP piping systems to review the piping layouts and support configurations. The purpose of the piping system walkdowns was to assess the adequacy of the installed piping deadweight spans and to review the existing thermal flexibility of the piping systems.

The portion of these BOP systems located in the Turbine Building was the focus of the walkdowns performed. The overall assessment from the walkdowns performed concluded that the existing BOP piping that was observed was adequately supported and contained adequate flexibility to accommodate the small temperature and pressure changes resulting from the SPU. Piping systems were determined to be adequately supported if the piping was supported by vertical supports, rod hangers or spring hangers, such that piping spans were consistent with the guidance presented in ASA B31.1-1955, Code for Pressure Piping. Piping systems were determined to have adequate flexibility if the following attributes were observed:

- Piping lengths and offsets were consistent with simplified industry methods of determining flexibility (for example, nomographs).
- There was no non-integral or integrally welded piping anchors installed.
- There was a sufficient and reasonable number of piping elbows installed providing thermal flexibility.

9.9.3 Description of Analysis and Evaluation

System operation at SPU conditions generally results in increased pipe stress levels and pipe support loads due to slightly higher operating temperatures, pressures, and flow rates internal to the piping. The following paragraphs describe the evaluation of the piping systems affected by the SPU conditions.

Pre-uprate and uprate system operating data (operating temperature, pressure, and flow rate) were obtained from heat balance diagrams, calculations and/or other applicable reference documents.

Change factors were determined, as required, to evaluate and compare the changes in operating conditions. The thermal, pressure, and flow rate "change factors" were based on the following ratios:

- The thermal change factor was based on the ratio of the SPU to pre-uprate operating temperature. That is, thermal change factor is $(T_{uprate}-70°F)/(T_{pre\text{-}}uprate-70°F)$.

- The pressure change factor was determined by the ratio of ($P_{uprate}/P_{pre\text{-}uprate}$).

- The flow rate change factor was determined by the ratio of ($Flow_{uprate}/Flow_{pre\text{-}uprate}$).

These thermal, pressure and flow rate change factors were used in determining the acceptability of piping systems for SPU conditions.

Based on the thermal, pressure, and flow rate change factors determined as described above, the following engineering activities were performed and/or conclusions reached.

For thermal, pressure, and flow rate change factors less than or equal to 1.0 (that is, the pre-uprate condition envelops or equals the SPU condition), the piping system was concluded to be acceptable for SPU conditions.

For thermal, pressure, and flow rate change factors greater than 1.0 through 1.05 (that is, a greater than zero and less than or equal to 5 percent increase in thermal expansion, pressure, and/or flow rate effects), this minor increase was concluded to be acceptable based on the following rationale. Certain levels of deviation from design basis conditions can be concluded to be permissible if that level of change doesn't alter the piping system results to an appreciable degree. Relatively small temperature changes can be concluded to be acceptable as the increase in pipe stresses, pipe support loads, nozzle loads, and piping displacements are correspondingly small and generally predictable. These increases are somewhat offset by conservatism in analytical methods used to calculate thermal and/or fluid transient stresses and loads. Conservatism may include the enveloping of multiple thermal operating conditions, as well as not considering pipe support gaps in thermal analyses. Also, for supports installed on safety-related systems that are evaluated for seismic loading effects, a potential 5-percent increase in a specific thermal loading condition will generally result in a less than 5-percent overall pipe support design load increase due to the existence of seismic earthquake loads.

For thermal, pressure, and flow rate change factors greater than 1.05, more detailed evaluations were performed to address the specific increase in temperature, pressure, and/or flow rate to document design basis compliance. Descriptions of the evaluations performed are provided in the following individual piping system sections.

In addition to the methodology described above, plant walkdowns were performed on portions of the BOP piping systems forming the main power cycle (that is, condensate, feedwater, extraction steam, FWHs vents, and drains). These walkdowns were performed to review the piping layouts and support configurations to assess adequacy of the deadweight spans and to review the thermal flexibility of the installed piping systems.

9.9.4 Acceptance Criteria for Analysis

Thermal, pressure, and flow rate change factors were determined to evaluate and compare changes in piping system operating parameters. In summary, thermal, pressure, and flow rate change factors less than or equal to 1.05 are acceptable and no further piping system evaluations are required. For thermal, pressure, and flow rate change factors greater than 1.05, more detailed evaluations were performed to address the specific increase in temperature, pressure, and/or flow rate to document design basis compliance.

9.9.5 Results and Conclusions

The results of these evaluations determined that piping stress levels, pipe support loads, and equipment nozzle loads associated with SPU conditions are within acceptable limits for all piping systems.

Since changes to operating temperatures, pressures, and flow rates for applicable high and moderate energy piping systems are sufficiently small and there are no new or revised pipe break locations, the existing design basis for pipe break, jet impingement, and pipe whip considerations remain acceptable for the SPU conditions.

The piping and pipe support evaluations performed concluded that all IP2 piping systems remain acceptable, and will continue to satisfy design basis requirements when considering the temperature, pressure, and flow rate effects resulting from SPU conditions.

An important element of successful operation of IP2 at SPU conditions is the monitoring and evaluation of piping vibration. Lessons learned from power uprates indicate that increased vibration of components in systems experiencing increased flowrates under uprated conditions has caused fatigue-induced failures, and that these conditions may not be readily identified during the analysis phase of the uprate project. Accordingly, in support of the SPU, piping vibration will be monitored during the power ascension to the SPU power level.

9.10 BOP Instrumentation and Controls

A review was performed on the following balance-of-plant (BOP) systems:

- Steam and Power Conversion Systems
- Auxiliary Systems
- Miscellaneous BOP Systems
- Electrical Systems

As a result of the review, it was concluded that the BOP Instrument and Controls Systems equipment will accommodate the Indian Point 2 (IP2) stretch power uprate (SPU) operation with only one minor modification replacement of the main control room digital wattmeter.

It was also determined that the changes in plant process values resulting from SPU conditions does not require re-scaling of any existing BOP instrumentation.

9.11 Area Ventilation (HVAC)

A review of Indian Point Unit 2 (IP2) area heating, ventilation, and air conditioning (HVAC) systems was performed to determine the impact of the stretch power uprate (SPU) on system operation. Systems reviewed were grouped as follows:

- Primary Auxiliary Building (PAB)/Electric Tunnel (ET)/Electrical Switchgear Room (SWGR)/Diesel Generator (DG) Building HVAC Systems
- Fuel-Handling Building (FHB) HVAC System
- Central Control Room HVAC System
- Containment Ventilation and Cooling HVAC System

9.11.1 HVAC PAB/ET/SWGR/DG Building

9.11.1.1 Introduction

The IP2 HVAC systems in the PAB, ET, SWGR, and DG Building are designed to remove heat generated from operating equipment and piping, and to maintain safe ambient operating temperatures for equipment and personnel. The HVAC systems associated with areas containing radioactive material also control airborne radioactive contamination, ensure air flow is from areas of low contamination to areas of higher contamination, provide for controlled cleanup of contaminated air, and provide for safe release to the environment.

9.11.1.2 Input Parameters and Assumptions

The primary input assumption associated with the evaluation of the HVAC systems in the PAB, ET, SWGR, and EDG Buildings was that the systems are capable of performing their required functions at the current power level. The systems were evaluated based on input parameters resulting from associated SPU operating conditions as compared with HVAC system design input parameters.

9.11.1.3 Description of Analysis and Evaluation

The IP2 HVAC systems associated with the PAB, ET, SWGR, and EDG were evaluated to determine if the existing system design is capable of performing intended functions under conditions associated with plant SPU to 3216 MWt. Expected SPU conditions were compared and evaluated against system design conditions.

The consideration of the need to perform additional analyses and/or modifications necessary to support SPU was included as part of the evaluation.

9.11.1.4 Acceptance Criteria

The overall acceptance criterion is that the PAB, ET, SWGR, and DG HVAC systems will remain capable of performing their design function under IP2 SPU operating conditions. System design parameters must bound SPU operating conditions.

9.11.1.5 Results and Conclusions

Operation at SPU conditions will not increase the heat load in the PAB above the bounding analysis level, and will not affect the potential airborne contamination in the building.

Operation under SPU conditions will not affect the heat load in the ET or operation of equipment in the ET.

Operation at SPU conditions will not increase the heat load in the electrical SWGR above the bounding analysis level (including considerations associated with the fans used to ventilate the Indian Point Unit 1 (IP1) "Water Factory"), and will not affect the potential airborne contamination in the building.

Operation at SPU conditions will not affect the heat load or operation of equipment in the EDG Building.

Based on the results discussed above, the impact of SPU operating conditions on the PAB, ET, SWGR, and DG HVAC systems will not adversely affect the operational ability of these systems. The systems will function as designed under SPU conditions without limitation. No plant modifications to the PAB, ET, SWGR, and DG HVAC systems are required to support SPU.

9.11.2 HVAC FHB

9.11.2.1 Introduction

The primary function of the HVAC system in the FHB is to provide ventilation air to remove heat and moisture buildup generated from spent fuel decay heat and from operating equipment, and to maintain safe ambient operating temperatures for equipment and personnel. The secondary function of the system is to remove potential airborne radioactive contamination from the area during an accident and provide for controlled cleanup of contaminated air for safe release to the environment.

The IP2 HVAC system in the FHB is evaluated to determine if existing system designs are capable of performing its intended functions subsequent to an SPU. Included as part of the evaluation is consideration of the need to perform additional analyses and/or modifications necessary to support such an uprate.

