19.2 Severe Accident Evaluations

19.2.1 Introduction

This section describes the U.S. EPR features utilized to prevent and mitigate a severe accident, the performance of the containment as a fission product barrier during a severe accident, accident management considerations and an evaluation of severe accident mitigation design alternatives. Technical bases and the analytical methodology defining AREVA NP's approach to severe accident safety issue resolution are detailed in the AREVA NP Topical Report ANP-10268P, "U.S. EPR Severe Accident Evaluation" (Reference 1). This methodology relies on the Modular Accident Analysis Program (MAAP) version 4.0.7 code (Reference 2) for the performance of analytical studies, supplemented by special purpose codes, as needed. Both the design and supporting analytical tools are products of an extensive experimental database developed for severe accident phenomena in general, and the U.S. EPR in particular, as described in Reference 1.

19.2.2 Severe Accident Prevention

The U.S. EPR includes design features aimed at preventing the onset of a severe accident, including the severe accident precursors identified in SECY-90-016 (Reference 3) and SECY-93-087 (Reference 4): ATWS, mid-loop operation, station blackout (SBO) event, fire, and an intersystem loss of coolant accident (ISLOCA).

19.2.2.1 Anticipated Transient Without Scram

An anticipated transient without scram (ATWS) is a very low probability event in which an anticipated operational occurrence (AOO) occurs and is not followed by an automatic reactor trip (RT) that is necessary to terminate the transient and to shut down the plant. The combination of the protection system (PS) and the control rod drive system is designed and tested to demonstrate the reliability of automatic reactor shutdown when required.

If an automatic reactor shutdown fails to occur, the U.S. EPR has design features to prevent or mitigate the consequences of the ATWS event. These include:

- A diverse actuation scram system with an independent reactor shutdown signal.
- Automatic actuation of the emergency feedwater (EFW) system on conditions indicative of an ATWS.
- An extra borating system (EBS) independent of the PS that can be used to inject heavily borated water to safely shut down the reactor.

The U.S. EPR ATWS response and the role of these features are described in Section 15.8.

19.2.2.2 Mid-Loop Operations

Mid-loop operation occurs during a plant shutdown where the reactor coolant system (RCS) is partially drained to support maintenance activities. The concern with mid-loop operation is that any loss of the RCS level control can greatly increase the risk of losing residual heat removal capabilities and ultimately lead to uncovering the core and subsequent core damage.

The U.S. EPR includes:

- Provisions for availability of reliable systems for decay heat removal.
- Instrumentation to provide reliable measurements of liquid levels in the RCS.
- Operational and procedural measures to provide reasonable assurance that the RCS remains stable and controlled while in a reduced inventory condition. These measures include both preventing a loss of residual heat removal (RHR) and enhanced monitoring criteria for a timely response to a loss of RHR should such a loss occur.
- Visible and audible indication of abnormal conditions in temperature, level, and RHR system performance parameters.
- Provisions to prevent damage to the RHR pumps due to overheating, cavitation, or loss of adequate pump suction fluid.
- Provisions for maintaining containment closure or for rapid closure of containment openings.

The RHR mid-loop operation is further described in Section 5.4.7. Provisions to prevent boron dilution during mid-loop operation are described in Section 15.4.

19.2.2.3 Station Blackout

An SBO event is defined as a loss of all offsite alternating current (AC) power to both essential and non-essential electrical buses and unavailability of the redundant onsite emergency AC power system. The Station Blackout Rule (from 10CFR50.63) specifies the need for alternative AC power sources to ensure that an operating plant can safely shut down in the event of a complete loss of AC power (offsite or onsite).

Consistent with the guidance in RG 1.155, the U.S. EPR includes two separate and independent station blackout diesel generator units (SBODG) capable of powering at least one complete set of shutdown loads. The SBODG sets have the capacity and capability to bring the plant to, and maintain the plant in, a safe shutdown condition with no support systems powered from the preferred power supply or emergency power supply system. In particular, decay heat removal is assured by powering the emergency feedwater pumps and essential instrumentation and control (I&C) systems.

