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 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 limiting events with respect to design limits within a spectrum of postulated loss of coolant accidents (LOCA) and secondary system pipe ruptures. DBA mitigation depends upon the high reliability of these containment systems. This section provides the design criteria, design features, and evaluations that demonstrate that 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 operations, 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 segregated into two zones delineating areas that are accessible during normal operation from those that are inaccessible. Equipment rooms immediately surrounding the Reactor Coolant System (RCS) are isolated from the rest of the containment during normal operation. Beyond this inner region, personnel access can be allowed for certain maintenance tasks. Separation is provided by structures and closed portals to minimize radiation exposure in the accessible areas. During power operation, the inaccessible areas inside containment (the "equipment space") experience higher temperatures than the accessible areas because they are exposed to the hot walls of the nuclear steam supply system. The cooler, accessible areas are the “service space.”

In the event of an accident, communication is established between the two zones by opening mixing dampers, foils, and safety-related doors, thereby transforming the containment into a single convective volume. This transformation into a single convective volume is performed by the CONVECT system and safety-related doors, which equalizes pressure between the containment compartments and promotes efficient mixing of the atmosphere by establishing a global convective pathway. The CONVECT system of convection foils, rupture foils, and mixing dampers is part of the combustible gas control system (CGCS).

The containment is designed so that the CONVECT system and safety-related doors, in conjunction with recirculation features built into the in-containment refueling water storage tank (IRWST) and re-alignment of the emergency core cooling system (ECCS) system to the hot legs, will rapidly reduce the containment pressure and temperature following a LOCA. These systems maintain the containment pressure and temperature 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 during normal operations, including anticipated operational occurrences, and from postulated accidents (GDC 64).

The containment conforms to the functional requirements of 10 CFR part 50, Appendix K, which defines acceptable evaluation models and calculation of minimum containment pressure for evaluating ECCS 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 part 50, Appendix K requirements.

Section 6.2.1.1 addresses those aspects of containment design and evaluation that relate to its accident mitigation functions. Containment performance during refueling operations and reduced primary inventory conditions are discussed in Section 6.2.1.1.1, and the disposition of GL 88-17 for the U.S. EPR design is provided in Section 5.4.7.2.1 and in Table 15.0-60. 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 fluid discharge, as well as those resulting from events and conditions outside the containment (e.g., hurricanes, 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 and 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 an 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 fluid discharge, 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, for example, a loss of offsite power (LOOP) or no LOOP.
- The building doors that are non-safety are conservatively not allowed to open during postulated pipe ruptures.
- Discharge coefficients (Cd) and backpressure values are assumed, so as to produce 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 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.
- The building doors that are non-safety related are conservatively not allowed to open during postulated pipe ruptures.
- 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 mass and energy release data for the MSLB.

The analyses of the primary and secondary pipe ruptures for the U.S. EPR design employ a multi-volume GOTHIC model to examine the long-term containment response. The model includes a mesh-style nodalization of the containment dome region to ensure that the potential for thermal stratification was adequately addressed.

Containment overpressurization events during refueling operations and reduced primary inventory conditions are mitigated by residual heat removal (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 closure.

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 design does not have an automatic containment spray system or containment air coolers for DBA mitigation. Thus, the U.S. EPR design 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, thereby eliminating the need to analyze for this event.

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 as the transport time exceeds 24 hours. Therefore, this is neglected from the analytical models described in Sections 6.2.1.3 and 6.2.1.4.

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 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 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

The principal containment design features impacting the post accident pressure and temperature response are the IRWST, the conversion from a two-room containment into a one-room containment and passive heat sinks inside the containment.

The function of the IRWST is to provide a large reserve of borated water. It is the safety-related source of water for emergency core cooling in the event of a LOCA, and is a source of water for containment cooling and for core melt cooling in the event of a severe accident. The IRWST contains a minimum of 500,000 gallons of borated water and is maintained at a temperature between 60°F and 122°F. The IRWST resides at the lowest point in the containment, and drain paths allow water discharged from the RCS to drain into the IRWST.

Containment heat removal is accomplished by recirculation of cooled IRWST water injected into the RCS where the ECCS absorbs residual energy. The heated ECCS returning to the IRWST is subsequently cooled by LHSI heat exchangers.

The CONVECT system, consisting of rupture and convection foils in the steam generator equipment room ceiling and mixing dampers in the wall between the lower accessible area and the IRWST air space, transforms the two-room containment into a one-room containment. A steel framework at the upper boundary of the steam generator equipment rooms, houses rupture and convection foils with a combined opening area of 870 ft^2.

Large break LOCA (LBLOCA) and small break LOCA (SBLOCA) events establish different requirements for flow cross sectional area which is achieved by the different system components. For the LBLOCA, the full cross sectional area (870 ft^2) is used to limit the pressure peak. As a consequence of low mass and energy release during a SBLOCA event, the opening of all the rupture foils may not occur. For effective steam distribution in the containment, a minimum free-flow cross-sectional area of 450 ft^2 is fulfilled by the convection foils alone.

Eight fail-safe-open mixing dampers, with a total free flow cross-sectional area of 64 ft^2, connect the IRWST air space to the lowest accessible room. These open via spring tension on loss of power in fail-safe mode. They can also be opened and closed manually.

Safety-related doors are required to complete the transformation of the two-room containment into a one-room containment following a break of the pressurizer surgeline inside the pressurizer compartment. These safety-related reactor containment building radiation doors are designed with a shear latch that allows them to open under the differential pressure following the accident, as described in this section.

The passive heat sinks inside the primary containment consist of all painted and unpainted concrete, steel structures and liner for the containment shell and IRWST surfaces. The IRWST heat sinks are exposed to the water in the pool. The remaining heat sinks are exposed to the containment atmosphere. These areas are approximately the same temperature as the containment ambient temperature during normal plant operation. The list of passive heat sinks in the U.S. EPR Containment and their parameters are listed in Table 6.2.1-4—Containment Heat Sink Inventory. Selected heat sinks were not included in the containment pressure-temperature analysis for conservatism. The minimum heat sink surface area for Tier 1, Section 2.1.1.1 is 699,644.3 ft^2.

