6.2 Containment Systems

The U.S. EPR containment systems include the containment, the containment isolation system, and the containment combustible gas control system. These systems contain any radionuclides released from the fuel during postulated accidents, preventing further release to the balance of the plant and the environment, and limit the accumulation of combustible gases generated during the accident.

The design basis accidents (DBA) for the containment systems are defined as the most severe event within a spectrum of postulated loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents. DBA mitigation depends upon the high reliability of these containment systems. This section provides the design criteria, design features, and evaluations that demonstrate these systems will function within their specified limits.

6.2.1 Containment Functional Design

The U.S. EPR Reactor Building consists of a cylindrical reinforced concrete outer Shield Building, a cylindrical post-tensioned concrete inner Containment Building with a steel liner, and an annular space between the two buildings. The Shield Building protects the Containment Building from external hazards.

The containment is designed to withstand the environmental and dynamic effects associated with both normal plant operation and postulated accidents (GDC 4).

The containment instrumentation is capable of monitoring variables and systems over their anticipated ranges for all normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to safety, including those variables and systems that can affect the containment and its associated systems. Appropriate controls maintain these variables and systems within prescribed operating ranges (GDC 13).

The containment and its associated systems establish a barrier against the uncontrolled release of radioactivity to the environment, and incorporate sufficient margin in their design so that conditions important to safety are not exceeded for as long as postulated accident conditions require (GDC 16).

The containment is designed so that, in conjunction with features built into the in-containment refueling water storage tank (IRWST) system, which rapidly reduce the containment pressure and temperature following any LOCA it can maintain them at acceptably low levels (GDC 38). As a result the containment can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA (GDC 50).

The containment is provided with the means for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents (GDC 64).

The containment conforms to the functional requirements of 10 CFR 50, Appendix K, which defines acceptable evaluation models and calculation of minimum containment pressure for evaluating emergency core cooling system capability, specifically for sources of heat during the LOCA and containment pressure control requirements. Refer to Section 15.6.5 for compliance with 10 CFR 50, Appendix K requirements.

Section 6.2.1.1 addresses those aspects of containment design and evaluation that relate to its accident mitigation functions. Section 3.8 provides a physical description of the containment and presents the design criteria relating to construction techniques, static loads, and seismic loads.

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The containment's structures, systems, and components (SSC) that are important to safety are designed to withstand the environmental and dynamic effects associated with both normal plant operation, including maintenance and testing, and postulated accidents. The environmental effects include the temperatures, pressures, and fluids encountered during normal and accident conditions. The dynamic effects include those arising from in-plant equipment failures or accidents, including missiles, pipe whipping, and discharging fluids, as well as those resulting from events and conditions outside the containment (e.g., tornadoes, earthquake, or aircraft impact).

The containment and its associated systems are designed to be a leaktight barrier against the release of radioactivity to the environment and are designed to remain functional during a DBA. By meeting these performance requirements, including requirements for access openings and penetrations of the structure and its internal compartments, the containment is designed to accommodate the calculated pressures and temperatures resulting from a LOCA without exceeding its designed leakage limits. It can do so with margin for extra energy sources, degraded engineered safety features (ESF), and using conservative calculational methods.

The radiological consequences of the DBA are presented in Section 15.0.3. The containment, containment systems, and ESF act to limit the release of radioactive material subsequent to a DBA, so that the release does not exceed the limits specified in 10 CFR 52.47(a)(2)(iv).

Containment design calculations assume the following for a RCS pipe rupture:

- The postulated rupture occurs concurrently with the worst single active failure.
- The systems used to mitigate the consequences of a postulated pipe rupture are protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit subject to design loadings from a safe shutdown earthquake.
- The offsite electrical power system is evaluated to provide the most limiting condition for each postulated break (i.e., loss of offsite power [LOOP] or no LOOP).
- Discharge coefficients (Cd) and the backpressure values are assumed, so as to result in the most limiting condition for each postulated break.
- Multiple pipe breaks do not occur simultaneously or consecutively.

The postulated RCS pipe ruptures are listed in Table 6.2.1-1—Loss of Coolant Accidents and are described in Section 6.2.1.3. Section 6.2.1.3.1 provides the mass and energy (M&E) release data for the LOCA.

Containment design calculations assume the following events occur for a secondary system pipe rupture:

- The postulated pipe rupture occurs with the worst single active failure of the main steam isolation valve (MSIV) for the MSLB.
- The offsite electrical power system is assumed to be available for the continued operation of the reactor coolant pumps (RCP) to maximize the primary to secondary heat transfer.
- Multiple pipe breaks do not occur simultaneously or consecutively.

The postulated secondary pipe ruptures are listed in Table 6.2.1-2—Main Steam Line Breaks and are discussed in Section 6.2.1.4. Section 6.2.1.4.3.2 discusses the M&E release data for the MSLB.

Containment overpressurization events during refueling operations and reduced primary inventory conditions are mitigated by RHR System design features as discussed in Section 5.4.7. These features prevent a loss of decay heat removal and do not result in a challenge to containment integrity under these plant conditions. Technical Specifications, Section 16.0, address containment integrity during fuel movement.

The loads on the internal structures are calculated using the differentials between the maximum calculated subcompartment pressures and 14.7 psia, the pressure of the containment atmosphere at the time of peak subcompartment pressure. These subcompartment pressures are specified conservatively for the analyses discussed in Section 6.2.1.2.

The U.S. EPR does not have an automatic containment spray system or containment air coolers for DBA mitigation. Thus, the U.S. EPR is not susceptible to inadvertent actuation of those systems, or the potential for damage because of the rapid reduction of the containment internal pressure that would result from such an inadvertent actuation. The severe accident heat removal system (SAHRS), described in Section 19.2, includes a manually actuated containment spray system dedicated to severe accident mitigation. This system is not used for DBAs. Because the SAHRS must be manually aligned and manually actuated, it is not subject to a single failure that could cause inadvertent actuation of containment spray; eliminating the need to analyze for this event. The following possible events were reviewed:

- Sudden Containment temperature reduction.
- Removal of IRWST inventory.
- HVAC pulldown of containment pressure.
- Postaccident cooldown.
- Post severe accident cooldown.

The limiting external pressure event was determined to be a sudden reduction in the containment ambient temperature. This reduction in temperature removes heat from the containment building, thereby removing water from the containment atmosphere via condensation on walls, stairs, and other containment structures. A sudden reduction in the containment temperature from normal ambient conditions to 59°F results in a pressure reduction of 2.92 psi inside the containment building, which is within the external design pressure of the building.