In case of a loss of all AC power including the SBODGs, critical plant features are fed from a 12-hour battery until an AC power source can be recovered. With the inclusion and associated performance of these U.S. EPR design features focused on SBO, the U.S. EPR complies with associated regulatory guidance.

A complete description of the SBO event for the U.S. EPR is given in Section 8.4.

19.2.2.4 Fire Protection

The U.S. EPR fire protection design basis is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection features of the U.S. EPR are capable of providing reasonable assurance that, in the event of a fire, the plant will not be subjected to an unrecoverable incident. Two separate safe shutdown systems provide ongoing fire protection capabilities to meet the following performance criteria in the event that one train has been become inoperable:

1. Reactivity Control – Reactivity control shall be capable of inserting negative reactivity to achieve and maintain sub-critical conditions. Negative reactivity insertion shall occur rapidly enough such that fuel design limits are not exceeded.

2. Inventory and Pressure Control – With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling the coolant level such that subcooling is maintained.

3. Decay-Heat Removal – Decay-heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel to maintain a safe and stable condition.

4. Vital Auxiliaries – Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that systems are capable of performing their required nuclear safety function.

5. Process Monitoring – Process monitoring shall be capable of providing the necessary indication to assure these criteria have been achieved and are being maintained.

Additionally, the fire protection system design basis ensures that radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable (ALARA) and shall not exceed applicable regulatory limits. The U.S. EPR fire protection system is described in Section 9.5.1.

19.2.2.5 Intersystem Loss of Coolant Accident

ISLOCAs are defined as a class of accidents that can result in the overpressurization and rupture of the systems that interface with the RCS. An ISLOCA occurs when high-pressure reactor coolant is introduced into a low-pressure system or line due to a valve failure or inadvertent valve actuation, resulting in a direct and potentially unisolable discharge from the RCS to the environment.

The U.S. EPR conforms to the regulatory guidance associated with ISLOCA as described in Reference 3 and Reference 4, along with their associated staff requirement memoranda (SRM). The following are the requirements for the U.S. EPR:

- Design the systems connected to the RCS to an ultimate rupture strength at least equal to full RCS pressure.
- Systems not designed for full RCS pressure should provide:
  - The capability for leak testing of the pressure isolation valves.
  - Valve position indication available in the MCR when isolation valve operators are de-energized.
  - High-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

Three systems that connect to the RCS have the potential for ISLOCA susceptibility. These systems include the EBS, the chemical and volume control system (CVCS), and residual heat removal system (RHRS).

The EBS piping connected to the RCS system is designed for 2540 psig, which is above the design pressure for the RCS. The system may be used to perform hydrostatic testing of the RCS. This can be conducted through a normally isolated line that is rated to a pressure of 3625 psig. The EBS has two valves on the piping line into the RCS that provide the normal isolation function. The combination of a higher design pressure of the EBS and the isolation valves makes the probability of an ISLOCA between the EBS and the RCS negligible. The EBS is described in more detail in Section 6.8.

The portions of the CVCS that could be exposed to RCS operating pressure are designed for a pressure of 3640 psia, which is greater than the RCS design pressure. The CVCS also has containment isolation valves (CIV). The CVCS is described in more detail in Section 9.3.4.

Similarly, the SIS/RHRS has been designed with an ultimate capacity pressure greater than the RCS design pressure. The RHRS is described in more detail in Section 5.4.7.

19.2.2.6 Other Severe Accident Preventative Features

Severe accidents are low probability events characterized by multiple failures and coincident occurrences up to the total loss of safety-grade systems. As such, plant safety systems engineered for design basis events, whether specifically designed for event prevention or mitigation, serve a preventive function for severe accidents. Of particular interest for severe accidents are those systems closely linked to the most likely severe accident initiating events. These systems and a cross-reference to their description, including relevant protection systems, are as follows:

- Reactor coolant pumps, Section 5.4.1 (in particular, the standstill shaft seal system).
- Residual heat removal system, Section 5.4.7.
- Engineered system features, Section 7.3.
- Offsite power, Section 8.2.
- Onsite power, Section 8.3.