The design pressure of the containment is 62 psig. 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

Containment and subcompartment design parameters are provided in Table 6.2.1-5—Containment Initial and Boundary Conditions, and Table 6.2.1-4. The general arrangement drawings for the reactor containment are provided in Section 3.8.1. The structural design of the containment and the subcompartments, as well as 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 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 an MSLB.

In the case of a LOCA, reactor coolant at the primary system temperature is the source of the mass and energy released into the containment. 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.

Following a postulated LBLOCA, the rupture foils burst open to join the accessible and inaccessible parts of the containment building. In addition, the mixing dampers open to enable atmospheric circulation within the whole containment. These combined measures establish a global atmospheric natural convection loop within the containment. In the case of the SBLOCA, the convection foils open along with the mixing dampers to enable atmospheric circulation within the whole containment.

The blowdown pressure rise is limited by the free volume of 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.

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. 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. Manual re-alignment of the LHSI to the hot legs suppresses core steaming and the long-term pressure peak is limited by the steam condensation on the passive heat sinks of the containment. The spectrum of postulated LOCA accidents analyzed is provided in Table 6.2.1-1.

In the case of the MSLB, the CONVECT system again allows atmospheric circulation within the whole containment via the rupture foils and the mixing dampers. The affected SG quickly depressurizes leading to full isolation of the main steam and main feedwater systems. A failure of a Main Steam Isolation Valve (MSIV) allows the faulted SG to completely blowdown into the containment. Additional feedwater (MFW or EFW) injected in the faulted SG vaporizes and is released to containment. The spectrum of postulated MSLB accidents analyzed is provided in Table 6.2.1-2. The analyses are terminated when the faulted SG empties and all sources of feedwater to the faulted SG are isolated.

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 for LOCAs is listed in Table 6.2.1-6, Table , and Table 6.2.1-8, and for MSLBs in Table 6.2.1-9.

Table 6.2.1-6, Table , and Table 6.2.1-8 present fifty-one separate cases for LOCA analysis for three postulated break locations. Thirty-eight cases are analyzed using a simple single-node containment model to evaluate the effects of maximum and minimum ECCS flow, break discharge coefficients, and single failures to establish the limiting scenarios for the multi-node analysis. For the LOCA, the limiting containment pressure occurs during the blowdown phase of a double-ended guillotine break in the RCS hot leg piping, with the worst single failure being the loss of one ESF train with an additional ESF train being out service for maintenance.

Table 6.2.1-9 lists 54 cases for the MSLB, with 14 break sizes ranging from the double-ended guillotine break to the 0.005 square foot break area, and power levels from 100 percent to 0 percent of rated thermal power (RTP). The peak containment pressure results from the assumed double-ended guillotine MSLB with a failure of one MSIV at 20 percent RTP.

The requirements of 10 CFR part 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 part 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 0.5 percent of rated thermal power, or 1.005, is applicable to the core power for the U.S. EPR design. 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 7.2b 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 6.2.1-5 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 hot leg break LOCA with the single active failure of one train of ECCS. A summary of the results of the containment pressure and temperature analyses for the spectrum of postulated LOCAs is listed in Table 6.2.1-6, Table , and Table 6.2.1-8, and for MSLBs in Table 6.2.1-9.

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 122°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. A graph illustrating 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 mass and energy 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 conservative assumption is applied with all break locations.

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 post-accident 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 fluid discharge, that result from equipment failures and from events and conditions outside the containment. The containment internal compartments are designed to 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 postulated pipe breaks. The subcompartment walls are challenged by the differential pressures resulting from a postulated high-energy line break (HELB) within individual compartments. These rooms are arranged to allow venting of HELBs to prevent differential pressures from reaching the structural limits of compartment walls.

Fluid systems are considered high energy when, during normal plant conditions, the systems are operated or maintained 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 design, 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 subcompartment, system components, and supports.

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. Appendix 3E presents the Reactor Building critical sections and the structural evaluations associated with them. Table 6.2.1-10 lists all rooms adjacent to a critical section which contain a HELB.

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.

Subcompartments are evaluated based on the mass and energy discharge from each high-energy line in the compartment. Table 6.2.1-11 lists the subcompartment, HELB line, conditions and energy discharge for the line with the highest energy discharge in each of subcompartments. Table 6.2.1-12 lists the mass and energy discharge rates for each of the limiting lines identified in Table 6.2.1-11. Subcompartments without high-energy lines are omitted from further analyses.

The U.S. EPR principal containment subcompartment design parameters include the volume and vent area for each subcompartment. 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) are conservatively treated. The doors are initially modeled in the closed position and remain closed unless classified as a safety grade door listed in Table 6.2.1-13.

6.2.1.2.3 Design Evaluation

High-energy lines identified for the subcompartments are compared based on the full-power operating conditions. The mass flux from the postulated pipe rupture is then calculated using the Moody and Henry-Fauske critical flow models. With the cross-sectional area for each of the high-energy lines known, the energy flow is calculated. The highest energy flow for each subcompartment will be selected for subsequent subcompartment analyses. The critical flow rate is held constant for most of the HELB calculations. Some subcompartment calculations shorten the duration based on a limited inventory (e.g., isolation of one side of the break), while others use the CRAFT2 or RELAP5 system analysis codes to determine the mass and energy release as a function of time. These mass and energy rates are listed in Table 6.2.1-14 and Table 6.2.1-15, respectively.

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.

Two detailed GOTHIC models are used for conducting the subcompartment analyses. The GOTHIC models define a subcompartment as any fully or partially enclosed volume in the primary containment that would limit the flow of fluid to the main containment volume. Large subcompartments are further divided into rooms at the elevations specified by the general arrangement drawings provided in Section 3.8.1. One GOTHIC model includes a node for each room in the equipment space, while the other includes a node for each room in the service space and pressurizer cavity. Together, they model each individual subcompartment and connect them hydraulically by junctions or flow paths.