9.11.2.2 Input Parameters and Assumptions

The primary input assumption associated with the evaluation of the HVAC system in the FHB is that the system is capable of performing its required functions at the current power level. The system is evaluated based on input parameters resulting from associated SPU operating conditions as compared with HVAC system design input parameters.

9.11.2.3 Description of Analysis and Evaluation

The FHB HVAC system was evaluated to determine if existing system design is capable of performing intended functions under conditions associated with plant SPU to 3216 MWt. Expected SPU conditions were compared and evaluated against system design conditions.

Included as part of the evaluation is consideration of the need to perform additional analyses and/or modifications necessary to support such an uprate.

9.11.2.4 Acceptance Criteria

The overall acceptance criterion is that the FHB HVAC system will remain capable of performing its design function IP2 SPU operating conditions. System design parameters must bound SPU operating conditions.

9.11.2.5 Results and Conclusions

Operation at SPU conditions will not increase the heat load in the FHB above the bounding analysis (that is, analysis bounds SPU operating conditions). Fuel decay heat will increase slightly as a result of SPU operation, but the normal spent fuel pit (SFP) temperature will not be affected by this slight increase.

Based on the results the evaluation, the impact of SPU on the FHB HVAC system does not adversely affect the operational ability of the system. The FHB HVAC system will function as designed under SPU conditions without limitation. No plant modifications to the FHB HVAC system is required to support SPU.

9.11.3 HVAC Central Control Room

9.11.3.1 Introduction

The IP2 central control room (CCR) HVAC system is designed to provide the following functions:

- Maintain the required design temperature and relative humidity inside the CCR during all modes of plant operation.
- Isolate the CCR to prevent infiltration of toxic gases and smoke, and cleanup of airborne radioactive particulates in the outdoor air entering the CCR during high radiation and/or safety injection conditions.
- Provide slight positive pressure in the CCR during normal and high radiation or safety injection modes of operation to prevent in-leakage of airborne contamination from adjoining space.

The IP2 CCR HVAC system was evaluated to determine if the existing system design was capable of performing its intended functions subsequent to a plant SPU to 3216 MWt. Included as part of the evaluation was the consideration of the need to perform additional analyses and/or modifications necessary to support such an uprate.

9.11.3.2 Input Parameters and Assumptions

The primary input assumption associated with the evaluation of the CCR HVAC system is that the system is capable of performing its required functions at the current power level. The system was evaluated based on input parameters resulting from associated SPU operating conditions as compared with HVAC system design input parameters.

9.11.3.3 Description of Analysis and Evaluations

The IP2 CCR HVAC system was evaluated to determine if the existing system design is capable of performing intended functions under conditions associated with plant SPU to 3216 MWt. Expected SPU conditions were compared and evaluated against system design conditions.

Included as part of the evaluation was the consideration of the need to perform additional analyses and/or modifications necessary to support such an uprate.

9.11.3.4 Acceptance Criteria

The overall acceptance criterion is that the IP2 CCR HVAC system will remain capable of performing its design function under SPU operating conditions. System design parameters must bound SPU operating conditions.

Note that the CCR HVAC system interfaces with the plant Auxiliary Steam System, which provides steam to the air handling unit heating coil during the heating season. The Auxiliary Steam System is a non-safety related system. The acceptance criteria for the plant Auxiliary Steam System operation under SPU operating conditions is addressed in Section 9.13.

The potential radiological exposure to the operators under post-accident conditions is addressed by the accident analyses. The loss-of-coolant accident (LOCA) analysis assumes an SPU power level of 3216 MWt. The current analysis must bound the SPU conditions.

9.11.3.5 Results and Conclusions

Based on the results discussed above, the impact of SPU on the CCR HVAC system does not adversely affect the operational ability of the system. The system will function as designed under SPU conditions without limitation. No plant modifications to the CCR HVAC system is required to support SPU.

9.11.4 HVAC – Containment Ventilation and Cooling

9.11.4.1 Introduction

The Containment Heating, Cooling, and Ventilation System is designed to accomplish the following functions:

- Remove of normal heat loss from equipment and piping to ensure a maximum ambient temperature of 130°F is not exceeded.
- Provide positive circulation of air across the refueling water surface to ensure personnel access and safety during shutdown.
- Provide containment heating to maintain a minimum containment temperature of 50°F before the reactor is taken above the cold shutdown condition.

- Provide for purging of the containment vessel to the plant vent for dispersion to the environment.

- Provide depressurization of the containment vessel following an accident.

- Provide continuous pressure relief via an exhaust system.

The above functions are accomplished in conjunction with the following subsystems:

- Containment Recirculation Cooling System
- Control Rod Drive Mechanism (CRDM) Cooling System
- Containment Purge and Pressure Relief System

9.11.4.2 Input Parameters and Assumptions

The primary input assumption associated with the evaluation of the Containment Heating, Cooling, and Ventilation System is that the system (including all subsystems) is capable of performing its required functions at the current power level. The system is evaluated based on input parameters resulting from associated SPU operating conditions as compared with HVAC system design input parameters.

9.11.4.3 Description of Analysis and Evaluation

The IP2 Containment Heating, Cooling, and Ventilation System (and associated subsystems) was evaluated to determine if the existing system design is capable of performing intended functions under conditions associated with plant SPU to 3216 MWt. Expected SPU conditions were compared and evaluated against system design conditions.

Included as part of the evaluation is consideration of the need to perform additional analyses and/or modifications necessary to support such an uprate.

9.11.4.4 Acceptance Criteria

The overall acceptance criterion is that the Containment Heating, Cooling, and Ventilation System (and associated subsystems) will remain capable of performing their design functions under power uprate operating conditions. System design parameters must bound SPU operating conditions.

9.11.4.5 Results and Conclusions

Results of the SPU evaluation in conjunction with subsystems making up the Containment Heating, Cooling, and Ventilation System is provided below. Based on these results, the impact of SPU on the Containment Heating, Cooling, and Ventilation System does not adversely affect the operational ability of the system and associated subsystems. The system and associated subsystems will function as designed under SPU conditions without limitation. No plant modifications to the CCR HVAC system is required to support SPU.

Containment Recirculation Cooling System

The Containment Recirculation Cooling System and associated filtration systems maintain ambient containment temperature at or below 130°F, remove heat from containment following an accident, and clean up post-accident containment atmosphere. During the normal mode operation under SPU conditions, the containment heat load will increase slightly. However, the fan cooling units (FCUs) in conjunction with station operating procedures will remain adequate for normal operation to maintain containment temperature below 130°F.

For post-accident conditions, the FCU cooling capacity performance was evaluated as a part of the accident analysis. The capacity to remove fission products from the containment atmosphere after an accident was also evaluated as part of the accident analysis. Under SPU conditions, the system design remains bounding.

Control Rod Drive Mechanism Cooling System

The CRDM Cooling System is designed to maintain the control rod drive operating coils stacks at or below their maximum operating temperature of 200°F. Operation under SPU conditions will not significantly affect heat loads or temperature associated with the CRDM. The CRDM Cooling System will continue to meet system functional requirements under SPU operating conditions.

Containment Purge and Pressure Relief System

Operation under SPU conditions will not affect operation of the Containment Purge and Pressure Relief System. The containment purge and make up capability is not impacted by SPU, and operation under SPU conditions will not affect pressure build up in Containment during reactor power operation, or the operation of the Containment Pressure Relief System. The Containment Purge and Pressure Relief System will continue to meet system functional requirements under SPU operating conditions.

9.12 Auxiliary Feedwater System

9.12.1 Introduction

The Auxiliary Feedwater System (AFWS) is designed to provide emergency cooling for the reactor by supplying water to the steam generators. The Feedwater System (FWS) provides water to the steam generators during power operation while the AFWS is used at low power when the steam is not available to operate the main feedwater system. The AFWS also operates under the following conditions:

- Loss of main feedwater
- Rupture of a main steam line
- Loss-of-coolant accident (LOCA)
- Loss-of-AC power (LOAC)
- Steam generator tube rupture (SGTR)
- Anticipated transient without scram (ATWS)
- Alternate safe shutdown
- Station blackout (SBO)

The AFWS also provides feedwater to the steam generators to support the ability to cool the RCS to the point at which the Residual Heat Removal System (RHRS) may be brought online to complete the cooldown process (during normal-operation or post-accident scenarios).

AFW is supplied by the actuation of 2 motor-driven AFW pumps (MDAFWPs), which are initiated by any of the following signals.

- Low-low water level in any steam generator
- Automatic trip (not manual) of any main feedwater pump turbine
- Any safety injection (SI) signal
- Manual actuation
- Loss-of-offsite power (LOOP) concurrent with unit trip

In addition, 1 turbine-driven AFW pump (TDAFWP) starts on any of the following actuation signals, although no automatic delivery of water to the steam generators occurs (the TDAFWP is automatically started, but must be manually aligned by the operator to allow delivery of AFW flow to the steam generators).

- Low-low water level in any 2 steam generators
- LOOP concurrent with unit trip and no safety injection signal
- Manual actuation

The MDAFWPs are powered by the emergency diesel generators (EDGs). The pumps take suction from the condensate storage tank (CST) for delivery to the steam generators. Each MDAFWP is designed to supply the minimum required flow within 60 seconds of the initiating signal. The TDAFWP is valved-out during normal operation. Therefore, although the TDAFWP is automatically actuated, this pump is not available to deliver flow to the steam generators until operator action is taken to align the TDAFWP.