Containment heat removal is performed by recirculation of the reactor coolant from the IRWST, through the low head safety injection (LHSI) heat exchangers, to the RCS, and through the break back to the IRWST. The LHSI is part of the safety injection system (SIS) discussed in Section 6.3. The effects of the containment heat removal function of the LHSI heat exchangers are included in the determination of the containment pressure and temperature response discussed in Sections 6.2.1.3 and 6.2.1.4. The containment design evaluation considers the most limiting single failures for the SIS in the development of the long-term model for containment pressure and temperature response.

The principal parameters affecting postaccident pressure reduction are the heat absorbed by the heat sinks inside the containment and the heat transferred to the containment sumps, which are contained in the IRWST. A conservative amount of heat sink material has been calculated, and its heat absorption capability has been considered in the containment design evaluation discussed in Section 6.2.1.1.3.

The amount of heat transferred through the containment wall and dome to the outside atmosphere is determined to be insignificant.

Heat is transferred from the containment to the outside environment during an accident via the LHSI heat exchangers, which are cooled by the component cooling water system (CCWS). The CCWS is in turn cooled by the essential service water system (ESWS). The ESWS is described in Section 9.2.1, and the CCWS is described in Section 9.2.2. Limiting single failures of the LHSI heat exchangers cooling chain are considered in the development of the long-term model for containment temperature and pressure response. The capabilities of the LHSI heat exchangers are provided in Table 6.2.1-3—LHSI Heat Exchanger Data.

To meet the containment safety design basis of limiting the release of radioactive material from a DBA LOCA to acceptable limits, the containment pressure is required to be reduced to less than 50 percent of the peak containment pressure within 24 hours after the DBA LOCA. Chapter 15 discusses the analysis of the offsite radiological consequences of the accident and provides the basis for the containment depressurization rate.

The determination and evaluation of the minimum containment pressure transient are addressed in Section 6.2.1.5.

6.2.1.1.2 Design Features

Containment and subcompartment design parameters are provided in Table 6.2.1-4—Containment Initial Conditions and Table 6.2.1-5—Containment Heat Sink Inventory. The general arrangement drawings for the reactor containment are provided in Section 3.8.1, and simplified drawings illustrating the model used for the containment subcompartment analyses are provided in this section. The structural design of the containment and the subcompartments, and the applicable codes, standards and guides that apply to the design of the containment structure, are addressed in Section 3.8. The structural design considers the effects of postulated piping ruptures, as discussed in Section 3.6.

The design pressure and temperature of the containment are 62 psig and 338°F, respectively. Calculated containment pressures, based on the conservative analyses, are described in Sections 6.2.1.3 and 6.2.1.4.

The functional capability and frequency of operation of the systems provided to maintain the containment and subcompartment atmospheres within prescribed pressures, temperatures, and humidity limits during normal operation are discussed in Section 9.4.7.

6.2.1.1.3 Design Evaluation

The severity of the temperature rise and pressure peak resulting from a LOCA or MSLB depends upon the nature, size, and location of the postulated rupture. The U.S. EPR containment is designed to contain the energy released from the RCS in the event of a LOCA or from the steam generator (SG) during a MSLB.

In the case of a LOCA, the M&E released into the containment is reactor coolant at the primary system temperature. A portion of the coolant is converted to steam, and will remain as steam if its enthalpy is sufficient. Coolant released from the primary system causes an increase in containment steam mass, which in turn increases pressure and temperature. This rise is limited by steam cooling and condensation on contact with the colder containment walls, and by M&E exchanges between steam and liquid in the containment. The containment pressure rises until pressure between the primary system and the containment equalizes, and flow through the break decreases to an equilibrium value. The containment pressure then begins to decrease because of the effects of the passive heat sink of the containment.

Following blowdown, the water vapor condenses on the containment heat sinks located throughout the containment building, and the saturated water drains along the intermediate floors, grates, stairwells, and walls to the heavy floor of the containment building. The effects of condensation induce circulation zones that promote a mixed atmosphere inside the building during and after blowdown. The saturated water drains from the heat sinks pools and forms a large condensation surface on the heavy floor. In the case of a LOCA, saturated and later subcooled water spills out of the break, splashes on the heavy floor, and induces waves in the pooled water, which provides constant circulation that further promotes condensation on the pool surface. Curbed grates in the heavy floor drain directly to the IRWST, which in turn creates a fully developed recirculation path from the IRWST to the reactor pressure vessel through the LHSI heat exchangers. Steam condensation on the heat sinks and the water pooled on the heavy floor results in long-term containment cooling and depressurization.

In addition to the generation of the blowdown pressure peak, long-term LOCA cases were analyzed at each break location to determine the limiting pressure and temperature. The spectrum of postulated accidents analyzed is provided in Tables 6.2.1-1 and 6.2.1-2.

The analytical model and computer code designed to predict containment pressure and temperature responses following the accidents are described in this section. A summary of the predictions is listed in Tables 6.2.1-6, 6.2.1-7, and 6.2.1-8 for short-term containment response for LOCAs, and Table 6.2.1-9 for the MSLB.

Table 6.2.1-6, Table 6.2.1-7, and Table 6.2.1-8 present thirty-eight separate cases for LOCA analysis for three postulated break locations. For the LOCA, the limiting containment pressure results from the double-ended guillotine (DEG) break in the RCS hot leg (HL) piping, with the worst single failure being the loss of one ESF train.

Table 6.2.1-9 lists twenty cases for the MSLB, with four break sizes ranging from the DEG break to the 0.5 square foot break area, and power levels from 100 percent down to zero percent of rated thermal power (RTP). The peak containment pressure results from the assumed DEG MSLB with a failure of one MSIV at 50 percent RTP.

The passive heat sink inside the primary containment consists of all painted and unpainted concrete and metal surfaces that are exposed to the primary containment atmosphere. These areas are approximately the same temperature as the containment ambient temperature during normal plant operation. The specific passive heat sinks considered in the containment pressure-temperature analysis and their parameters are listed in Table 6.2.1-5—Containment Heat Sink Inventory. A minimum heat sink surface area was considered for conservatism.

The requirements of 10 CFR 50, Appendix K, part I.A list the required features of the evaluation models for sources of heat during the LOCA. For the heat sources of 10 CFR 50, Appendix K, it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times rated thermal power to allow for instrumentation error. The assumed power level may be decreased provided the proposed alternative value has been demonstrated to account for uncertainties of power level with a lower instrumentation error. The core power is measured using a secondary side heat balance with feedwater flow rate. A heat balance measurement uncertainty of approximately one-half percent of rated thermal power, or 1.005, is applicable to the core power for the U.S. EPR. This value is achieved with the use of an ultrasonic flow meter for the feedwater flow rate. This value is consistent with the assumption used in the safety analysis in Section 15.0.0.3.1.