The analysis approach is to inject the mass and energy release from the break into the relevant containment subcompartment. The appropriate GOTHIC model is selected based on the location of the break. The HELBs listed in Table 6.2.1-11 are either analyzed using GOTHIC or addressed by the analysis of a symmetrical room.

Each room, represented by a single node that yields a pressure increase greater than 5 psi, was subdivided into multiple cells, as shown in Figure 6.2.1-45 for the +45 ft Room 3, so that the initial peaks or blowdown peaks inside the rooms are fully captured. These nodalization sensitivity studies showed that the pressure response either decreased or increased by less than 1 psi as a result of more rigorous nodalization. The principal containment subcompartment design parameters are provided in Table 6.2.1-16 and Table 6.2.1-17, which include the room volume and vent area for each subcompartment where the pressure increases more than 5 psi.

Assumptions for the distribution of mass and energy release are biased towards maximizing the subcompartment pressure. The vent flow behavior through the junctions in the model is based on a homogenous mixture in thermal equilibrium with 100 percent water entrainment. GOTHIC code options are used to force thermal equilibrium and disable drop to liquid conversion. A small break drop size is used to obtain velocity equilibrium. The NRC-accepted homogeneous equilibrium model (HEM) critical flow model for air-steam-water mixtures is used for the vent and other downstream junctions. The GOTHIC compressibility option is also used. The compressibility option has the effect of slightly increasing the loss coefficient because of increased density of upstream fluid when pressure drop across the junction becomes large.

Initial atmospheric conditions are chosen to maximize differential pressures. NRC-accepted initial conditions with air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity are used in each node of the GOTHIC models.

Peak pressure results are presented in Table 6.2.1-18 for subcompartments with pressure increases greater than 5 psi. Plotted pressures are shown in Figure 6.2.1-46 to Figure 6.2.1-64, where the pressure increases greater than 5 psi that are adjacent to a critical section. The results of these evaluations are used in Section 3.8.3 to show that the subcompartments can withstand the applied loads, including the subcompartment pressures, and remain in allowable limits. The structural load calculations include a minimum load of 5 psi on the critical sections with an additional factor of 1.4 applied to the peak pressure predictions from these analyses prior to their use as inputs in the design of the structures.

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

The containment pressure response to a LOCA in a U.S. EPR is similar to that of a conventional Pressurized Water Reactor (PWR) with a large dry containment. However, containment sprays are not an engineered safety feature used to mitigate the containment pressure response in the U.S. EPR design. Termination of a LOCA event is achieved by quenching core region steaming with pumped safety injection.

Following steam quench, hot liquid leaving the reactor coolant system drains to the IRWST, which is attached to an LHSI heat exchanger cooling chain providing the ultimate heat sink.

For the U.S. EPR design, the spectrum of LOCA breaks analyzed includes a range of cold leg pump discharge, cold leg pump suction, and hot leg breaks, ranging from a three-inch SBLOCA up to the largest postulated double-ended guillotine break. The double ended guillotine break of a large RCS pipe is the most limiting event for the purposes of containment pressure because it adds the greatest mass and energy to containment in the shortest period of time.

From the perspective of the reactor coolant system, the course of an LBLOCA is divided into five phases characterized by distinct phenomena:

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 flowrate from the RCS to containment varies depending upon the nature, size, and location of the break. Core cooling during this phase is accomplished by film boiling heat transfer from 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.

Coolant released from the primary system causes an increase in containment steam mass, which in turn increases pressure and temperature. In response to the initial pressure wave and increase in temperature, rupture and convection foils located above the U.S. EPR equipment rooms (i.e., compartments containing the steam generators and reactor coolant pumps) open, exposing the released mass and energy to the full containment volume. Mixing dampers located low in the containment, on the walls separating the containment's accessible area and the IRWST air space, also open to complete a flow circuit that allows the air/steam mixture to circulate. Containment pressure rises until pressure between the primary system and the containment equalizes. This is considered the end of the blowdown phase.

Following blowdown, a refill period occurs where the SIS provides sufficient liquid to fill the reactor vessel lower head and plenum regions. Within the reactor pressure vessel (RPV), residual steam and hot wall effects generate steam in the lower plenum. Some of this steam escapes through the break. The speed with which the lower plenum refills depends on the total coolant delivery rate, the steam/water interfacial interactions, and the break size and location. The refill phase ends when the water level reaches the core inlet elevation.

During the next phase, reflood, the water level rises from the bottom to the top of the reactor core. Simultaneous gravity-driven reflooding of the core and the interaction of cold ECCS water (both MHSI and LHSI) with steam in the cold legs and downcomer cause both manometric- and condensation-driven oscillations in flow rate and pressure. High reflood rates from accumulator discharge rapidly drive water toward the hot fuel surface, producing steam. As the steam expands, the water is pushed away from the fuel surface. The gravity head of the water in the downcomer pushes back on the steam, returning coolant to the core where the resident steam can condense. These manometric oscillations slowly dampen as the quench front progresses through the core. Separately, steam leaving the core and traveling through the intact cold legs meets subcooled accumulator and safety injection coolant, and the subsequent condensation of the steam decreases the local pressure, which impacts delivery rates into the reactor vessel.

If the pipe rupture is in one of the RCS hot legs, 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, 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 post-reflood phase begins following core quench. LHSI coolant temperatures rise as the LHSI heat exchanger counters the increase in the IRWST temperature as hot water leaves the RCS and flows back into this source for LHSI. Nucleate boiling heat transfer in the core produces a two-phase mixture that rises above the core, into the upper plenum, hot legs, and steam generators. For cold leg breaks, the bulk of the remaining fluid sensible heat in the secondary-side of the steam generators is removed by the two-phase mixture residing in the steam generator tubes. This causes superheated steam to exit the steam generator primary side. The remaining RCS structure sensible heat is released to the circulating coolant and delivered to the containment during this LBLOCA phase. For hot leg breaks, heat removal from these sources can still occur; however, the break location in the hot leg causes a significant bypass of coolant away from the steam generators and intact loop piping.