The worst single failure modeled in the limiting non-LOCA analyses (see Sections 6.3.7 and 6.3.8) was the loss of 1 of the 2 MDAFWPs. This resulted in the availability of only 1 MDAFWP automatically supplying a minimum total AFW flow of 380 gpm, distributed equally between 2 of the 4 steam generators. Additional flow from the second MDAFWP or TDAFWP was assumed to be available only following operator action to start the second MDAFWP or to align the TDAFWP. This operator action was assumed to provide an additional 380 gpm of AFW flow distributed equally to the other 2 steam generators not receiving AFW automatically, and was assumed to occur at 10 minutes after the reactor trip due to a low-low steam generator water level signal.

The AFWS consists of 2 distinct safety-grade subsystems (that is, 2 pumping systems using different sources of motive power for their pumps) to ensure reliability of the feedwater supply. The plant original design consisted of a subsystem with 2 trains, each with a 100-percent capacity, MDAFWP designed to deliver flow to 2 of the 4 steam generators. The second subsystem consisted of a 200-percent capacity, TDAFWP designed to deliver flow to all 4 steam generators.

There are 2 independent water supplies available to the AFWS. These 2 sources are configured so that there are 2 redundant suction flow paths to each AFW pump. One flow path is a single line from the CST, and the second flow path is a single line from the city water storage tank. Only 1 source is aligned to the pumps at one time.

9.12.2 Input Parameters

The required AFWS flow and capacity are proportional to the amount of decay heat that must be removed from the core during accident conditions. The AFWS functions associated with normal plant startup and shutdown are not dependent on core power, and therefore are not affected by SPU.

The following AFWS parameters are not impacted by SPU and will remain unchanged:

- Maximum AFWS flowrate for steam line break
- AFW flow from each MDAFWP

- AFW flow from the TDAFWP
- Maximum AFW temperature (that is, 120.0°F).

The limiting transient with respect to CST inventory is the LOAC to station auxiliaries transient. IP2 licensing basis dictates that in the event of a LOAC as described in subsection 6.3.8 of this report, sufficient CST inventory must be available to bring the unit from full power to hot standby conditions, and maintain the plant in hot standby for 24 hours. The SPU CST minimum useable volume requirement is 291,381 gallons.

9.12.3 Description of Analysis and Evaluation

Evaluation of the AFWS consists of documenting the current system functional requirements for transients/accidents and the extent to which SPU impacts these AFWS functions.

This evaluation compared AFWS component and equipment pressure and temperature design with the SPU pressure/temperature associated with AFWS operating conditions (that is, AFWS functions associated with normal plant startup and shutdown).

The evaluation also considered the extent to which sufficient AFW flow is provided to the steam generators following a design basis accident (DBA), and the extent to which adequate water inventory is available in the CST to satisfy AFWS functional requirements. The limiting transient (that is, design basis) with respect to CST inventory is LOAC as described in subsection 6.3.8 of this document.

9.12.4 Acceptance Criteria

The AFWS is considered acceptable under SPU conditions provided the following conditions are met:

- AFWS piping and component pressure and temperature design bounds pressure and temperature conditions under SPU off-normal operation (see subsection 9.12.1).

- Sufficient AFW flow is provided to the steam generators following a DBA and AFW pump operation is within acceptable margin of pump design parameters (for example, flow and total discharge head [TDH]).

- Based on the limiting transient (that is, LOAC as described in subsection 6.3.8 of this report) design basis, sufficient 120°F AFW inventory is available to maintain the IP2 plant in hot standby for 24 hours following a reactor trip from full power.

9.12.5 Results and Conclusions

The volume of water contained in the IP2 CST is adequate to support SPU.

AFWS component and equipment pressure and temperature design bounds maximum pressure and temperature conditions expected under SPU operation. AFWS components and equipment are considered acceptable for SPU operation.

The requirement to remove heat from steam generators under transient and accident conditions is the basis for AFWS minimum flow requirement. Currently, in the event of a loss of normal feedwater or a LOAC to station auxiliaries, a minimum flow of 380 gpm is assumed to 2 of the steam generators as a result of automatic actuation of 1 motor-driven AFW pump due to a low-low steam generator water level trip signal. Under SPU conditions an additional minimum AFW flow of 380 gpm, split evenly between the 2 remaining steam generators is required, and can be provided as a result of operator action taken to start the remaining motor-driven AFW pump or steam-driven AFW pump.

The AFWS is acceptable for operation under SPU conditions. No system modifications are required.

9.13 Structural Analysis

9.13.1 Fuel-Handling Building Structural Analysis

The stretch power uprate (SPU) for Indian Point Unit 2 (IP2) results in fuel with increased radioactivity in the fuel assemblies being transferred from the reactor to the pit during refueling. The addition of a non-uprate-related effect (the removal of fuel sooner from the reactor after shutdown, that is, 84 hours instead of 100 hours after shutdown), also results in an increased level of radioactivity. The combination of these two conditions may result in the heating of the concrete pit structure by the gamma radiation emanating from the fuel, in the event the fuel is placed in fuel rack cells adjacent to the concrete walls.

The pit water temperature will be maintained within the limits defined in the system description for the spent fuel pit cooling loop. With the conservative assumption that as fuel is offloaded it will be immediately placed adjacent to the concrete walls of the pit structure, with no older spent fuel between it and the concrete, bounding estimates of the concrete temperature effects can be made. The gamma heating is limited to the lower 13 feet of the exterior walls which are 4-feet thick and in contact with the rock or soil backfill on the outside face and the 5-feet thick interior fuel transfer canal wall. This 13-feet height comprises the active fuel length, 12 feet, and an additional foot, to account for the floor of the rack and the lower end of the fuel bundle.

This section describes the analysis made to address the effects of gamma heating on the concrete structure.

9.13.1.1 Input Parameters and Assumptions

The input parameters and the assumptions used in the evaluation of the gamma-heated concrete pit structure are summarized in the following paragraphs.

- The reference temperature, T_{ref}, for the concrete, that is the temperature at which no thermal expansion or contraction occurs, is 70°F.

- The maximum expected pit water temperature is 120°F during fuel offloading. At a temperature of 125°F, the spent fuel pit temperature alarm is activated and additional supplemental cooling will be provided to maintain the temperature at or below 120°F and continue the refueling.

- The loads induced by the gamma-heating gradient is included as part of the temperature effects associated with loading combinations.

- The active fuel length in a fuel bundle is 12 feet and this defines the height of the wall above the mat, 13 feet, that is heated by the gamma radiation.

- The non-linear thermal gradient due to gamma heating can be decomposed into a uniform thermal expansion across the section and a linear gradient across the section, producing an equivalent compression and tension and an equivalent bending moment as the non-linear gradient.

- The volume of the concrete wall affected by gamma heating is approximately equal to the height of the fuel bundle, 13 feet, measured from the fuel pit floor liner. The thickness of the gamma heated volume is defined by the gradients calculated for the IP2 spent fuel pit. The width of the gamma-heated zone resulting from the radiation from a single fuel bundle is equal to the width of the bundle.

- There is no reduction in the gradient through the wall due to propagation of the heat, away from the gamma-heated volume, obliquely through the wall.

- The 180°F thermal gradient applies to the entire height of the wall and is not limited to the upper portion exposed to the ambient environment.

- The soil backfill or rock adjacent to the below-grade portions of the spent fuel pit walls is considered to restrain the pit structure. The analysis of the pit structure implies that the rock constrains/supports the pit structure. The backfill is usually neglected when considering the effect of hydro-static and hydro-dynamic pit water pressure loading on the walls and mat. Both conditions are addressed in the analysis.

- The temperature of the soil in contact with the fuel pit mat and lower part of the walls is 50°F.

- The design basis gradient for the 5-feet thick wall is 60°F based on a 180°F pit side face temperature, and 120°F refueling canal-side face temperature.

- The entire fuel pit is assumed to be exposed to "fresh" fuel, therefore conservative allowables were used. The more realistic case, were only specified locations will see fresh fuel, results in local effects that have higher allowables.

9.13.1.2 Description of Analysis and Evaluation

The non-linear thermal gradient arising from the gamma heating of the concrete structure is converted to a linear gradient producing an equivalent compression and bending moment. The

conversion requires the integration of the nonlinear gradient relative to reference temperature, 70°F. This reference temperature is applicable to each face of the concrete walls. The loads associated with the thermal gradient due to gamma heating, measured from the reference temperature, is the temperature loading to be factored and added to the dead and seismic loads in the loading combination.

The gamma heating gradients in the concrete walls and mat are developed assuming a pit water temperature of 120°F at the start of refueling. The equivalent linear gradient for gamma heating is compared to the design basis gradient. The design basis gradient is subtracted from the equivalent linear gradient due to gamma heating. The reinforcing stresses associated with this net gradient are determined and considered as incremental stresses.

In accordance with the assumptions of Section 9.13.1.1 the results of the gamma heating analysis are evaluated assuming either no support from the wall backfill or full support from the backfill or rock. Assuming no support from the backfill or rock, the sign or sense of the stresses created by gamma heating is compared to those generated by the hydrostatic and hydrodynamic loadings and the resulting rebar stresses are added or subtracted as appropriate. In general, the gamma heating stresses act in the opposite sense to the stresses due to the dead and seismic loads. Where the stresses act in the same sense, the incremental stresses are less than the available margins.

The alternative approach assumes the full restraint of the exterior pit walls is provided by the surrounding rock and/or backfill. This is not applicable to the interior 5-feet thick wall.

- There are no (or low) stresses in the wall reinforcement due to hydrostatic or hydrodynamic loads since these forces acting out of the plane of the wall are transmitted directly to the rock.