The heat removal due to safety injection system/residual heat removal (SIS/RHR) system operation is simulated in the GOTHIC Version 7.2 computer code by specifying heat exchanger input values from Table 6.2.1-3. The GOTHIC heat exchanger model was benchmarked against heat exchanger performance data to provide a conservative representation.

Table 6.2.1-4 lists the initial containment conditions, based on the range of the normal expected conditions within the containment, with consideration given to maximizing the calculated peak containment pressure. Selection of these conditions is described in

Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC. (Reference 1)

The highest calculated containment pressure is produced by a MSLB break with the single active failure of one MSIV. A summary of the results of the containment pressure temperature analyses for the spectrum of postulated accidents is tabulated in Table 6.2.1-9—Peak Containment Pressure and Temperature for MSLB.

The IRWST is located near the basement floor of the containment building. The tank contains a minimum of 500,000 gallons of borated water and is maintained at a temperature between 60°F and 120°F. For the most limiting DBA, the IRWST temperature remains within the limit that supports continuous operation of the safety injection pumps to mitigate the consequences of the accident. The operation of the safety injection pumps provides the necessary cooling to limit containment temperature and pressure within design requirements. IRWST temperature versus time is presented in Figure 6.3-7.

The SIS has four accumulators to provide water to the RCS in the event of a LOCA. The accumulators' non-condensable cover gas (nitrogen) and the M&E release rates of the accumulators are included in the short-term model, and are supplied as input boundary conditions to the forcing functions in the long-term GOTHIC model. The nitrogen is assumed to enter the containment starting at time zero and is completely released at time 20 seconds, although the actual release of the nitrogen does not occur until the accumulators' liquid empties into the RCS loops. The calculations require that the nitrogen gas be assigned a temperature value. Since the nitrogen is stored within the accumulators above the water volume, the gas expands as the water drains from the accumulator into the RCS. The expansion results in polytrophic (pV^n) cooling. The cooled gas flows from the accumulators through the RCS piping, and to the containment atmosphere, where it mixes with the RCS coolant causing the nitrogen temperature to rise to the RCS coolant temperature. Since the RCS is depressurizing through the break, the RCS temperature would be lower than the normal operating temperatures. A bounding value of 565.5°F is assigned to the nitrogen. This value corresponds to the RCS cold leg temperature. This assumption is applied with all break locations and is conservative.

The long-term system behavior during various LOCAs has been evaluated to verify the ability of the SIS/RHR system to keep the reactor vessel flooded and maintain the containment within design conditions following a LOCA. This evaluation is based on the conservative predictions of the performance of the ESF consistent with the single failures assumed for each accident analyzed.

After a DBA, the conditions in containment are measured by postaccident monitoring instrumentation described in Section 7.5.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Basis

The containment internal compartments protect against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that result from equipment failures and from events and conditions outside the containment. The containment internal compartments accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. (GDC 4)

The reactor containment structure design, including access openings and penetrations, allows the containment internal compartments to accommodate the calculated pressure and temperature conditions resulting from any LOCA. The design must withstand these conditions without exceeding the design leakage rate requirement. (GDC 50)

Subcompartments within containment can withstand the transient differential pressures of a postulated pipe breaks. The subcompartment walls are challenged by the differential pressures resulting from a postulated break in a high energy line breaks (HELB) within individual compartments. These rooms are arranged to allow venting of high energy line breaks to prevent differential pressures from reaching structural limits of compartment walls.

Fluid systems are considered high-energy when, during normal plant conditions, the systems are operated or maintained pressurized under conditions where either or both of the following criteria are met:

- Operating temperature exceeds 200°F.
- Operating pressure exceeds 275 psig.

Fluid systems are considered moderate-energy systems when operated at the above conditions for 2 percent or less of the time the systems are in operation, or for less than 1 percent of the plant operation time.

For the U.S. EPR, the leak before break (LBB) concept is applied (Section 3.6.3) to preclude the need to design components, piping, and supports for the structural dynamic effects of postulated large or double-ended primary system pipe ruptures equal to the pressurizer surge line area or larger. The LBB concept also precludes the need to consider double-ended steam line ruptures in the structural design of the steam system components and supports.

6.2.1.2.2 Design Features

The general arrangement drawings for the reactor containment are provided in Section 3.8.1. These drawings form the basis of the subcompartment analysis models. Simplified drawings illustrating the model used for the containment subcompartment analyses are provided in this section.

The subcompartments identified to undergo the highest concentrated loading conditions (e.g., NSSS supports) are selected for determination of the differential pressures across the supporting walls. The combination of the NSSS concentrated loads and the subcompartment differential pressure creates critical loading scenarios on the supporting structural elements. These elements are then labeled as critical sections for the Reactor Building interior structures. Chapter 3, Appendix 3E presents the Reactor Building critical sections and the structural evaluations associated with them.

Subcompartments adjacent to critical sections are evaluated based on the M&E flux from each high energy line in the compartment (Table 6.2.1-10). Subcompartment without high energy lines are omitted from further analyses.

The U.S. EPR principal containment subcompartment design parameters are provided in Table 6.2.1-11, Table 6.2.1-12, Table 6.2.1-13, and Table 6.2.1-14, which includes the free volume and vent area for each critical subcompartment along with the neighboring subcompartment volumes. The vent paths considered in the subcompartment analysis include open doors, grates, and through wall openings. The effects of vent areas that become available after the occurrence of a postulated pipe break (e.g., blowout panels, hinged doors, collapsing insulation) are specifically noted and conservatively treated.

6.2.1.2.3 Design Evaluation

High-energy lines identified for the critical subcompartments are compared based on the full-power operating conditions. The mass flux from the postulated double-ended pipe rupture is then calculated using the Homogeneous Equilibrium Model (HEM). With the cross-sectional area for each of the high-energy lines known, the energy flux is calculated. The highest energy flux for each critical subcompartment will be selected for subsequent subcompartment analyses. These are listed in Table 6.2.1-15, Table 6.2.1-16, and Table 6.2.1-17.

The HEM model was used for the HELBs presented in Table 6.2-15 with the exception of the pipe breaks inside the RCP and SG cavities (identified as rooms 30UJA15-006, 30UJA23-004 and 30UJA29-004). The mass and energy release data for these breaks were calculated using the system analysis code CRAFT2 and are presented in Table 6.2-16 and Table 6.2-17. The GOTHIC computer code is used to determine the differential pressure across subcompartment walls. The calculation of the pressure load uses aspects of the NRC approved GOTHIC containment methodology pertinent to subcompartment pressure response in the Containment Response Topical Report (Reference 1). The suitability of the GOTHIC computer code to calculate differential pressures has been demonstrated in various experimental verifications.