Subsequent steam flow through the remaining RCS piping is sufficient to keep the piping clear of accumulating liquid, including the horizontal segment approaching the reactor coolant pump suction (i.e., crossover leg). This steam and water mixture is carried to the break location as in the reflood phase, but in a manner described as a "boiling pot." When steam flow decreases to a level at which it no longer can prevent the filling of the crossover leg, the post-reflood phase ends.

Like the post-reflood phase, the decay heat phase is characterized as a simple "boiling pot" in which decay heat is the only significant heat source remaining in the RCS. In contrast to the post-reflood phase, it is expected that fluid flow through the RCS loops is significantly reduced by lower steam generation coming from the core and the formation of loop seals in the horizontal piping segment near the reactor coolant pump suction.

For the U.S. EPR design, a manual re-alignment of at least 75 percent of the LHSI from the cold leg to the hot leg injection location takes place early in this final LBLOCA phase (about 60 minutes after the initiating event). This re-alignment serves both as a mechanism for removing core decay heat, leading to complete steam suppression, and for maintaining core boron concentrations below the threshold concentration for precipitation.

Mass and energy releases are impacted in two ways by re-alignment of SI: coolant mixing in the upper plenum and core region and condensation efficiency between steam flows and safety injection in the hot legs and upper plenum. With regard to steam condensation, this phenomenon reduces the overall steam flow through the loops to the break. Safety injection water penetrates the upper plenum and the periphery of the core below the hot leg nozzle, providing emergency core coolant. During this later LBLOCA phase, the hot-leg break is mitigated, such that the reduction in LHSI from the possible loss of one train to the break is not penalizing to containment pressure (or fuel cladding temperatures). In addition, there is efficient ECCS mixing since the colder safety injection coolant falls along the core periphery before flowing back up through hotter fuel assemblies.

A break in the hot leg piping is shown to produce the highest containment pressure. A double-ended guillotine break of this pipe allows the initial RCS mass and energy to enter the containment early in the transient, before the passive heat sinks of the containment can effectively absorb the energy addition. Once the reflood phase begins, the mass and energy release decreases and the building heat sinks begin to reduce the pressure As break size decreases, the mass and energy release slows, but residence time increases. This allows additional energy to be transferred from the core and SGs.

The limiting break configuration for the hot leg break scenarios includes a double-ended guillotine (DEG) break with minimum available safety injection with offsite power available. Figure 6.2.1-10, Figure 6.2.1-11, Figure 6.2.1-12, and Figure 6.2.1-13 provide the short- and long-term pressure and temperature results for the limiting hot leg scenarios. The temperature profile corresponds to the temperature in the hottest node in the containment dome. Figure 6.2.1-36 shows the temperature profiles in the dome region at various elevations. This figure shows that thermal stratification does not occur in the long term. Figure 6.2.1-37 provides temperature profiles in rooms below the dome. The rate of pressure increase is proportional to the rate at which the energy is added. The mass and energy 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 determines the containment pressure response for a LOCA in the hot leg.

After the blowdown phase for the hot leg breaks, the RCS is essentially in pressure equilibrium with the containment. As the system begins to refill, the mass and energy effluent to containment decreases. 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 begins to decrease. Accumulator injection quickly quenches the core, expelling the stored energy to containment. Thereafter, the energy addition to containment due to core boiling is only that caused by decay heat. While the SIS injects into the cold legs, little flow travels through the SGs to the break. Manual re-alignment of a majority of the LHSI from the cold leg to the hot leg injection location reduces the amount of ECCS available for core cooling as the realigned LHSI spills on the floor. Containment pressure starts to increase until the MHSI and remaining LHSI can completely suppress the production of steam.

Figure 6.2.1-10 through Figure 6.2.1-13 show the trends of containment pressure increase during the blowdown phase. At the end of the blowdown phase, the containment pressure peaks and begins to decrease until re-alignment of the LHSI from the cold leg to the hot legs. A blowdown peak of 70.89 psia occurs at 32.0 seconds. Following ECCS re-alignment, the containment pressure begins rise until the remaining LHSI and the MHSI still injecting in the cold leg can suppress core steam production. The post-reflood peak of 48.2 psia occurs at 8302 seconds; the containment pressure continues to decrease, reaching 34.7 psia by the end of the analysis at 24 hours.

A break in the cold leg pump suction piping does not produce the limiting blowdown peak pressure of the containment. The resistance of the pumps delays the blowdown as compared to the hot leg break scenarios, and the RCS depressurizes at a slower rate compared to the hot leg break. As a result, the accumulator injection is delayed compared to the 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. During the reflood phase of the event, steam generated in the core superheats. It approaches saturated conditions as the core water level increases. Steam from the core traverses the SG, absorbing additional energy from the secondary system. 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. A reduction in the steam flow from the decreasing decay heat allows the crossover legs to begin to fill and form loop seals. The most penalizing condition occurs when the three intact loops no longer provide a vent path to the break such that steam from the core flows to the containment by a path that circumvents cold ECCS injection water. This condition causes a further increase in the containment pressure until the manual switchover of at least 75 percent of the LHSI to the hot legs.

The limiting break configuration for the cold leg pump suction (CLPS) break scenario is a double-ended split (DES) break of break size of 0.6 with minimum safety injection supplied to the two cross-connected intact loops with offsite power available. Figure 6.2.1-14 through Figure 6.2.1-17 provide the pressure and temperature results for the most limiting cold leg pump suction scenarios. The temperature profile corresponds to the temperature in the hottest node in the containment dome. Figure 6.2.1-38 shows the temperature profiles in the dome region at various elevations. This figure demonstrates that thermal stratification does not occur in the long term. Figure 6.2.1-39 shows the temperature profiles in different rooms below the dome area.