- The only stresses in the reinforcement bars are due to the thermal gradient associated with normal operation and gamma heating of the concrete during refueling.

- The confining compressive stress in the wall occurs in the horizontal direction only, since the wall is free to grow vertically due to thermal expansion. This is also valid for the interior 5-feet thick wall.

- Assuming that the entire length of the wall is subjected to gamma heating, the resulting thermal compressive strain due to gamma heating results from the equivalent temperature increase which is calculated to produce compression equal to that of the non-linear gamma heating thermal gradient.

The in-plane compression strain resulting in the horizontal direction due to the confinement of the surrounding rock is equal to $\alpha\Delta T$, where the equivalent uniform temperature increase is estimated to be 88.5°F.

9.13.1.3 Acceptance Criteria

The allowable reinforcement stress is based on the limits in the Building Code Requirements for Reinforced Concrete. These limits are 0.9 times the reinforcement yield stress, 60,000 psi, or 54,000 psi.

The peak temperatures are conservative estimates with a low likelihood of occurrence. These are recognized as conservative values and not "best estimates" for the following reasons:

- Refueling is assumed to begin at the maximum spent fuel pit operating temperature as opposed to more realistic temperatures on the order of 100°F. The pit water temperature will depend on the river water temperature and the upper bound limit for the river water heat sink is chosen to ensure that the pit water cooling system is properly sized. Under the most likely conditions the pit water temperature will remain near 100°F.

- The gamma heating analysis and heat transfer analysis is one-dimensional (1-D) and neglects the ability of the reinforcement to distribute the heat laterally and vertically away from the gamma-heated zone where the reinforcement is concentrated on the inside face of the walls. The reinforcement lies in the zone with the maximum gamma heating, but is not considered.

- The assumptions made regarding the placement of the off-loaded fuel are intended to produce the maximum concrete temperature.

- The parameters chosen to characterize the fuel will maximize the gamma radiation (radial peaking factor).

Consequently, the occurrence of these peak temperatures is considered an unlikely event with and, therefore, abnormal.

The mechanisms associated with the weakening of concrete due to elevated temperatures are mitigated by the specific conditions associated with the spent fuel pit structure. It is a below grade structure with the exterior walls exposed to a moist environment, the area corresponding to the gamma heated concrete. If a water proofing membrane is present on the outside face the moisture migration will be further restricted. The liner ensures that ambient air cannot readily desiccate the concrete and limits the moisture migration typically accompanying heating. The

large wall thickness slow the water vapor migration in concrete making it less susceptible to loss of water. Moisture recovery following the cessation of gamma heating, driven by the partial pressure differential of water vapor in concrete pores, occurs to the extent that significant differences in the partial pressure exists.

For the reasons cited, the elevated temperatures reported in the results and conclusions are deemed acceptable as abnormal (low likelihood of occurrence) thermal loads.

9.13.1.4 Results and Conclusions

The analysis has demonstrated that the incremental reinforcement stresses associated with the gamma heating are within the available margins cited in the design basis and the maximum stresses in the bars are within the limits established by the governing design criteria. The compressive strain in the concrete assumed to arise due to the confinement of the rock or backfill is approximately 1/6 the value typically used to represent the ultimate strain. This compressive strain is acceptable.

The elevated temperatures in the zone of the gamma-heated concrete are deemed acceptable since these temperatures are considered abnormal loads.

9.13.2 AFW Building Structural Analysis

One possible consequence of SPU is an increase to the outside containment compartment differential pressures due to high-energy line break (HELB). The compartment differential pressure due to HELB is addressed for the Auxiliary Boiler Feed Pump Building, in this section of the report. The Auxiliary Boiler Feed Pump Building includes the "shield wall area" consisting of the steam and feedline penetration area, auxiliary feed pump room, and the blower room.

A main steam line break or feedline break are the sources of the postulated accident differential pressure challenging the capacity of the structure, since the smaller breaks do not produce significant differential pressure. Since the postulated break in the auxiliary feed pump room is the double ended rupture of a 4-inch diameter steam line to the auxiliary boiler feed pump turbine, the resulting HELB pressure in this compartment is small.

9.13.2.1 Input Parameters and Assumptions

The input parameters for this evaluation are the HELB differential pressure transients in the outside containment compartments for SPU. Also, the compartment differential pressures for the current licensed power levels are provided in the plant evaluation of harsh environment areas.

The assumption applicable to this section is that the SPU does not result in changes to the locations of existing, postulated pipe break locations or to the type of break.

9.13.2.2 Description of Analysis and Evaluation

The outside containment HELB pressure due to SPU conditions will be compared to the design pressure capacity or the HELB differential pressure for current licensed thermal power conditions for each of the compartments in the Auxiliary Boiler Feed Pump Building. The structural pressure withstand capacity is reviewed to support the conclusions that the SPU does not govern the compartment design for pressurization. For the steam and feedline penetration area, the sheet metal siding is calculated to commence failure at a differential pressure of 0.46 psig and is completely failed at a pressure of 1.26 psig.

9.13.2.3 Acceptance Criteria

The acceptance criteria for the auxiliary feed pump room and the blower room are the current licensed thermal power HELB differential pressures for each cubicle or the differential pressure used as the design basis for the structure. Pressures below these values are deemed to meet the acceptance criteria.

Since the enclosure of the steam and feedline penetration area of the Auxiliary Boiler Feed Pump Building is assumed to fail at a differential pressure exceeding 0.46 psig, there are no acceptance criteria for this area. Failure of the sheet metal siding is acceptable.

9.13.2.4 Results and Conclusions

The SPU does not result in HELB pressurization exceeding the structural capacity of the affected compartments therefore the structural capacity of the affected compartments is acceptable under SPU conditions.

9.13.3 Miscellaneous Structures

9.13.3.1 Structural Analysis

The SPU project does not affect the Primary Auxiliary Building (PAB). Outside containment HELB is the only PAB structural issue affected by power uprate and no changes result from HELB.

9.13.3.2 Turbine Building Structural Analysis

The SPU project does not affect the Turbine Building. Outside containment HELB is the only Turbine Building structural issue affected by power uprate, and no changes result from HELB.

10.0 GENERIC ISSUES AND PROGRAMS

The Indian Point Unit 2 (IP2) Stretch Power Uprate (SPU) Program has the potential to affect plant programs and generic issues that have been developed and implemented at IP2 in compliance with various design, maintenance, and licensing requirements. The plant programs and generic issues listed in Table 10.0-1 were identified for review and evaluation of the effect of the SPU.

For the programs and generic issues listed in Table 10.0-1, a review of the documentation was performed and discussions with cognizant station personnel were conducted. Based upon review of this information, the effect of implementation of the SPU on the program and generic issue was determined.

Table 10.0-1 identifies if a program/generic issue is either "affected/potentially affected" or "not affected" by the SPU. Programs/generic issues are "not affected" by the SPU if:

- The SPU does not affect key inputs to the program/generic issue, or
- The program/generic issue is based on information/parameters that bound the conditions that will result from implementation of the SPU, or
- Existing program requirements, procedures, or activities will be utilized or applied in support of implementation of the SPU.

Table 10.0-1 Effect of SPU on Generic Issues and Programs			
Section	Program/Generic Issue	Not Affected	Affected/Potentially Affected
10.1	Fire Protection (10CFR50 Appendix R) Program	X	
10.2	Generic Letter 89-10 Motor-Operated Valve (MOV) Program		X
10.3	Flow-Accelerated Corrosion (FAC) Program		X
10.4	Flooding	X	
10.5	Probabilistic Safety Assessment (PSA)		X
10.6	Station Blackout (SBO)	X	
10.7	In-service Inspection (ISI)/In-service Test (IST) Programs	X	
10.8	Electrical Equipment Environmental Qualification (EQ) Program		X
10.9	Chemistry Program	X	
10.10	Generic Letter 95-07	X	
10.11	Generic Letter 96-06	X	
10.12	Generic Letter 89-13	X	
10.13	IP2 Plant Simulator		X
10.14	Containment Leak Rate Testing	X	
10.15	Plant Operations		X

10.1 Fire Protection (10CFR50 Appendix R) Program

The Indian Point Unit 2 (IP2) Fire Protection Program Plan (FPPP) has been evaluated against the criteria of Appendix A to Branch Technical Position 9.5-1 (Reference 1) and 10CFR50 Appendix R (Reference 2).

The FPPP describes the Shutdown Model that is relied upon for Appendix R Fire Protection reviews, and provides a description of the Alternate Safe Shutdown System (ASSS) installed at IP2. The general approach used in the Appendix R shutdown model was to rely on a set of equipment powered from the ASSS as one shutdown path, with its counterpart being similar components powered from IP2 sources. The ASSS is designed to provide required shutdown functions with a loss of normal offsite power, as well as loss of all emergency diesel generators (EDGs) (for example, due to fire). The initial blackout (loss of all AC power) could last until one of the gas turbines is started and connected to the bus powering the ASSS.

In order to determine the time period in which the operator would have to establish a supply of feedwater for the secondary side of the steam generators in order to avoid boiling the steam generators dry, a "steam generator dryout time" is determined. For the SPU, the steam generator dryout time provides adequate time for the operator to supply feedwater to the secondary side of the steam generator.

The Appendix R plant cooldown analysis under stretch power uprate (SPU) conditions shows that IP2 complies with the Appendix R requirement that cold shutdown be achieved within 72 hours after reactor trip following a fire.