A multi-node GOTHIC model is used as the analysis tool for conducting the subcompartment analysis because it utilizes the modeling of individual regions or subcompartments and connects them hydraulically by junctions or flow paths. The subcompartments rooms have been logically grouped together based on whether the rooms or regions experience gas flows significant enough to be considered well mixed.

Adjacent subcompartments with sufficient openings create a free exchange of gas flows between subcompartments and are grouped together into a lumped node. Subcompartments that include critical walls are separated from the lumped node and modeled as a single subcompartment or critical room node (Figure 6.2.1-1 through Figure 6.2.1-4). This re-nodalization is performed so that the magnitude of the initial peak or blowdown peak inside the critical room of interest is captured.

Each of the critical rooms, represented by a single node, were re-nodalized in the circumferential direction, into four nodes so that the initial peaks or blowdown peaks inside the various critical rooms are fully captured. These nodalization sensitivity studies showed that the pressure response generally varied by less than 1 psi as a result of more rigorous nodalization.

The HEM model in GOTHIC for air-steam-water mixtures is actuated for the junctions connected to the blowdown volume. The critical flow regime is expected to exist, if at all, in the junctions connected to the blowdown volume. The compressibility option is actuated for the flow paths connected to the blowdown volume. The compressibility option has the effect of slightly increasing loss coefficient because of increased density of upstream fluid when pressure drop across the junction becomes large. By selecting the HEM model for the flow paths connected to the blowdown volume, 100 percent droplet entrainment effects for these flow paths are captured.

The analysis approach is to inject the M&E release from the HELB into the relevant containment node that comprises the critical room or section that is exposed to the HELB. Initial conditions (e.g., containment pressure, temperature, relative humidity) at the receiving node and surrounding nodes are imposed to maximize the resultant differential pressure across the affected node. The axial effect is accounted for by injecting the M&E release at the elevation where the high energy line is located within the node.

The design pressure transients generated from postulated pipe breaks presented in Table 6.2.1-15 and Figure 6.2.1-5 through Figure 6.2.1-9 for the identified critical sections are designed for as shown in Chapter 3, Appendix 3E. The results of these evaluations show that the critical sections can withstand the applied loads including the subcompartment pressures and remain within allowable limits. The structural load calculations apply a factor of 1.4 to the peak pressure predictions from this analysis prior to their use as inputs in the design of the structures of interest.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss of Coolant Accidents

The large break LOCA (LBLOCA) determined to be the most limiting for the purposes of containment analysis a is a rupture of a large RCS pipe because it adds the greatest M&E to containment in the shortest period of time. This condition leads to a peak for containment temperature and pressure that is referred to as the blowdown peak.

An LBLOCA occurs in five phases:

1. Blowdown.

2. Refill.

3. Reflood.

4. Post-reflood.

5. Decay heat.

During the blowdown phase, there is a rapid depressurization of the RCS and the RCS and containment pressures eventually equalize. The coolant flow rate from the RCS to containment varies depending upon the nature, size, and location of the break. This flow leads to a maximum in the containment temperature and pressure referred to as the "blowdown peak." Core cooling during this phase is by film boiling on the surface of fuel rods. Because film boiling is inadequate to remove the heat contained within the fuel and the decay heat generated by the core, the fuel temperature increases.

As the RCS pressure falls below the pressure within the SIS accumulators, check valves open and water is added to the RCS. As long as there is a pressure gradient between the RCS and containment, water from the accumulators is entrained in the steam exiting through the pipe break. The SIS water cools the steam, and some of it condenses and remains within the primary system. The resulting condensation increases the core coolant flow velocities, and this begins to slow the rise in the fuel rod cladding temperature. The blowdown ends when the RCS pressure is approximately equal to the containment pressure.

Following blowdown, a refill period occurs where the SIS provides sufficient liquid to fill the reactor vessel's lower head and plenum regions. By this time, the SIS medium head safety injection (MHSI) pumps have started and provide a limited coolant flow to the RCS. Significant flow from the MHSI pumps is not available until later in the event. When the water level reaches the top of the lower plenum of the core, the refill phase is complete.

During the next phase, reflood, the water level rises from the bottom to the top of the reactor core. As the water level rises, the relatively cool water contacts the hot fuel cladding. This heat transfer mechanism can be violent in nature and may produce a "chugging" of the RCS flow. Chugging occurs as the water contacts the hot fuel surface and flashes to steam. As the steam expands, the water is rapidly pushed away from the fuel surface. Incoming water moves the steam away from the fuel area where it encounters cooler water and the steam condenses. This process of steam production and condensing produces the chugging effect of RCS flow through the entire core.

If the pipe rupture is in one of the RCS HL, the saturated steam and water mixture exits the break directly into the containment. If the pipe rupture is in one of the RCS cold legs (CL), the two-phase mixture may travel through the SGs, absorb more energy from the secondary side fluid, and become superheated before exiting to the containment. This reflood phase ends once the mixture reaches a level sufficient to quench the core. At this point, the fuel cladding temperature approaches the temperature of the fluid, and both temperatures approach the saturation temperature corresponding to the containment pressure.

The next phase is post-reflood, as the core is recovered by water and long-term cooling occurs. This period is characterized by the core decay heat generation of a significant two-phase mixture in the core region. This steam and water mixture is carried to the break location as in the reflood phase, but in more of a condition that is described as "pot boiling." In addition to decay heat and heat generated from the metal-water reaction, the reactor coolant system heat, the secondary side fluid heat, and the primary and secondary metal sensible heat are all released to the containment during this phase. The final phase of the LBLOCA is the decay heat period, during which the energy produced by the decay of fission products is removed.

For the U.S. EPR, the spectrum of LOCA breaks analyzed includes a range of CL pump discharge (CLPD), CL pump suction (CLPS), and HL breaks, ranging up to the largest postulated DEG break. These breaks range in area from 3.123 to 5.205 square feet.

A break in the HL piping was shown to produce the highest containment pressure during the blowdown phase of the accident. A DEG break of this pipe will allow the initial RCS M&E to enter the containment early in the transient, before the passive heat sink of the containment can effectively absorb the energy addition. Once the reflood phase begins, the M&E release decreases and the building heat sinks begin to reduce the pressure. As noted above, some of the water injected by the MHSI pumps and accumulators turns to steam in the core and exits the break. The steam will initially be superheated, but it will quickly become saturated as the core temperature is reduced. Containment pressure will continue to decrease from the peak established during the blowdown phase. As break size decreases, the M&E release slows, but the residence time increases. This allows additional energy to be transferred from the core and SGs.