A blowdown peak of 67.3 psia occurs at 34.2 seconds. The containment pressure begins rise following refill until ECCS injecting in the hot legs can suppress core steam production. The post-reflood peak of 70.5 psia occurs at 3600 seconds when at least 1720 gpm of each of the available LHSI trains is aligned to the hot legs. The containment pressure continues to decrease, reaching 32.5 psia by the end of the analysis at 24 hours.

A break in the cold leg pump discharge (CLPD) piping produces the lowest peak containment pressure. The blowdown phase is similar in duration to the cold leg pump suction break and produces a similar containment pressure response. However, the reflood and post-reflood phases of the cold leg pump discharge event are less limiting than the pump suction break. Unlike the pump suction break, coolant delivery to the loop seal piping segment is significantly reduced because of a weir in the U.S. EPR reactor coolant pump design. As a result, the formation of loop seals is not likely until after re-alignment of the LHSI to the hot legs. The steam that goes through the intact loop must pass pumped injection locations on the way to the reactor vessel (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 lower enthalpy.

The limiting break configuration for the cold leg pump discharge break scenario is a double-ended split break of break size of 0.8 with minimum available SIS and postulated LOOP. Figure 6.2.1-18 through Figure 6.2.1-21 provide the pressure and temperature results for the limiting cold leg pump discharge scenarios. The temperature profile corresponds to the temperature in the hottest node in the containment dome. Figure 6.2.1-40 shows the temperature profiles in the dome region at various elevations. This figure shows that there is adequate mixing in the dome region following the LOCA accident; therefore, thermal stratification does not occur. Figure 6.2.1-41 shows the temperature profiles in different rooms below the dome area. A blowdown peak of 66.2 psia occurs at 23.6 seconds. The containment pressure rises following refill until ECCS suppresses core steam production. The post-reflood peak of 68.5 psia occurs at 2280.1 seconds, and the containment pressure continues to decrease, reaching 32.6 psia by the end of the analysis at 24 hours.

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 evaluate the smaller breaks to confirm that sufficient vent area between the equipment area and the accessible area exists. For these smaller breaks, the rupture foils in the CONVECT system are conservatively deactivated in the GOTHIC model and venting is delayed until the containment temperature at the pressure equalization ceiling (PEC) exceeds the temperature setpoint of the convection foils. The break location studies include a break of the pressurizer surgeline inside the pressurizer compartment to confirm venting from the pressurizer compartment to the accessible space is adequate. Critical parameters for the six safety-related doors credited in this analysis are included in Table 6.2.1-13.

6.2.1.3.1 Mass and Energy Release Data

Blowdown mass and energy release data are presented in Table 6.2.1-19 through Table 6.2.1-21 for the limiting cases at each of the three break locations analyzed. The mass and energy at break in the above tables represent both short-term and long-term releases. For the short-term period, the phasic mass and energy release on both sides of the breaks (vessel side and steam generator side) are presented. For the long-term period, mass and energy releases are calculated internally by the GOTHIC with one break junction representing RCS connection to containment. As a result, only one set of data (RV side) is provided.

To maximize the containment peak pressure and temperature, the U.S. EPR LBLOCA and SBLOCA analyses use conservative assumptions that maximize the mass and energy 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 mass and energy into the containment. In addition, the analyses assume that doors between different rooms in the containment remain closed during the entire transient. To provide conservatism, 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 6.2.1-22—Input Summary for Mass and Energy Release provides a summary of the initial conditions for the calculation of the mass and energy release.

The blowdown phase mass and energy release rates are calculated by the thermal-hydraulic analysis code, RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis (Reference 3). The NRC has reviewed and approved this code as meeting the requirements of 10 CFR part 50, Appendix K for pressurized water reactors with recirculating SGs. These 10 CFR part 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 mass and energy 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 part 50, Appendix K requirements is consistent with NUREG-0800 and ANSI/ANS-56.4. This adjusted model is used to calculate the mass and energy released to containment from the beginning of the long-term cooling phase, or time of core quench.

Post-reflood mass and energy release rates are referred to as long-term LOCA and are determined by the GOTHIC Version 7.2b computer code, as presented in Section 6.2.1.1.3, using the model described in Section 6.2.1.3.4.

6.2.1.3.2 Energy Sources

The sources of stored and generated energy used in all of 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 part 50, Appendix K method.

6.2.1.3.3 Description of Short-Term Mass and Energy Release Model

A description of the RELAP5/MOD2-B&W model used to determine the mass and energy 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. All significant correlations are discussed in these reports.

The short-term break flow is calculated using the Moody critical flow model, which conservatively over-predicts the discharge rate in comparison to experimental data. The thermal energy equations in RELAP5-BW are expressed in terms of phasic internal energies. Therefore, a kinetic energy term is explicitly added to the energy release model in GOTHIC to capture the total energy contribution to the containment.

After the blowdown phase, the reactor vessel lower plenum is refilled by the ECCS. For LOCA mass and energy release rate calculations, a conservative refill period is one that is minimized to advance the heat transfer from the fuel elements to the fluid in the core.

The IRWST pool in the U.S. EPR is designed to be cooled and mixed by recirculating water through the heat exchanger of the RHR system. The containment analysis conservatively assumes there is no heat transfer between the containment vapor and IRWST liquid regions for the entire transient.

The transition between the short-term model and the long-term model is determined by the formation of loop seals for the cold leg pump suction breaks or the initiation of hot leg injection for the cold leg pump discharge and hot leg breaks.

6.2.1.3.4 Description of Long-Term Mass and Energy Release Model

Once the core is quenched, the LBLOCA proceeds into the long-term cooling phase of the analysis (post-reflood and decay heat phases). Reactor vessel coolant is in a quasi-steady-state condition, characterized by the vessel level recovered to the RCS loop nozzle elevations and the ECCS injection maintaining the core covered so that core decay heat and sensible heat are being removed. The long-term mass and energy releases are modeled in GOTHIC by adding a node to represent the RCS, and decay heat and sensible heat were modeled by the code as multiple heater components.

GOTHIC models are provided for the IRWST recirculation, the emergency core cooling function of the ECCS (SIS/RHR pumps and RHR heat exchangers), and containment heat removal by steam condensation/convection on the containment passive heat sinks.