References

1. Appendix A to BTP 9.5-1, *Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976.*

2. 10CFR50, Appendix R, *Fire Protection Program for Nuclear Power Facility Operating Prior to January 1, 1979,* February 17, 1981.

10.2 Generic Letter 89-10 Motor-Operated Valve Program

Generic Letter (GL) 89-10 (Reference 1) requires that safety-related motor-operated valves (MOVs) be analyzed and controlled to ensure they are capable of performing their required functions. Indian Point Unit 2 (IP2) has established a GL 89-10 MOV Program. The IP2 GL 89-10 MOV Program screening evaluation defines the criteria for performing the screening to determine which MOVs should be included in, or excluded from, the scope of the IP2 MOV GL 89-10 Program, documents the results of the screening process, and identifies all MOVs included in the GL 89-10 Program.

An operational database was used to consolidate and document MOV operating conditions/parameters required as inputs for calculations for MOVs in the GL 89-10 Program. The operating parameters in this database are taken from the differential pressure calculations performed for the MOVs in the program. The following operating conditions/parameters are determined/reported in the MOV differential pressure calculations:

- Maximum open and close differential pressure
- Maximum open and close line pressure
- Flow rate
- Fluid
- Fluid temperature

The results of the MOV differential pressure calculations are used as inputs in other GL 89-10 Program MOV calculations, for example, analysis of MOV thrust and torque limits.

In support of the evaluation of the effects of high ambient temperature on AC motor performance, the MOV differential pressure calculations identify the worst case maximum ambient air temperature at the MOV location. The MOV differential pressure calculations reference the Electrical Equipment Environmental Qualification (EQ) Program for this data.

MOVs are scheduled for periodic verification of actuator settings in accordance with IP2 commitments to the Nuclear Regulatory Commission (NRC) with respect to GL 96-05 (Reference 2).

The following is a summary of the effect of the SPU on the above-listed operating parameters determined/reported in the MOV differential pressure calculations for GL 89-10 Program MOVs in balance-of-plant (BOP) systems:

- Feedwater System MOVs (feedwater pump discharge isolation valves): MOV flowrate increases due to the SPU (refer to Section 9.4). Other MOV operating parameters are not affected.

For MOVs in Nuclear Steam Supply Systems (NSSSs) (that is, Reactor Coolant System [RCS], Chemical and Volume Control System [CVCS], Residual Heat Removal System [RHRS], Component Cooling Water System [CCWS], Safety Injection System [SIS], Primary Sampling System, and Containment Spray System [CSS]), the changes in system flows, pressures, and temperatures resulting from the SPU have been documented. There are no changes in pressures and flows for the NSSS systems that affect the conclusions of the MOV Program for the NSSS MOVs.

The effect of MOV operating parameter changes on related GL 89-10 parameters (for example, valve dynamic thrust values) has been evaluated and determined to be acceptable.

The effect of the SPU on environmental conditions (that is, ambient temperatures) in plant locations containing environmentally qualified equipment has been determined. The environmental data review determined that the changes in maximum ambient temperatures at MOVs locations in the GL 89-10 Program are acceptable.

The analysis of a steamline break inside containment under SPU conditions takes credit for operation of the feedwater flow control valve isolation MOVs. These valves were not previously credited in a safety analysis and are not currently included in the GL 89-10 Program. Accordingly, they will be added to the GL 89-10 Program.

The SPU does not effect the schedule for periodic verification of MOV settings per GL 96-05.

References

1. NRC Generic Letter 89-10, *Safety-Related Motor Operated Valve Testing and Surveillance*, June 28, 1989, and supplements.

2. NRC Generic Letter 96-05, *Periodic Verification of Design Basis Capability of Safety-Related Motor-Operated Valves*, September 18, 1996.

10.3 Flow-Accelerated Corrosion Program

The primary objective of the IP2 Flow-Accelerated Corrosion (FAC) Program, is to maintain the long-term process of FAC detection and monitoring in piping systems so that pipe wall thinning can be mitigated to prevent pipe failures.

Systems/subsystems that have been identified as being susceptible to FAC at IP2 include the following:

- Main steam and steam traps
- Condensate and feedwater pump suction
- Feedwater (boiler feed)
- Extraction steam and steam traps
- Heater drains and vents
- Moisture separator and reheater drains and vents
- Feedwater pump turbine steam lines, drains, and vents
- Steam Supply and Condensate Return System
- Superheater building service boilers
- Steam Generator Blowdown and Blowdown Sample System
- Auxiliary Steam Supply and Condensate Return System
- Moisture separator reheaters vent chamber discharge
- Moisture Preseparator System

The specific lines/fittings in these systems that are included in the FAC Program (FACP) are identified on drawings.

The FACP was established to consolidate information and plans concerning wet steam corrosion issues into a single umbrella document.

The SPU will result in changes in fluid flow velocities and temperatures in the Main Feedwater and Condensate System, Heater Drain System, Main Steam System (MSS), Extraction Steam System, and Steam Generator Blowdown System (SGBS). Evaluations of the effect of the SPU on FAC for the piping in these systems were performed. The following are the key elements of these evaluations:

- Calculation and documentation of piping velocities for lines and equipment nozzles in the system, including lines in the IP2 FACP. Except for drains from the moisture pre-separators, piping velocities under SPU conditions in drain lines were calculated as single-phase (water) flow.

- Comparison of the calculated piping/nozzle velocities with standard industry velocity criteria as a measure of whether there was an increased potential for FAC.

- Evaluation of any effect of calculated operating temperatures under SPU conditions on FAC in pipelines and nozzles.

Major results and conclusions from these evaluations are summarized as follows (details are included in Sections 9.1, 9.2, 9.3, 9.4, and 9.5):

- The majority of piping and nozzle velocities under SPU conditions are within the standard industry criteria. Many of these lines are included in the IP2 FACP.

- Most of the pipelines and nozzles that had velocities which exceed the standard industry criteria are included in the IP2 FACP or have been removed from the FACP due to piping material upgrade.

- Based on review of flow velocities and operating temperatures, certain pipelines and nozzles in the condensate and extraction steam systems are being added to the IP2 FACP as appropriate.

10.4 Flooding

10.4.1 Flooding Outside Containment

IP2 identified potential sources of flooding outside containment that could affect safety-related equipment. The sources of potential flooding included the following non-seismic Class I designed systems/equipment:

- Circulating Water System (CWS)
- Fire Protection System
- Refueling water storage tank (RWST)
- Spent fuel pit (SFP) cooling loop
- Sampling System
- Chemical Volume and Control System (CVCS)
- Primary Water Makeup System
- Feedwater System (FWS)

The areas which were subject to flooding from failure of non-seismic Class I designed systems were identified as:

- Diesel Generator Building
- Fuel Storage Building
- Service water pump area
- Control Building
- Turbine Hall
- Primary Auxiliary Building (PAB)
- Auxiliary feedwater pump room

The following measures were identified in support of minimizing vulnerability to flooding: installation of level alarm switches in the Indian Point Unit 1 (IP1) condenser pit, installing a flap panel in the PAB door to the transformer yard, installing a flap panel in the door of the auxiliary feedwater pump room to the transformer yard, and installing an alternate safe shutdown capability independent of the 480-volt switchgear at the 15' elevation in the Control Building.

An evaluation of areas/systems potentially affected by the SPU follows.

Circulating Water System

A break in a circulating water line would actuate the level alarm switches in the IP1 condenser pit about 3 minutes after the break. This analysis in not affected by the SPU because there are no modifications to the CWS or flow rate changes resulting from the SPU.

Auxiliary Feedwater Pump Room

Failure of the main feedwater lines, located above and outside of the auxiliary feedwater pump room, would result in water accumulating at the 18'-6" elevation. In order to ensure that there is sufficient flow area to preclude flooding to elevation 19'-8" in the auxiliary feedwater pump room (below which there is no safety-related equipment), the door from the room at the 19'-8" elevation to the transformer yard has been modified to include a flap panel.

Flood water from the area in the Auxiliary Feedwater Building containing the feedwater lines (outside the auxiliary feedwater pump room) would only propagate into the auxiliary feedwater pump room through an interconnecting fire door with a small (½ inch) gap. There would not be a significant accumulation of water in the auxiliary feedwater pump room due to the drainage pathways available from that area and the relatively small rate of egress of water from the area containing the feedwater lines via the small gap under the interconnecting door. The interconnecting door is unlikely to fail open since the door swings into the area containing the feedwater lines.

Feedwater pump flow increases above the current flow at 100-percent power under SPU conditions (refer to Section 9.4 of this report). However, as previously described, adequate provisions are in place for assuring that flooding from the failure of a main feedwater line under uprated conditions will not affect safety-related equipment in the Auxiliary Feedwater Pump Room.

10.4.2 Flooding Inside Containment

The flood level inside containment resulting from a large-break loss-of-coolant accident (LBLOCA), documented in the IP2 Environmental Equipment (EQ) Program Plan, is at elevation 50'-1", and the flooded depth is 4'-1" above the containment floor. Safety-related equipment inside containment is located above this flood level, with the exception of certain electrical cables (refer to Section 10.8 of this document).

The flood level inside containment documented in the EQ Program Plan will not be affected by the SPU.

10.5 Probabilistic Safety Assessment

A Probabilistic Safety Assessment (PSA) is a tool that is useful for a quantitative and qualitative assessment of the likelihood and consequences of damage that could potentially result from events occurring during plant operation.