The limiting break configuration for the HL break scenarios included a DEG break with minimum available safety injection and a postulated LOOP. Figure 6.2.1-10, Figure 6.2.1-11, Figure 6.2.1-12, and Figure 6.2.1-13 provide the pressure and temperature results for the limiting HL scenarios. During the blowdown phase of the event, the majority of the liquid that resides in the RCS is expelled to containment. The majority of the energy is the stored energy of this fluid. For a HL break, the introduction of this energy to containment is rapid and the containment heat sinks will not have an opportunity to absorb any appreciable amount of the energy; the containment pressure rises rapidly. The rate of pressure increase is proportional to the rate at which the energy is added. The M&E release model is set up to maximize the heat removal from the RCS, in particular the core region and SG. Steps have been taken to delay departure from nucleate boiling (DNB), for example, to maximize the heat transfer from the fuel to the RCS fluid. The liquid in the RCS may have an opportunity to gain additional heat depending on the transit time and path it takes to reach the break. The additional energy also affects the containment pressure. The initial stored energy of the RCS fluid, the rate at which the fluid is expelled, and any heat that the fluid gains as it traverses the system to the break will define the containment pressure response for a LOCA in the HL.

After the blowdown phase, the RCS is essentially in pressure equilibrium with the containment. As the system begins to refill, the M&E effluent to containment will decrease. It is at this point that the containment heat sinks are able to absorb the energy that has been added to containment and the pressure will begin to decrease. The accumulator injection will quickly quench the core, expelling the stored energy to containment. Thereafter, the energy addition to containment is just that generated by core boiling. Because the SIS injects into the cold legs, little flow will travel through the SGs to the break.

Figure 6.2.1-10 through Figure 6.2.1-13 clearly show the trends of containment pressure increases during the blowdown phase. At the end of blowdown phase, the containment pressure peaks and begins to decrease for the duration of the event. The variation in the peak pressure among the cases is very small. The postpeak behavior of these cases is also very similar. With consistent safety injection among the cases, the energy addition to containment during this time period is similar. At the end of the run, the variation in pressure among the cases is approximately 1 psi. Any of these cases could be considered the limiting case. Ample conservatism is included in the model, and the variation in the results is well within the accuracy of the model.

A break in the CLPS piping will not produce a limiting blowdown peak pressure of the containment. The resistance of the pumps will delay the blowdown as compared to the HL break scenarios, and the RCS depressurizes slightly at a lower rate compared to the HL break as a result, the accumulator injection is slightly delayed compared to the HL break. During the reflood phase of the event, initial steam generated in the core will be superheated and begin to approach a saturated condition as the core water level increases. Steam generated in the core passes through the SGs before exiting the break. The secondary side of the SGs is isolated and remains at a higher temperature and pressure than the RCS. Thus, the secondary side of the SGs acts as a heat source and transfers energy from the secondary to the primary system. Consequently, as the steam on the primary side passes through the SGs, it becomes superheated before it is discharged to the SG outlet plenum.

The limiting break configuration for the CLPS break scenarios included a split break with minimum available safety injection and no postulated LOOP. Figure 6.2.1-14 through Figure 6.2.1-17 provide the pressure and temperature results for the most limiting CLPS scenarios. The initial enthalpy of the break is lower for a CLPS break because of the location of the break; the energy addition rate starts off more slowly. However, the blowdown is longer due to the location of the break. Further, more fluid will traverse the SGs to get to the break than for the HL break due to the break location. At the end of blowdown, the transit time and path will allow slightly more M&E into containment. However, the containment heat sinks will have additional time to absorb this energy. As a result, the peak containment pressure predicted during blowdown is less than that predicted for a hot leg break.

After blowdown, the accumulators and pumped safety injection begin to quench the core, removing its stored energy. As the quench front builds, the break effluent stabilizes, allowing the containment heat sinks time to absorb the energy in containment and reduce the pressure. The core steaming rate will eventually increase, increasing the steam mass flow rate through the break. As the steam from the core traverses the SG, additional energy will be added. As a result, the energy content of the break effluent increases beyond the capacity of the containment heat structures and the containment pressure begins to rise again. The break energy will decrease as the core residual heat decays, and the SIS slowly starts to suppress the core boiling rate or the SG secondary temperature reaches equilibrium with the RCS fluid temperature. In either case, the steam energy should be near saturation.

A number of cases showed that once the accumulators have emptied, the break flow initially exceeds the minimum pumped safety injection. As a result, the liquid volume in the RV decreases in the short term. During this time, the core is cooled by a mixture of steam and liquid. At some point, the volume of liquid in the RV decreases until it can no longer maintain quench front. Consequently, steam production in the core will cease, and the decay heat will begin to heat up the fuel and cladding instead of producing steam. The end result is that there is no steam to expel to containment and the containment pressure will begin to decrease.

A break in the CLPD piping produces the lowest peak containment pressure, although the pipe area of the CLPD is the same as the CLPS piping, and even although the blowdown phase will be similar in duration and produce a similar containment pressure response. During the refill and reflood phases of the CLPD event, the SIS pump flow injects into the CLPD piping and out the break. Steam exiting the break from the reactor vessel side condenses because of the lower temperature from the SIS liquid, and the break effluent is in a saturated condition. For this break configuration, the SIS flow from the MHSI pump and the accumulator in the broken loop exit directly to the containment. With the reduced SIS inventory, the core reflooding rate will be reduced and the effluent quantity through the CLPD is similar to that predicted for the CLPS break. The reduced SIS flow will reduce the mass flow rate through the break, and as a result, the total break energy delivered to the containment. Therefore, the CLPD break will result in the lowest containment pressure.

The limiting break configuration for the CLPD break scenarios included a split break with minimum available SIS and no postulated LOOP. Figure 6.2.1-18 through Figure 6.2.1-21 provide the pressure and temperature results for the most limiting CLPD scenarios.

Subsequent to blowdown, the containment pressure for all the cases continuously decreases without a large variation among the cases. In all cases, the downcomer has formed a liquid seal so that all of the steam generated in the core must traverse the hot legs and the SGs on the way to the break. The steam that goes through the intact loop must pass pumped injection locations on the way to the RV downcomer and through the break. As a result of the condensation on the safety injection fluid, the effluent through the RV side of the break has a low enthalpy. Because there are no condensation sources on the way, the steam that goes through the broken loop to the break will be superheated.

The U.S. EPR LOCA analyses examine a spectrum of breaks and include variations in the SIS flow, offsite power availability, and pipe break size and configuration. In addition, sensitivity studies were performed to determine the appropriate containment pressure boundary condition for the M&E release analysis.