The long-term sources of energy are:

- Core decay heat.
- Primary system fluid stored energy.
- Primary system passive metal stored energy (including core metal stored energy).
- Secondary system stored energy (fluid + metal).
- Safety injection pump heat addition.

The final temperature of the secondary fluid and metal is forced to be equal to or less than the saturation temperature in the containment at 24 hours. The release of the secondary stored energy continues until the total energy is transferred to the containment. The treatment of each of these sources in the long-term GOTHIC model is described in “Applicability of AREVA NP Containment Response Evaluation Methodology to the U.S. EPR for Large Break LOCA Analysis” (Reference 15).

In the long-term phase, the majority of LHSI discharge is switched over to the hot legs sixty minutes following the break to terminate core steaming. Re-alignment of LHSI to hot legs might recover an LHSI train that otherwise is delivered directly to the break if it is located in the cold leg pump discharge piping.

This injected water flows into the upper plenum, partially condensing any steam in the region. The cooler safety injection water then falls as a plume into the core and passes through several peripheral fuel assemblies located below the hot leg nozzles, suppressing steaming from these fuel assemblies.

As this water falls into the core, much of it traverses into the adjacent fuel assemblies and reduces boiling in these adjacent assemblies. The result is vigorous circulation in the core that reduces and eventually suppresses net steaming from the core. The water that is not drawn into the adjacent assemblies continues down into the lower plenum. The warmed safety injection water leaves the RCS via a path leading into the lower head, the downcomer, and to the cold leg break location.

6.2.1.3.5 Single Failure Analysis

The effect of single failures of system components on the mass and energy releases is included in these LOCA analyses. For cases where offsite power is unavailable, the failures considered include the failure of one emergency diesel generator that causes the loss of one complete train of ESF equipment, failure of a component in the CONVECT system to open, and the failure of safety-related doors in the pressurizer compartment. For cases where offsite power is available, the failure of the emergency diesel generator is replaced by the failure of a LHSI train, including the pump and associated heat exchanger.

An additional analysis is performed to show that allowing non-safety doors to open following a DBA does not have an adverse affect on the circulation patterns in the containment.

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 part 50, Appendix K.

6.2.1.3.7 Energy Inventories

Inventories of the energy transferred from the primary and secondary systems to the containment, as well as the energy remaining in the primary and secondary systems for the limiting cold leg pump suction break, is provided in Table 6.2.1-23.

6.2.1.3.8 Additional Information Required for Confirmatory Analysis

System parameters and hydraulic characteristics needed to perform confirmatory analysis are provided in Table 6.2.1-22 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

Secondary pipe ruptures inside a reactor containment structure can produce significant releases of high energy fluid to the containment environment, producing high containment temperatures and pressures. Reactor trip is actuated automatically on high rate of SG pressure decrease, low SG pressure, or high containment pressure.

Engineered safety functions that mitigate the MSLB event are main steam isolation and main feedwater isolation. Isolation of the main steam lines prevents inventory from the three intact steam generators from exiting the break. Isolation of the main feedwater system limits the mass introduced to the faulted steam generator. Actuation of emergency feedwater can be a benefit or detriment depending on the break size and is discussed in Section 6.2.1.4.1.3.

The rapid depressurization of the faulted steam generator increases the heat transfer across the steam generator tubes and can cause a severe cooldown of the reactor coolant system. Due to the negative moderator temperature coefficient and negative Doppler temperature coefficient, the cooldown of the reactor coolant system might be sufficient to insert positive reactivity in excess of the inserted control rod shutdown worth, thereby leading to a post-reactor trip return to criticality.

The mass and energy release following a secondary pipe rupture depends upon the configuration of the plant's main steam system, main feedwater 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.

The general plant system response is similar for all secondary system pipe ruptures. However, the MSLB always produces the limiting mass and energy release rate for a secondary pipe rupture because of three important factors:

1. Break size.
2. Integrated energy.
3. Break effluent conditions.

The SG Main Feedwater (MFW) nozzle has an ID of 1.453 ft or an area of 1.658 ft^2, which is greater than the integral flow restrictor at the steam generator exit nozzle (1.4 ft^2). However, the critical flow area for a Main Feedwater Line Break (MFWLB) is determined by the MFW ring area of 0.916 ft^2. The break spectrum selected for evaluating the MSLB envelopes the MFWLB. This provides a limiting sequence of events for both break locations.

The integrated energy of the break fluid is always greater for an MSLB than the MFWLB. Once the SG tubes uncover during an MSLB, any additional feedwater injected into the SG prior to complete isolation of the feedwater system absorbs energy from the primary system. This additional energy is deposited directly in containment as the SG inventory flashes and exits the break at the steam nozzle. In the case of an

MFWLB, any MFW that is injected into the containment through the break does not have an opportunity to acquire additional primary system energy. Thus, the integrated energy for an MSLB is always greater than that of the MFWLB.

The break effluent for an MSLB progresses from single-phase steam to two-phase mixture, and then back to a single-phase vapor release. The MFWLB event initially produces liquid discharge, then progresses to a two-phase release and, finally, to a single-phase steam discharge. Regardless of the break effluent progression, both scenarios cause the complete blowdown of a single steam generator, as well as additional main feedwater, either to the containment (MFWLB) or to the SG (MSLB), until terminated.

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.) The scenarios include an evaluation of potential single active failures. The main feedwater (MFW) system includes redundant isolation valves, which preclude an over-feeding failure. In all scenarios, isolation of the MFW system uses the longest delay time and the slowest isolation valve closure time. These scenarios use the Diffusion Layer Model for condensation and the default option for revaporization. This enables the GOTHIC code to calculate the fraction of condensate that is revaporized.