The model used in the IP2 PSA analyses is maintained and updated in accordance with plant procedures. Plant modifications that have the potential to significantly effect core damage frequency (CDF) or large early release frequency (LERF) are evaluated and incorporated, as appropriate, into the model following implementation of the change.

The effect of the SPU on the IP2 PSA will be evaluated, including the effect of plant modifications due to the SPU. The PSA "levels" to be addressed for the IP2 SPU are in accordance with existing procedures.

10.6 Station Blackout

The Station Blackout (SBO) Rule, 10CFR50.63 (Reference 1), requires that nuclear power plants be capable of withstanding a total loss of offsite AC power and onsite emergency AC power supplies. The Nuclear Regulatory Commission (NRC) issued Regulatory Guide (RG) 1.155 (Reference 2) to provide guidance in responding to the SBO Rule. This RG endorses a publication of the Nuclear Management and Resource Council (NUMARC), NUMARC 87-00 (Reference 3). Both RG 1.155 and NUMARC 87-00 were utilized in evaluating SBO at Indian Point Unit 2 (IP2).

The SBO minimum required coping duration for IP2 is determined to be 8 hours. The alternate AC (AAC) power source consists of combustion gas turbines. Since an AAC power source is utilized, IP2 is required to document the capability to cope without AC power for 1 hour.

Section 8.2.1 of the *Updated Final Safety Analysis Report* (UFSAR) states that: (1) at least 1 gas turbine generator and associated switchgear and breakers shall be operable at all times, and that a minimum of 94,870 gallons of fuel for the operable gas turbine shall be available at all times, and (2) if either of these requirements cannot be met, then, within the next 7 days, either the operable condition shall be corrected or an alternate independent power system shall be established. A 10CFR50.59 Safety Evaluation establishes that the Indian Point Unit 3 (IP3) Appendix R diesel generator and the IP2 emergency diesel generators (EDGs) can be used as an "alternate independent power system" in the event that all 3 gas turbines are simultaneously inoperable.

The IP2 SBO coping analysis addresses the following topics:

- Condensate inventory for decay heat removal
- Class 1E battery capacity
- Compressed air
- Effects of loss of ventilation
- Containment isolation

The following is a discussion of the effect of the SPU on the plant capabilities for coping with an SBO event for each of these topics.

Condensate Inventory for Decay Heat Removal

The condensate inventory for decay heat removal was determined using the methodology in NUMARC 87-00, which provides a bounding analysis for assessing condensate inventory. For

the SPU, the volume of water required for 8 hours of decay heat removal and primary system cooldown was determined acceptable. The Technical Specifications require that a minimum of 360,000 gallons of water must be available in the condensate storage tank (CST) during plant operation above 350°F. There is a large margin between the minimum required volume of water in the CST and the volume of water required for coping with an SBO event.

Class 1E Battery Capacity

Evaluation of plant fluid systems affected by operation at SPU conditions shows that there are no new SBO loads that require 125VDC control or motive power, and that there is no need to modify existing SBO loads that require 125VDC control or motive power. Accordingly, there is no change in the ability of IP2 to cope with an SBO event under SPU conditions.

Compressed Air

Based on existing plant SBO analyses and associated NRC safety evaluations:

- The air operated valves needed to cope with an SBO can either be operated manually or have sufficient backup sources independent of AC power for 1 hour coping duration, at which time the AAC power source will become available.

- A turbine-driven auxiliary feedwater pump, which operates during an SBO, requires the operation of pneumatic valves to admit water to the steam generators. When instrument air is not available, these valves can be operated locally or with backup nitrogen supply.

- The atmospheric relief valves (ARVs) are pneumatically operated, with nitrogen back-up. However, the ARVs are not required to maintain the unit in a hot shutdown condition, since the main steam safety valves (MSSVs) are set to maintain reactor coolant temperature at approximately no-load temperature.

- All other air-operated valves are designed to fail in the correct or safe position.

The power uprate does not affect these conclusions. The MSSVs have been evaluated for operability under SPU conditions and found to be acceptable (refer to Section 9.1 of this report).

Effects of Loss of Ventilation

Existing plant SBO analyses identified the auxiliary feedwater pump room as the only area of concern in accordance with the criteria of NUMARC 87-00. The relevant inputs and assumptions used to analyze this space bound the process conditions identified for the SPU.

As a case in point, main steam temperatures are well within the margin of temperatures used in the auxiliary feedwater loss of ventilation scenario. The inputs and assumptions for the other spaces discussed in the loss of ventilation analysis are not affected by the SPU.

Containment Isolation

An evaluation was performed confirming that appropriate containment integrity can be provided during an SBO event, where "appropriate containment integrity" is defined as providing the capability for valve position indication and closure of containment isolation valves independent of the preferred or Class 1E power supplies. The initial step consisted of reviewing the listing of containment isolation valves and excluding the following valves from consideration:

- valves normally locked closed during operation
- valves that fail closed on loss of power or air
- check valves
- valves in non-radioactive closed-loop systems not expected to be breached in an SBO
- all valves less than 3-inch nominal diameter
- valves that are normally closed and fail as-is.

The resulting list was reviewed to identify those containment isolation valves requiring manual operation and closure capability to cope with an SBO. A total of 13 valves was identified requiring local manual closure to establish containment integrity. Eight of the 13 valves are addressed in the Emergency Operating Procedure (EOP) for loss of all AC power (LOAC). The remaining 5 valves, which are not addressed in the LOAC EOP, isolate charging, and safety injection functions. It was determined that these 5 valves should remain in the open position under SBO conditions, thus assuring timely performance of the vital safety functions performed by these valves. (Operation of these 5 valves is addressed in the EOPs for transfer to hot-leg and cold-leg recirculation).

The SPU does not affect this evaluation.

References

1. 10CFR50.63, *Loss of All Alternating Current Power*, June 21, 1988.

2. Regulatory Guide 1.155, *Station Blackout*, August 1, 1988.

3. NUMARC 87-00, *Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout and Light Water Reactors*, November 1987.

10.7 In-Service Inspection/In-Service Testing Programs

In-Service Inspection Program

The "Third Ten-Year Inspection Interval (July 1,1994 – June 30, 2004) Inservice Inspection Plan for Indian Point Unit 2" (In-Service Inspection [ISI] Plan) outlines the requirements for the examination of ISI Class 1, 2, and 3 pressure retaining components and their supports at IP2. This plan was developed in accordance with the requirements of the 1989 Edition of the *ASME Boiler and Pressure Vessel Code*, Section XI, with subsections and exceptions as noted in the ISI Plan.

The system classifications used for the ISI Program are based on the requirements of 10CFR50 and Regulatory Guide (RG) 1.26 (Reference 1). Code Class 1, 2, and 3 designations, as referred to in 10CFR50.55a, are assigned to various components at IP2 in accordance with ASME Section XI, subarticle IWA-1320.

For modifications required in support of the stretch power uprate (SPU), the effect of the changes on the ISI Program will be evaluated as part of the engineering change process.

In-Service Testing Program

The In-Service Testing (IST) Program has been developed as required by Section 50.55a of 10CFR50 to implement the requirements of ASME Boiler and Pressure Vessel Code, Section XI. The Program is applicable to IST of pumps and valves for IP2's third 10-year interval (July 1, 1994 – May 18, 2005). The applicable edition of ASME Section XI for this interval is the 1989 Edition. This edition of the Code requires pump and valve testing to be performed in accordance with the requirements stated in ASME/ANSI OM, Parts 6 and 10.

The purpose of the IST Program is to assess the operational readiness of selected pumps and valves to perform a specific function. The pumps and active/passive valves covered under the Program are those which are required to perform a specific function in mitigating the consequences of an accident, or shutting down and maintaining the reactor in a safe shutdown condition.

For modifications required in support of the SPU, the effect of the changes on the ISI Program will be evaluated as part of the engineering change process.

References

1. NRC Regulatory Guide 1.26, Rev. 3, *Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste – Containing Components of Nuclear Power Plants.*

2. ASME/ANSI Operations and Maintenance Standard, Part 6 (OM-6), *Inservice Testing of Pumps in Light Water Reactor Plants*, 1987 Edition.

3. ASME/ANSI Operations and Maintenance Standard, Part 10 (OM-10), *Inservice Testing of Valves in Light Water Reactor Plants*, 1987.

10.8 Electrical Equipment Environmental Qualification Program

10.8.1 Introduction

The electrical equipment that is covered by the Electrical Equipment Environmental Qualification (EQ) Program has been reviewed for effects to its qualification as a result of the Indian Point Unit 2 (IP2) stretch power uprate (SPU). The review has been performed primarily by comparison of the new accident temperatures and radiation dose associated with the uprate to environmental conditions in the EQ Program.

The environmental parameters of pressure, humidity and chemical spray, and submergence are also addressed.

10.8.2 Environmental Parameters Inside Containment

The SPU has no effect on the qualification of equipment inside containment with respect to the temperature, but does have an effect with respect to qualification to radiation dose.

10.8.2.1 Normal Operating Temperature

The temperature during normal operation is unchanged from the qualification basis of 120°F. The qualified life of all EQ equipment inside the containment is unchanged.

10.8.2.2 Accident Temperature

The pre-uprate accident temperature profile used for the EQ Program bounds the containment re-analysis temperature profile from the loss-of-coolant accident (LOCA). IP2 does not use the main steamline break (MSLB) inside containment as a basis for EQ since it is licensed to Division of Operating Reactors (DOR) EQ requirements. Therefore, a composite LOCA/MSLB temperature profile is not evaluated for the power uprate review of EQ.

The equipment inside containment remains qualified on the existing bases for the temperature conditions associated with the SPU.