6.2.1.3.1 Mass and Energy Release Data

Blowdown M&E release data are presented in Table 6.2.1-18 through Table 6.2.1-20 for the three break locations analyzed. These data represent the limiting configuration with respect to SIS flow, offsite power availability, and containment pressure.

To maximize the containment peak pressure and temperature, the U.S. EPR LOCA analysis uses conservative assumptions that maximize the M&E released from the RCS to the containment atmosphere. These assumptions maximize the primary system inventory and the heat into the RCS, and also maximize transfer of M&E into the containment. To achieve these conservative assumptions, the computer models:

- Maximize the initial reactor power level.
- Maximize the pressurizer volume (thereby increasing primary system inventory).
- Minimize the rate of power decrease.
- Maximize the reactor decay heat.
- Maximize the heat transfer from the secondary system into the primary system.

Table 6.2.1-21—Input Summary for Mass and Energy Release provides a summary of the initial conditions for the calculation of the M&E release.

The blowdown phase M&E release rates are calculated by the thermal-hydraulic analysis code, RELAP5/MOD2-B&W. The NRC has reviewed and approved this code as meeting the requirements of 10 CFR 50, Appendix K for pressurized water reactors with recirculating SGs. 10 CFR 50, Appendix K methods limit the energy transfer from the fuel elements to the RCS fluid to maximize the cladding temperature. While this approach is appropriate for analyses pursuant to 10 CFR 50.46, it is not sufficient for the calculation of M&E release rates for containment analyses. Therefore, the method is modified to maximize core heat removal to maximize the containment temperature and pressure response following a LOCA. This adjustment of the method from the 10 CFR 50, Appendix K requirements is consistent with NUREG-0800 and ANSI/ANS-56.4. This adjusted model is used to calculate the M&E released to containment from the beginning of the long-term cooling phase, or time of core quench.

Post-reflood M&E release rates are referred to as long-term LOCA and are determined by the GOTHIC Version 7.2 computer code, as presented in Section 6.2.1.1.3, using the model described in Section 6.1.2.3.4.

6.2.1.3.2 Energy Sources

The sources of stored and generated energy used in the LOCA analyses include:

- Reactor power.
- Decay heat.
- Stored energy in the core.
- Stored energy in the RCS fluid and metal, including the reactor vessel and internals.

- Metal-water reaction energy.
- Stored energy in the secondary system, including the SG tubing and secondary water.

The initial reactor power level for the analyses is the RTP level plus an appropriate calorimetric uncertainty. Reactivity components are chosen to provide a conservative insertion of negative reactivity. An appropriate initial stored energy in the core is obtained by using a conservatively high initial fuel temperature. The RCS metal is modeled accurately with respect to its size, location, and composition. The SG secondary side metal mass that is in contact with RCS fluid is also explicitly modeled, and the code includes appropriate computation of the heat transfer across the SG tubes. The energy addition due to the metal-water reaction is calculated based on the same correlation (Baker-Just) specified in the approved 10 CFR 50, Appendix K method.

6.2.1.3.3 Description of Blowdown Model

A description of the RELAP5/MOD2-B&W model used to determine the M&E released from the RCS during the blowdown phase of a postulated LOCA is provided in BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants (Reference 2) and RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis (Reference 3). All significant correlations are discussed in these reports.

6.2.1.3.4 Description of Long-Term Cooling Model

Long-term M&E release rates are determined by the GOTHIC Version 7.2 computer code, as described in Section 6.2.1.1.3. The long-term energy release rates for RCS metal and SG metal were modeled by the code as multiple heater components and as coupled boundary conditions. The long-term RELAP5/MOD2-B&W transients are analyzed to produce M&E releases following a LOCA. These transients are terminated at between 5000 and 10,000 seconds, depending upon the break location. During the event, the core is quenched, and the MHSI flow removes decay heat from the reactor. To continue the M&E release calculation for the long term, data obtained from the RELAP5/MOD2-B&W analysis is provided for each long-term case. The following information about the RCS primary and second systems at the end of these transients is included in the calculation of the long-term M&E release rates.

- The RCS primary side metal stored energy relative to 32°F, including the metal in the pressurizer and SG tubes.
- The stored energy content of the core relative to 32°F.
- The secondary side metal stored energy relative to 32°F.

- The RCS primary side fluid mass and internal energy, including the pressurizer.
- The secondary side fluid mass and internal energy, from feedwater inlet to the exit of the steam from the SG.
- The SIS injection flow and enthalpy.
- The RCS pressure, the average temperature for the vapor and liquid, and the final liquid-to-volume fraction.
- The mass flow of the SIS.

6.2.1.3.5 Single Failure Analysis

The effect of single failures of various SIS/RHR system components on the M&E releases is included in these analyses. No single failure is assumed in determining the M&E releases for the maximum safeguards case. For the minimum safeguards case, the single failure assumed is the loss of one emergency diesel generator that results the loss of one complete train of ESF equipment, including the loss of one safety injection train. The analysis of both maximum and minimum safeguards cases bounds the effects of credible single failures.

6.2.1.3.6 Metal-Water Reaction

The exothermic metal-water reaction is calculated using the Baker-Just correlation, as specified in 10 CFR 50, Appendix K.

6.2.1.3.7 Additional Information Required for Confirmatory Analysis

System parameters and hydraulic characteristics needed to perform confirmatory analysis are provided in Table 6.2.1-21 and Figure 6.2.1-22 through Figure 6.2.1-33.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures inside Containment

Steam line ruptures inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, producing high containment temperatures and pressures. The M&E release following a main steam line break (MSLB) depends upon the configuration of the plant's main steam system, the containment design, the plant operating conditions, and the size of the pipe rupture. This section describes the methods used to determine the containment response to these conditions.

Table 6.2.1-9—Peak Containment Pressure and Temperature for MSLB lists the scenarios analyzed to determine the limiting case (i.e., highest containment pressures and temperatures) following a MSLB. The scenarios assume a single active failure of one MSIV. The main feedwater (MFW) system includes redundant isolation valves, which preclude an over feeding failure. In all the scenarios, isolation of the MFW system uses the longest delay time and the slowest isolation valve as allowed by NUREG-0588 (Reference 4) model analysis. These scenarios use an eight percent re-vaporization fraction combined with the Uchida heat transfer correlation.

Containment pressure and temperature response to a MSLB depends upon the amount of break effluent that enters the containment atmosphere as steam, and whether that steam is at saturated or superheated conditions. During the depressurization of the SG, there are two phenomena that could reduce the steam contribution by forcing liquid effluent into the containment. The first is the entrainment of liquid drops that are swept out the break because of the high steam velocities. The second is the rapid voiding of the SG that causes a liquid level swell that extends to the break location, and that allows the discharge of saturated liquid. This swelling of liquid level to the steam outlet nozzle and subsequent discharge at the break, is caused by void formation or flashing in the liquid regions of the SG.