Containment pressure and temperature response to an MSLB depends upon the amount of break effluent that enters the containment atmosphere as steam, and whether that steam is saturated or superheated. During the depressurization of the SG, two phenomena can 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 analytical models are biased to reduce the amount of entrained liquid that exits the SG. Any entrained liquid that exits the break is conservatively converted to steam in the analysis. The second is the rapid voiding of the SG that causes a liquid level swell that extends to the break location. This causes the discharge of saturated liquid. The swelling of liquid mixture 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 mass and energy released from the MSLB and include 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 1.3 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 mass and energy release following an 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 generally produces 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 mass and energy release, HZP and power levels spanning the operating range are analyzed.

6.2.1.4.1.2 Main Feedwater System Design

The rapid depressurization following an 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 close rapidly to limit feedwater addition during the event. The MFW piping layout downstream of these isolation valves impacts the event because it affects the volume of liquid in the feedwater piping that can enter the 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 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, so it decreases the steam pressure reducing the flowrate 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 mass and energy release during a MSLB, so bounding conditions are provided in all MSLB scenarios. During periods of 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 that 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 an MSIV, 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 cools the SG steam inventory, which decreases the driving pressure of the break flow. At the same time, the EFWS water absorbs heat from the SG tubes and other metal structures. This provides an additional transport mechanism for energy to the containment.

In a large double-ended guillotine break, the peak containment pressure and temperature occur early in the transient. Therefore, the introduction of cool EFWS water decreases the driving pressure of the break flow, thereby slightly reducing peak containment pressure and temperature. In small split-break events, the peak containment pressure and temperature occur much later. The additional mass and energy released to containment over time due to the EFWS water increases peak containment temperature and pressure. Because of these competing effects, EFWS has only a small influence on MSLB containment temperature and pressure response.

Emergency feedwater is initiated on either low SG level or a safety injection signal coincident with LOOP. However, the LOOP is not credited during this event, so in the MSLB analysis, the emergency feedwater actuation only occurs 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 SGs.

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. Because peak containment pressure and temperature occur before 30 minutes, emergency feedwater flow is available for the duration of the analysis.

6.2.1.4.1.4 Postulated Break Size, Type, and Location

Releases are analyzed for five MSL breaks: the double-ended guillotine break and break sizes of 1.0, 0.7, 0.52, and 0.3 square feet in area. Each of the break sizes are analyzed at seven initial power levels. Sensitivities on large split break sizes are performed only at lower power levels because the limiting case occurs at low power. Additional sensitivities of very small break sizes are also performed only at low power. The very small break size sensitivity studies are performed because the affected SG still has significant inventory at the time of EFWS isolation at 30 minutes. The break sensitivities are performed to verify that the limiting break size is identified.

Each SG is equipped with a flow orifice that limits the effective area for the MSLB. Although the area of the main steam line is 4.1 ft^2, the effective break area after main steam isolation is no greater than the flow restrictor throat area of 1.4 square feet.

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 reduces the rate of blowdown, but it does not significantly change the total release of energy to the containment. Therefore, piping pressure drop from the affected SG to the break location is ignored in all analyses. Because the location of the break within containment can affect the containment temperature and pressure response, break locations are analyzed in both the accessible and inaccessible areas.

6.2.1.4.1.5 Availability of Offsite Power

The U.S. EPR design 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 mass and energy 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

Six potential single failure scenarios for the MSLB event are considered:

1. Failure of the MFW pump to trip.
2. MFW isolation valves failure.
3. MFW control valve failure.
4. MSIV failure.
5. CONVECT system failure.
6. Containment door failure.

The most severe single active failure is the failure of an MSIV on the main steam line of the affected steam generator. An MSIV failure would provide additional fluid that is released to the containment via the break. This fluid comes from the blowdown of the steam piping between the break location and the isolation valves in the intact loops. This single failure is more severe than the failure of an MSIV on an intact loop because of the volume of the steam line that is isolated.

The single failure of the MSIV on the affected steam generator is more severe than any failure in the MFW system because of the redundancy of the valves in the main feedwater system. A failure in the CONVECT system would involve only a single foil, not the entire CONVECT system, and therefore a single failure of the CONVECT system is not as limiting as the MSIV failure. A containment door can not be the limiting single failure because no containment doors are credited in the analyses.

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 an MSLB, this energy is 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 is transferred to the primary coolant as the 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 containment response analyses.

The RELAP5/MOD2-B&W computer code incorporates a full spectrum of heat transfer modes, including single-phase convection, nucleate boiling, critical heat flux, transition film boiling, film boiling, and condensation. The appropriate heat transfer correlation is determined by RELAP5/MOD2-B&W for each heat conductor in the model based on the calculated thermodynamic conditions at each time step.

6.2.1.4.2 Description of Blowdown Model

A description of the RELAP5/MOD2-B&W model used to determine the mass and energy 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). Significant correlations are discussed in these reports.

Figure 6.2.1-42 presents the results of the SG pressure blowdown of the limiting MSLB case. Figure 6.2.1-43 and Figure 6.2.1-44 present the integrated break mass and integrated break energy. Integrated break mass and energy increase rapidly upon initiation of the event. The rate of mass and energy release to containment decreases as the SG blows down. Since no EFW is conservative for the limiting case, the integrated mass and energy release to containment remains constant after SG blowdown.

6.2.1.4.3 Containment Response Analysis

The containment response to postulated MSLBs was analyzed with GOTHIC Version 7.2b (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 provided in Section 6.2.1.1.3.

6.2.1.4.3.2 Mass and Energy Release Data

Table 6.2.1-24—MSLB Mass and Energy Release Data presents the mass and energy release data used to determine the containment pressure and temperature responses for the limiting MSLB, a double-ended guillotine break of the main steam line at 20% RTP concurrent with the single active failure of the MSIV on the faulted SG steam line. A break location sensitivity study showed that a break in the accessible space outside the SG towers produces the most limiting containment pressure.

Feedwater isolation for the full and partial double-ended guillotine breaks depends on signals generated by the ESF instrumentation. The feedwater flow rates used in the analyses credit the longest isolation valve stroke time of 40 seconds. Valve leakage is not considered in the analyses because it is bounded by the emergency feedwater injection. Table 6.2.1-25—MSLB Reactor Trip and Isolation Signal Summary presents the reactor trip, isolation trips, and foil opening trip times for each MSLB case analyzed.