10.8.2.3 Accident Pressure

The LOCA pressure inside containment is bounded by the EQ pressure profile.

The equipment inside containment remains qualified on the existing bases for the pressure conditions associated with the SPU.

10.8.2.4 Radiation

The SPU radiation doses have increased as a result of the increased power the associated allowance for instrument error and the fuel cycle extension to 24 months. The total integrated dose for 40-year normal operation and accident radiation of 2.54×10^8 rads exceeds the documented qualification doses for several equipment types. For equipment that is not currently qualified for the 2.54×10^8 rads, further evaluation of the exposure of the critical radiation sensitive parts was made. This analysis takes into account the installation configuration of the equipment with respect to gamma and beta shielding and the construction of the equipment with respect to self-shielding against beta radiation. All equipment was determined to be acceptable for use within the requirements of the EQ Program.

10.8.2.5 Submergence

Flood level inside containment is discussed in Section 10.4. The cables in the EQ Program inside containment are qualified for submergence. Radiation doses to submerged cables increase as a result of the SPU and the fuel cycle change. To provide the SPU qualification for submergence, the scaling factor was applied for the SPU, and the normal 40-year operating dose for 3216 MWt added. All potentially submerged cables are qualified for the SPU with large margins.

10.8.2.6 Humidity

The normal and accident humidity has not been affected by the SPU.

10.8.2.7 Chemical Spray

The spray and sump water chemistry has been marginally affected by the SPU. The slight change of a fraction of a pH level is within the range of pH values covered in the EQ Program prior to SPU.

10.8.3 Environmental Parameters Outside Containment

The power uprate has little effect on the qualification of equipment outside containment with respect to the temperature, except for equipment in the main steam (MS) penetration area. There is also a small increase in the radiation levels for the SPU due to the recirculation of reactor coolant or sump water.

10.8.3.1 Normal Operating Temperature

The temperature during normal operation is unchanged and is bounded by the qualification basis of 105°F.

10.8.3.2 Accident Temperature

The 3 bounding high-energy line breaks (HELBs) for EQ equipment outside containment are:

- The MSLB in the steam and feedline penetration area
- The MS supply line to the turbine drive of the auxiliary feedwater pump in the auxiliary feedwater pump room
- The steam generator blowdown line break in the pipe penetration area

Main Steam to Auxiliary Feedwater Pump Turbine HELB

The existing HELB temperature analysis bounds the conditions of the SPU.

Steam Generator Blowdown Line High Energy Line Break

A check of process conditions was performed to determine the effect of the SPU on the steam generator blowdown line break. The Zaloudek correlation was used to compare the blowdown conditions that have been used for the existing equipment qualification to the conditions that will be present for the SPU.

The critical flow under the SPU conditions is only slightly greater than the pre-uprate conditions. The difference in the mass and energy (M&E) release is considered by engineering judgment within the conservatism in the M&E release analysis and, therefore, no change in the accident temperatures is necessary.

All equipment located in the areas that are affected by the steam generator blowdown HELB that were qualified remain qualified.

Equipment in the Primary Auxiliary Building (PAB) such as the residual heat removal (RHR) pumps and the safety injection (SI) pumps are in areas where there is no HELB effect. The only harsh environmental parameter is the LOCA radiation dose from the recirculated sump or reactor vessel water. The accident temperature is the same as the normal temperature.

MSLBs in Steam and Feedline Penetration Area

A spectrum of MSLBs have been reanalyzed for the SPU. The peak accident temperature for the break building area is above the qualification temperatures for the EQ equipment in these areas.

The equipment that is required to respond to these HELBs has been evaluated further using thermal lag analysis of the equipment response to the break environment for the spectrum of breaks. The limiting break for equipment qualification was identified as a 1-ft^2 break. The equipment in the steam and feedline penetration area is qualified considering the thermal lag analysis.

10.8.3.3 Radiation

The SPU effect on radiation outside containment has been evaluated. The beta radiation dose to EQ equipment outside containment is negligible. The radiation sources are inside process equipment and piping. In the event of a LOCA inside containment, the highly radioactive water is recirculated within process equipment and piping in the PAB and pipe tunnel. This water has a slightly higher radiation dose than before the uprate, but the effect on EQ is acceptable.

10.8.3.4 Humidity

The SPU does not change the normal operational humidity or the accident humidity outside containment.

10.8.3.5 Flooding

Flooding outside the containment is addressed in Section 10.4 of this document.

10.8.4 SPU Equipment Qualification Evaluation

Equipment Inside Containment

All equipment inside reactor containment is qualified for SPU conditions when the considerations discussed earlier in subsection 10.8.2 are made.

The equipment qualified life and post accident operability time are not impacted by the SPU.

Equipment Outside Containment

Accident temperatures outside containment in the steam and feedline penetration area have been reanalyzed and result in higher temperatures. All other areas outside containment experience insignificant temperature increases. All equipment outside containment required for accident response has been justified as qualified.

10.9 Chemistry Program

Primary Chemistry Program

The Indian Point Unit 2 (IP2) Primary Chemistry Program has been developed in line with the industry guidelines presented in Electric Power Research Institute (EPRI) Primary Water Chemistry Guidelines, and the technical basis for the IP2 reactor coolant chemistry parameters are consistent with the information contained in the EPRI Guidelines.

As addressed in subsection 4.1.2.1, the IP2 plant chemistry limits based on industry guidelines are still applicable after the IP2 SPU, and no changes to the Primary Chemistry Program are required for the IP2 stretch power uprate (SPU).

Secondary Chemistry Program

The purpose of the Secondary Chemistry Program is to maximize the useful life of the steam generators, to minimize unplanned or extended maintenance costs associated with steam generator repairs and inspections, and to maintain the integrity of other secondary plant equipment.

The IP2 steam generators were replaced during the 2000 refueling outage with Model 44F steam generators containing thermally treated Alloy 600 tubing and A-240 type 405 tube support plates. Secondary side steam generator chemistry has contributed to steam generator tube cracking in some units. The impact of the uprate on the potential for stress corrosion cracking of the Alloy 600 tubing is addressed in subsection 5.6.7 of this report.

10.10 Generic Letter 95-07

In 1995 the NRC issued Generic Letter (GL) 95-07, requesting that certain actions be taken by utilities regarding the susceptibility and evaluation of power-operated gate valves to the phenomena of pressure locking and thermal binding. Power-operated valves (POVs) include safety-related motor-operated valves (MOVs) and air-operated valves (AOVs).

The current evaluation of Indian Point Unit 2 (IP2) power-operated valves for pressure locking and thermal binding considered two types of pressure locking: pressure-induced pressure locking and thermal-induced pressure locking. It considered 2 types of thermal binding: seating effect and valve stem growth effect. However, only the valve stem growth effect was determined to be of potential concern. The results of this evaluation identified that potential pressure locking and thermal binding conditions will not prevent the plant from achieving safe shutdown, as all valves evaluated remain operable. This conclusion is based on valve design; plant configuration during normal, accident, and post-accident operating modes; and sufficient actuator thrust to open the valve.

The effect of the SPU on the current pressure locking and thermal binding evaluation of MOVs/AOVs was reviewed. It was determined that the stretch power uprate (SPU) does not introduce any increased challenge for thermal binding and/or pressure locking and does not effect the results and conclusions of the current evaluation.

References

1. NRC Generic Letter 95-07, *Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves*, August 17, 1995.

10.11 Generic Letter 96-06

Generic Letter (GL) 96-06 (Reference 1) requested that nuclear utilities address the susceptibility of: (1) containment air cooler cooling water systems to either waterhammer or two-phase flow conditions during postulated accident conditions, and (2) piping systems that penetrate containment to thermal expansion of fluid that could cause overpressurization of piping.

Regarding overpressurization of isolated water-filled piping sections, Indian Point Unit 2 (IP2) identified 9 piping segments subject to overpressurization. Three of these lines were determined to be acceptable based on relief valve lift pressure. The remaining 6 lines were determined to exceed *Updated Final Safety Analysis Report* (UFSAR) allowable stress limits; however, corrective actions and modifications restored these 6 lines to their UFSAR design requirements.

Regarding the potential for two-phase flow in the containment air cooler cooling water lines, IP2 determined that during a design-basis accident the discharge piping from the containment fan cooler units is susceptible to two-phase flow conditions. Analyses showed that although the flow would be reduced, the containment fan cooler units would still meet their required design heat removal rates.

Regarding GL 96-06 waterhammer issues, IP2 demonstrated that the methodology used in the tests and analyses at IP2 was consistent with EPRI guidelines for analyzing column closure waterhammer and condensation induced waterhammer. The Service Water System (SWS) has been shown by both in-plant testing and structural analysis to be acceptable, and the methods used to verify system acceptability have been evaluated and shown to be consistent with EPRI guidance.

Review of the evaluations of the piping segments subject to overpressurization leads to the conclusion that the only evaluation potentially affected by the stretch power uprate (SPU) is the evaluation of the return line from loop No. 2 hot leg to the suction of the residual heat removal (RHR) pumps. This piping insulation on this line was changed from 2 inches of calcium silicate to 6 inches of glass wool. An isolated water condition is assumed to exist between 2 MOVs in this line inside containment. The curve for containment temperature as a function of time following a large-break loss-of-coolant accident (LBLOCA) is an input used in the analysis of this piping segment. Due to the relatively small differences between the containment temperature profile used in this analysis and the containment temperature profile for a LBLOCA under SPU conditions, and a greater than 30-percent margin between the calculated maximum pressure and the maximum allowable pressure under UFSAR criteria, the stresses in this line under SPU conditions continue to remain within UFSAR allowables.