The containment response analyses are based on M&E released from the MSLB and includes the effects of superheated steam. Smaller breaks require an iterative process for determining the reactor protection system response time to execute a reactor trip signal. The analytical trip setpoint for the ESF actuation system for containment pressure is 4.0 psig plus an additional 0.5 psig uncertainty. An additional 1.0 second delay for the high containment pressure signal applies to all breaks, so that the time required for the reactor trip is conservative.

6.2.1.4.1 Significant Parameters Affecting Steam Line Break Mass and Energy Releases

A number of important system design, plant operation, and rupture event parameters affect the containment response to secondary side events. For each of these, there are four major factors that influence the M&E release following a MSLB:

- SG fluid inventory.
- Primary-to-secondary heat transfer.
- Protective system operation.
- State of the secondary system fluid blowdown.

These factors are addressed in the following descriptions of how the important plant parameters impact containment response to secondary side rupture events.

6.2.1.4.1.1 Plant Power Level

MSLBs are postulated to occur with the plant in operating conditions ranging from hot zero power (HZP) to 100 percent RTP. The mass of water in the SG decreases with increasing power level. Therefore, an MSLB occurring at a low power level would be expected to produce a greater total mass release to the containment than one occurring at 100 percent RTP. However, because of greater primary system stored energy, increased heat transfer in the SGs, and the additional energy generation in the nuclear fuel, the energy released to the containment from postulated breaks that occur during power operation may be greater than the energy released with the plant at HZP. Additionally, steam pressure and the dynamic conditions in the SGs change with increasing power, and these have significant influence on both the blowdown rate and the amount of moisture entrained in the fluid that exits during the MSLB sequence. Because of these opposing effects of power level on M&E release, no single power level will be the limiting condition for MSLB events. Therefore, HZP and power levels spanning the operating range have been analyzed.

6.2.1.4.1.2 Main Feedwater System Design

The rapid depressurization following a MSLB can cause a large volume of water to be added to the SGs by the MFW system. Therefore, the MFW lines have isolation valves that can rapidly close to limit feedwater addition during the event. The MFW piping layout downstream of these isolation valves impacts the event because it affects the volume in the feedwater piping that would enter SGs. As the SG pressure decreases, the fluid in this MFW piping flashes into steam and provides additional secondary fluid to exit the rupture. The feedwater volume and duration of flow can influence the SG blowdown in three ways:

- The rapid addition increases the amount of entrained water in large break cases by lowering the bulk quality of the SG inventory.
- The water entering the SG is subcooled, and it decreases the steam pressure reducing the flow rate out of the break.
- The increased flow causes an increase in the heat transfer rate from the primary-to-secondary system, resulting in greater energy release out the break.

These are competing effects on the total M&E release during a MSLB, so bounding conditions are provided in all MSLB scenarios. During periods of predicted entrainment (i.e., the entrainment of water and steam exiting the rupture), the break energy is set to the energy of saturated steam. The MFW system includes isolation valves and control valves, which close upon receipt of an isolation signal, terminating MFW flow. The MSLB analysis examines the single failure of a MFW isolation valve, a MFW control valve, failure of the MFW pumps to trip, and the failure of a main steam isolation valve, in order to determine the most limiting scenario.

6.2.1.4.1.3 Emergency Feedwater System Design

Actuation of the emergency feedwater system (EFWS) during an MSLB increases the SG mass available for release to containment. The temperature of the emergency feedwater is low compared to the temperature of the SG inventory. The EFWS water addition cools the SG inventory and decreases the enthalpy of the break flow, but at the same time, the EFWS water absorbs heat from the SG tubes and other metal structures. This latter process provides an additional transport mechanism for that energy to the containment, and the net effect of the emergency feedwater actuation is to increase the containment's peak temperature and pressure. Because of these competing phenomena, the overall effect of the EFWS on the MSLB containment response is minimal.

Emergency feedwater can be initiated by either low SG level or a safety injection signal coincident with a LOOP signal. However, the LOOP is not credited during this event, so in the MSLB analysis, the emergency feedwater actuation can only occur on low SG level. The MSLB analysis model does not consider SG level, so emergency feedwater activation is conservatively assumed to occur coincident with reactor trip. Upon activation, the time required for the EFW pump to reach full flow is conservatively modeled as 1 second, at which time full EFW flow is delivered to the SG.

Emergency feedwater flow depends on the discharge pressure. However, MSLB analysis does not consider this, so the analysis assumes the highest possible flow to the affected SG.

The emergency feedwater system isolates on high level in the SG. Since SG level is not modeled, isolation of the emergency feedwater is assumed to occur by operator action 30 minutes after the start of the event. The MSLB transient simulation proceeds until peak containment pressure and temperature occurs, and this occurs within 30 minutes. Therefore, MSLB analysis assumes emergency feedwater flow for the duration of the event.

6.2.1.4.1.4 Postulated Break Size, Type, and Location

The postulated break area is important when evaluating MSLBs because it controls the rate of the releases to the containment, it exerts significant influence on the steam pressure decay, and it impacts the entrainment of water in the blowdown flow.

Releases are analyzed for four MSL breaks: the DEG break and break sizes of 1.0, 0.7 and 0.5 square feet in area. The resulting containment pressure decreases with the break size. Each of the breaks sizes are analyzed at five power levels.

Each SG is equipped with a flow limiting orifice that limits the effective area for the MSLB. Following the isolation of the SG with the broken line, the effective break area can be no greater than the flow restrictor throat area of 1.4 square feet.

The largest DEG break, for which the steam pipe is completely severed, is limited by the throat area of the SG flow restrictor. This type of break influences the M&E

releases to containment by altering both the nature of the steam blowdown from the piping of the SG, and the effective break area fed by each SG prior to steam line isolation.

Break location affects steam line blowdown by virtue of the pressure losses that occur in the length of piping between the SG and the break location. The effect of the pressure loss is to reduce the effective break area seen by the SG. This would reduce the rate of blowdown, but it would not significantly change the total release of energy to the containment. Therefore, piping loss effects have been ignored in all blowdown analyses.

6.2.1.4.1.5 Availability of Offsite Power

The U.S. EPR does not have a containment spray system or containment fan coolers as part of the engineered safety features that would be delayed if there was a LOOP. Therefore, offsite power is assumed to be available, and the M&E released from the break are maximized due to continued operation of the reactor coolant pumps. The energy transferred from the reactor coolant system to the SGs, with continued operation of the MFW and EFW pumps, maximizes the SG inventories available for release to containment.