6.2.1.4.3.3 Containment Pressure and Temperature Results

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

The worst single active failure for the MSLB is the loss of the MSIV on the main steam line of the affected steam generator. Because the U.S. EPR design includes redundant safety-related feedwater isolation valves, the failure of the main feedwater isolation valve is not specifically analyzed. However, as an additional conservatism, the MFW valve with the longest stroke time is used for isolation. Feedwater flow is credited from the beginning of the transient until the isolation valve is fully closed.

The RELAP5 model used for this purpose includes a partial representation of the main feedwater system. The piping downstream of the control and isolation valves is included in the RELAP5 model. Some, but not all, of the piping upstream of the control and isolation valves is modeled explicitly. Consistent with the description in Section 6.2.1.4.1.2 this model allows injection of additional feedwater into the steam generator resulting from flashing or swelling of the water contained in the unisolated section of the main feedwater piping.

The loss of one MSIV is assumed for the spectrum of break sizes and power levels analyzed. As illustrated in Table 6.2.1-9, a full double-ended guillotine break of the main steam line in the accessible space outside the SG towers at 20 percent RTP produces a peak pressure of 66.4 psia. This is less than the design pressure of 62.0 psig. This case represents the peak calculated containment pressure for the spectrum of breaks analyzed.

The calculated containment vapor temperature exceeds the saturation temperature for some cases for a short period of time. While the analyses show the vapor space is superheated, the containment walls and structures are not. The primary mode of heat transfer during this time period is condensation on the building surfaces. Therefore, the building surface temperature is 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 is above the saturation temperature for approximately 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 the S-RELAP5 code. ICECON is a variant of the CONTEMPT/LT-022 containment code series. The tabular mass and energy 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 mass and energy releases to the containment are described in Section 15.6 and conform to the realistic ECCS evaluation models of 10 CFR 50.46(a)(1)(i).

6.2.1.5.2 Initial Containment Internal Conditions

The U.S. EPR containment in ICECON is modeled as a dry containment with only one compartment: the drywell compartment. The reactor vessel and primary system are represented as a mass and energy source to the containment volume. The containment building is modeled as being in contact with the containment volume on the interior side and the containment annulus on the exterior side.

The dominant phenomenon of interest related to the ICECON containment model is the effect of containment pressure on PCT. Containment pressure is treated statistically in the RLBLOCA methodology by ranging the containment volume from the best estimate value to the maximum possible free volume. For each case in the RLBLOCA analysis, the initial values for the containment volume conditions are representative of 100 percent rated thermal power and a pressure of 14.664 psia. The containment volume temperature is sampled between 100°F and 131°F. The containment vapor and liquid, including the liquid in the IRWST, are modeled at the same sampled temperature. The relative humidity of the vapor region is 100 percent.

A containment annulus temperature of 45°F and relative humidity of 70 percent are assumed and modeled within the ICECON module. The 45°F temperature is the minimum winter design value for the containment annulus based on site design envelope temperatures. The heat transfer coefficient for heat transfer to the containment annulus is 5.0 Btu/hr-ft^2-°F. The value of 5 Btu/hr-ft^2-°F is used for free convection in air and is the upper range of values for a free convection application, as stated in *Principles of Heat Transfer* (Reference 16). The containment pressure response using 1.0 Tagami plus 1.0 Uchida was compared to 1.7 Uchida alone. For the U.S. EPR design, using 1.7 Uchida for condensation heat transfer produces a lower containment pressure than 1.0 Tagami + 1.0 Uchida. In addition, the 1.7 Uchida coefficient was found to be conservative with respect to experimental data. Therefore, 1.7 Uchida is used to calculate the minimum containment pressure.

The containment purge system can be used during normal plant operation. However, the containment purge valves are shut after the containment pressure reaches the high containment pressure setpoint. The inclusion of containment purge in the minimum containment pressure analysis was considered. However, ICECON cannot model the purge system's valve closure when the high containment pressure setpoint is reached and therefore the system is not accounted-for in the minimum containment pressure analysis. A modified GOTHIC containment model determined that including the containment purge subsystem produced only a slight decrease in containment pressure, which translates to an insignificant effect on the PCT results.

6.2.1.5.3 Other Parameters

The RLBLOCA methodology sets the initial containment pressure by sampling the containment volume. The combined containment free volume is 2.888 x 10^6 ft^3 which represents the sum of the nominal containment free volume and the nominal IRWST water volume, and is the lower bound of the containment volume sampling range. The sum of the combined containment volume and the internal structure volume yields the maximum containment free volume, 3.934 x 10^6 ft^3, the upper bound of the containment volume sampling range. The maximum containment free volume is conservative because a lower containment backpressure results in the highest calculated peak cladding temperature.

Heat transfer between the IRWST water and containment vapor is treated in a conservative manner. The IRWST is assumed to be well mixed, so the liquid temperature at the interface between the IRWST water and the containment vapor space is the bulk liquid temperature. This neglects heating of the surface water and maximizes the temperature differential for heat transfer. 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.

Developing the heat sinks in the ICECON model begins with the heat structure groups in the U.S. EPR GOTHIC containment model. Assumptions used in the GOTHIC model are then assessed for applicability to a conservative minimum back-pressure calculation. 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.” Thus, an additional heat sink representing the uninsulated systems and components is incorporated into the ICECON model. However, the volume impact from this additional heat sink is not considered in the combined containment free volume or the maximum containment free volume calculations. An additional assumption increases the nominal heat transfer surface areas by 10 percent to increase the energy removed from the containment atmosphere, which is consistent with a conservative PCT calculation.

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, and testing of safety-related reactor containment building radiation doors, are also included in the Technical Specifications (Chapter 16).

6.2.1.7 Instrumentation Requirements

Refer to Section 7.3 for engineered safety features instrumentation. Refer to Section 11.5 and Section 12.3.4 for radiation monitoring instrumentation.

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