The effect of the SPU on GL 96-06 waterhammer issues was evaluated. It was concluded that the column closure waterhammer and the trapping and condensing of steam (steam bubble or void collapse) waterhammer will not be significantly affected by the small (less than 3 percent) increase in accident peak containment temperature under SPU conditions. That is, the velocity (critical parameter) of column closure and the volume (critical parameter) of steam bubble formation are not significantly changed by the small increase in containment ambient temperature.

References

1. NRC Generic Letter 96-06, *Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions*, September 30, 1999 and Supplement, November 13, 1997.

10.12 Generic Letter 89-13

Generic Letter (GL) 89-13 (Reference 1) identified a number of concerns affecting safety-related equipment associated with Service Water Systems (SWSs), and put forth recommended actions in the areas of surveillance, testing, inspection and maintenance to ensure these systems are in compliance with regulations.

The following safety-related heat exchangers are included within the scope of GL 89-13 at IP2:

- Containment fan cooler units
- Containment fan cooler unit motor coolers
- Emergency diesel generator (EDG) jacket water coolers
- EDG lube oil coolers
- Component cooling water (CCW) heat exchangers
- Instrument air compressor closed cooling water heat exchangers
- Radiation monitor sample coolers

In addition, the piping and components that supply these heat exchangers are addressed for GL 89-13.

The IP2 GL 89-13 testing consists of the initial testing program and a periodic re-testing. Test and analysis methodologies are provided to effectively monitor heat exchanger performance. Acceptance criteria for heat exchanger performance testing (for example, values for maximum total fouling) have been developed.

The effect of the SPU on service water heat exchanger heat loads is addressed in Section 9.6 of this report. The GL 89-13 heat exchangers experiencing an increase in heat load due to the SPU are the containment fan cooler units, the containment fan motor coolers, and the CCW heat exchangers. However, these increases in heat load are bounded by the original equipment/system design values.

Continued inspection of all GL 89-13 heat exchangers and testing of designated GL 89-13 heat exchangers post-uprate will ensure that the performance of all heat exchangers remains acceptable following the SPU.

The SPU does not affect the test and analysis methodologies for effectively monitoring heat exchanger performance.

The acceptance criteria for heat exchanger performance testing continues to be applicable under SPU conditions.

References

1. Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, July 18, 1989.

10.13 Plant Simulator

Indian Point Unit 2 (IP2) has a unit-specific simulator, which mimics the plant control room. The simulator computer systems provide simulator responses that are intended to match, to the greatest extent possible, actual plant conditions for the simulation of accidents and transients.

Appendix A to 10CFR55 (Reference 1) permits use of simulators for operator training. Regulatory Guide 1.149 (Reference 2), states that the requirements established by ANSI/ANS 3.5, *Nuclear Power Plant Simulators for Use in Operator Training*, for specifying the functional capability of a simulator and for comparing a simulator to its reference plant are acceptable to the Nuclear Regulatory Commission (NRC), subject to provisions identified in the Regulatory Guide. The IP2 simulator is currently certified to ANSI/ANS 3.5-1985 (Reference 3).

The implementation of the stretch power uprate (SPU) will result in changes in plant operating characteristics (software changes). These changes will range from simple changes in process parameters (for example flow rates) to changes in plant responses to transients and accidents.

Modifications in support of the SPU will be implemented in accordance with the IP2 engineering change process. This process requires review of "program impacts," which includes review of the impact of modifications on the IP2 simulator.

References

1. 10CFR55, Appendix A, *Operator License Eligibility and Loss of Simulation Facilities in Operator Licensing*, October 17, 2001.

2. Regulatory Guide 1.149, *Nuclear Power Plant Simulators for Use in Operator Training*, April 1981.

3. ANSI/ANS 3.5, *Nuclear Power Plant Simulators for Use in Operator Training*, 1985.

10.14 Containment Leak Rate Testing

Appendix J of 10CFR50 (Reference 1) requires a reactor containment to meet the containment leakage requirements specified therein. The Indian Point Unit 2 (IP2) Technical Specifications (Reference 2) specify the testing required to ensure that the plant meets Appendix J requirements. The test requirements provide for periodic verification by tests of the leak-tight integrity of the containment, and systems and components that penetrate containment, and establish the acceptance criteria for such tests.

The results of accident analyses under stretch power uprate (SPU) conditions show that the calculated peak containment pressure following these accidents is less than the containment design pressure. Accordingly, the SPU does not impose additional requirements on IP2 containment leakage testing.

References

1. 10CFR50, *Industry Codes and Standards, Amended Requirements*, Appendix J, September 26, 2002.

2. Indian Point Unit 2 Technical Specification 4.4, *Containment Tests*, Amendment No. 190.

10.15 Plant Operations

Entergy does not expect the changes to plant conditions and operation associated with the SPU to result in significant new or increased challenges to plant operators and other support personnel. Entergy recognizes the importance of this element of plant performance and safety and has taken specific actions to address the areas of potential concern. Operations has participated in the preparation and review of changes to the plant and associated operating procedures. Furthermore, appropriate training and simulator changes will be implemented prior to the SPU to ensure operations personnel are familiar with and prepared for any associated changes to the plant response.

10.15.1 Procedures

Although numerous, no significant changes to plant procedures will be required for the SPU. Changes associated with the SPU will be treated in a manner consistent with any other plant modification. The Emergency Operating Procedure (EOP) changes will be made as necessary to reflect the new power level and setpoint changes. The EOP step for addition of supplemental feedwater to steam generators after a trip already exists and has been demonstrated to be accomplished in less than 10 minutes. This procedure will be revised to provide specificity for the flow and time requirements.

10.15.2 Effect on Operator Actions and Training

Engineered Safety Feature System design and procedural controls have not changed with the SPU. Various setpoint changes will be required. However, the operator response to any event will be unaffected by a change in rated thermal power.

During or before the SPU is implemented, the Nuclear Instrumentation System, including alarm setpoints, will be adjusted to satisfy new analysis values contained in this report, and changes in improved Technical Specifications. The operator response to existing alarms is anticipated to remain as before.

The changes in operating procedures and setpoints will be part of operator training to be conducted prior to implementation of the SPU.

10.15.3 Plant Integrated Computer System

Process parameter setpoint and scaling changes will be made, as required, to the Plant Integrated Computer System (PICS). There are no other effects to the PICS from the SPU.

10.15.4 Startup Testing

Startup from the refueling outage when plant modifications will be made to accomplish the SPU will be treated as a special evolution. As in the previous uprate for the 1.4-percent MUR, the power escalation will be controlled by a specific procedure called a Temporary Operating Instruction (TOI). This procedure will detail controls for power escalation, hold points, and data collection requirements. Performance monitoring for plant modifications, such as the new high-pressure turbine, will be accomplished at specified power levels. Setpoint changes will be controlled under plant procedure requirements. Monitoring of NSSS plant performance will include monitoring of margins to activation of alarms and trips such as overpower ΔT and overtemperature ΔT. A vibration monitoring activity will be included to monitor plant response at various power levels. The TOI will be subjected to dry runs on the plant simulator to assure plant responses are as predicted. The results of the startup testing will be documented and maintained as plant records.

11.0 ENVIRONMENTAL IMPACTS

11.1 Introduction

The environmental issues associated with the issuance of an operating license for Indian Point Unit 2 (IP2) were originally evaluated in the IP2 Final Environmental Statement (FES) (Volume 1, page I-1 Section I) and addressed plant operation up to a maximum calculated thermal power of 3216 MWt. The Atomic Energy Agency (AEC), predecessor of the Nuclear Regulatory Commission (NRC), approved the FES in September 1972.

The Indian Point State Pollutant Discharge Elimination System (SPDES) restrictions on discharge temperatures and discharge flow rates for the station were evaluated along with the flow limits set forth in IP2 Consent Order.

11.2 Input Parameters and Assumptions

The IP2 FES that was approved by the AEC in September 1972 for a maximum calculated thermal power of 3,216 MWt envelops the SPU condition.

The Indian Point SPDES restrictions on discharge temperatures and discharge flow rates for the station were used in the SPU evaluation along with the flow limits set forth in the IP2 Consent Order.

The SPU evaluation assumes the existing Circulating Water System (CWS) pumps are not modified and continue to operate at the same flow rates. Since the CWS inlet temperatures from the Hudson River are not affected by the SPU, circulating flow is unchanged. As main condenser duty and exhaust flows will increase, the CWS discharge temperature to the Hudson River will increase.

Heat load increases due to SPU in the normal and emergency Service Water System (SWS) will result in increase in the SWS discharge temperature to the Hudson River.

11.3 Description of Analysis and Evaluations

IP2 operation at the SPU power level of 3216 MWt will increase the exhaust steam flow and duty of the main condenser and, therefore, increase the heat load rejected by the CWS and discharge temperature to the Hudson River. CWS flows were verified to be within the original design basis.

Heat load increases due to SPU in the normal and emergency SWS will result in increase in the SWS discharge temperature to the Hudson River. SWS system flows were verified to be within the original design basis.

11.4 Acceptance Criteria

The environmental impacts associated with the proposed changes are acceptable when they are within the existing regulatory release permits.

11.5 Results and Conclusions

Increased heat rejection to the CWS and SWS is expected to result in a nominal calculated increase in discharge temperature to the river. This temperature increase falls within the applicable SPDES permit thermal limits for Indian Point.