6.2.1.4.1.6 Safety System Failures

The most severe single active failure is the failure of a MSIV. An MSIV failure would provide additional fluid that could be released to the containment via the break. This fluid comes from the blowdown of all the steam piping between the break location and the isolation valves in the intact loops.

6.2.1.4.1.7 Steam Generator Reverse Heat Transfer and Reactor Coolant System Metal Heat Capacity

Following isolation of the intact SGs, energy is transferred to the containment building via the primary coolant. As the steam from the SG enters containment, the reduction in secondary side pressure creates a cooling effect on the primary system. Heat transfer occurs as the temperature of the primary coolant flowing in the SG tubes falls below the bulk temperature of the intact SGs. With a MSLB, this energy is then available to be transferred to the SG.

Similarly, the sensible heat of primary system heated structures must be considered, including the:

- Reactor coolant piping.
- Reactor pressure vessel.
- Reactor coolant pumps.

Heat from these components will be transferred to the primary coolant as cooldown progresses; during an MSLB, this energy is available to be transferred to the SGs. The effects of both the reactor coolant system metal and the reverse SG heat transfer are included in the results presented in containment response analyses.

6.2.1.4.2 Description of Blowdown Model

A description of the RELAP5/MOD2-B&W model used to determine the M&E released during the blowdown phase of a postulated MSLB is provided in B&W Safety Analysis Methodology for Recirculating Steam Generator Plants (Reference 5) and the RELAP5 Topical Report (Reference 3). All significant correlations are discussed in these reports.

6.2.1.4.3 Containment Response Analysis

The containment response to postulated MSLBs was analyzed with GOTHIC Version 7.2 (see Section 6.2.1.1.3). The containment model was developed in accordance with the Containment Response Topical Report (Reference 1).

6.2.1.4.3.1 Initial Conditions

The initial conditions used in the GOTHIC model are addressed in Section 6.2.1.1.3.

6.2.1.4.3.2 Mass and Energy Release Data

Table 6.2.1-22—MSLB Mass and Energy Release Data present the M&E release data used to determine the containment pressure and temperature responses for the limiting MSLB, a DEG break of the main steam line at 50% RTP concurrent with the single active failure of a single MSIV.

Feedwater isolation for the full and partial DEG breaks depends on signals generated by the ESF's instrumentation. The feedwater flow rates used in the analyses credited the longest isolation valve stroke time of 40 seconds. Valve leakage is not considered in the analyses as it is bounded by the emergency feedwater injection.

6.2.1.4.3.3 Containment Pressure and Temperature Results

Figure 6.2.1-34 and Figure 6.2.1-35 provide the pressure and temperature results for the most limiting MSLB scenario. Table 6.2.1-9 summarizes the results of all the cases analyzed.

The worst single active failure for the MSLB is the loss of one MSIV. Because the U.S. EPR includes redundant safety-grade feedwater isolation valves, the failure of the main feedwater isolation valve was not specifically analyzed. However, as an additional conservatism, the MFW valve with the longest stroke time is credited for isolation. The additional feedwater that would be released following isolation of the feedwater system was not explicitly modeled. Full feedwater flow is credited from the beginning of the transient until the isolation valve is fully closed. The volume of water in the feedwater pipe is considered small and does not significantly affect the pressure and temperature response of the containment.

The loss of one MSIV has been assumed for the spectrum of break sizes and power levels analyzed. As illustrated in Table 6.2.1-9, a full DEG break of the main steam line at 50 percent Rated Thermal Power results in a peak pressure of 54.14 psig. This case represents the peak calculated containment pressure for the spectrum of breaks analyzed.

For the spectrum of breaks analyzed, the calculated containment vapor temperature for some cases exceeds the specified containment design temperature of 338°F for a short period of time. While the analyses show the vapor space is superheated the containment walls and structures certainly are not. The primary mode of heat transfer during this time period is condensation on the building surfaces. Therefore, the building surface temperature will be no greater than the saturation temperature at building design pressure of 62 psig, or 309.1°F. Figure 6.2.1-35 shows that the analysis predicts that the containment vapor temperature remains above the design temperature for less than two minutes.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System

6.2.1.5.1 Mass and Energy Release Data

Containment pressure calculations are performed by the ICECON module within S-RELAP5 code. ICECON is a variant to the CONTEMPT containment code series. The RLBLOCA methodology treats containment pressure as a statistically varied parameter with a random sampling of the containment volume. The tabular M&E release data are not explicitly generated because they are part of the internal code calculations at each time step. The mathematical models that calculate the M&E releases to the containment are described in Section 15.6 and conform to ECCS evaluation models of 10 CFR Part 50, Appendix K.

6.2.1.5.2 Initial Containment Internal Conditions

The initial values for the containment conditions are representative of 100 percent rated thermal power, and are a pressure of 14.7 psia and temperature of 59°F. The outside atmospheric temperature of 20°F and relative humidity of 100 percent are assumed and modeled within the ICECON module, which also assumes an outside atmospheric temperature of 20°F and a relative humidity of 100 percent. Inside the containment, the IRWST water temperature is expected to be at the containment temperature of approximately 60°F, but could range as high as 120°F, which is the

Technical Specification maximum value. The RLBLOCA methodology uses the value of 120°F.

6.2.1.5.3 Other Parameters

The containment pressure varies and the RLBLOCA methodology determines it by sampling the containment volume. The nominal or best-estimate value of the containment volume is 2.888×10^6 ft^3. The upper estimate value for the containment volume is 3.645x106 ft^3 and represents the empty volume of the containment dome and cylinder and also neglects the volume displaced by the internal walls and structures. This latter value is conservative because a lower containment backpressure results in the highest calculated peak cladding temperature. Heat transfer between the IRWST water and the containment vapor space is not considered in the analysis. Water spillage rates from the accumulator in the broken loop are determined as part of the core reflooding calculation, and are included in the containment code calculational model. The passive heat sinks and thermo-physical properties were derived in accordance with Branch Technical Position 6-2, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

6.2.1.6 Tests and Inspections

Refer to Section 3.8.1.7 and Section 3.8.2.7 for testing and inspection requirements for the containment structure. Refer to Section 6.2.6 for the containment leakage rate testing program, and Section 6.6 for inservice inspection of ASME Class 2 and 3 components. Containment testing and inspections are also included in the Technical Specification (Chapter 16), which specifies the inservice inspection and testing required for the plant's operating license.

6.2.1.7 Instrumentation Requirements

Refer to Section 7.3 for engineered safety features instrumentation. Refer to Section 12.3.4 for radiation monitoring instrumentation.