System depressurization triggers the actuation of accumulators and the safety injection system, if available. As such, the primary depressurization system (PDS) serves a unique role in the prevention of severe accident. As discussed in Section 19.2.5.5, the PDS is an integral part of any severe accident management strategy. Opening of the PDS valves provides a time window in which the introduction of core cooling can continue by employing all means available to the operators. If core cooling can be recovered, a severe accident is averted. As such, the PDS is distinguished as the last preventive measure available to plant operators prior to the transition from emergency operator procedures (EOP) to severe accident management guidelines (SAMG).

19.2.3 Severe Accident Mitigation

19.2.3.1 Overview of Containment Design

The U.S. EPR Reactor Building is composed of a Reactor Containment Building (RCB) and a Reactor Shield Building (RSB) separated by an annular region. The RCB is a post-tensioned concrete pressure vessel located inside the reinforced concrete RSB. A leak-tight steel liner plate covers the entire inner surface of the RCB, including the basemat. Within containment are the RCS, the in-containment refueling water storage tank (IRWST), and parts of the main steam and feedwater lines. The containment includes a large free volume of approximately 2.8 x 10^6 ft^3 and has a design pressure of 62 psig.

The containment systems implemented for severe accident mitigation are the combustible gas control system (CGCS), core melt stabilization system (CMSS), and the severe accident heat removal system (SAHRS). Figure 19.2-1—Core Melt Stabilization System, and Figure 19.2-2—Severe Accident Heat Removal System, present an illustration of the CMSS and SAHRS designs and their relationship to the reactor pressure vessel.

The containment is able to withstand the maximum pressure and temperature resulting from the release of stored energy during a loss of coolant accident (LOCA), main steam line break (MSLB), or severe accident and maintains its role as a barrier to prevent the uncontrolled release of fission products to the environment. A description of the containment's functional design is given in Section 6.2.1. The physical description of the containment is given in Section 3.8. Section 3.8.1.4.11 specifies results on the containment ultimate capacity pressure.

19.2.3.2 Severe Accident Progression

The U.S. EPR employs an ex-vessel strategy for the mitigation of severe accidents. As such, both in-vessel and ex-vessel processes and phenomena contribute to the eventual end state. To introduce the principal severe accident phenomena for the more likely scenarios, a hypothetical phenomenologically-bounding severe accident is described in this section. The description of the in-vessel phase is summarized from Reference 1, Section 4.0. The description of the ex-vessel phase is paraphrased from the component descriptions given in Reference 1, Section 2.0. Reference 1, Section 4.0 also summarizes the phenomena for the complete severe accident progression and correlates them with the Reference 4 safety issues.

The principal consideration in identifying the hypothetical phenomenologically-bounding severe accident is the role of the PDS. The consequence of this highly reliable feature is the rapid depressurization of the RCS. Rapid depressurization removes a degree of uncertainty associated with postulated scenarios since many such events become very similar to a large-break LOCA (LBLOCA). Therefore, for the purpose of identifying important U.S. EPR severe accident phenomena, the hypothetical phenomenologically-bounding severe accident is an initiating large primary system pipe rupture coincident with a SBO. Subsequent accumulator injection is credited to initially flood the core, providing the hydrogen source (i.e., water) necessary to maximize hydrogen generation.

19.2.3.2.1 In-Vessel Melt Progression

Characteristic of an SBO, a complete failure of all active safety systems is assumed. Without safety injection the core begins to heat-up and progressively dry-out. Unmitigated fuel ballooning, rupture and melting follows. Exothermic chemical reactions, primarily between zirconium and residual water and steam, result in significant hydrogen generation. The hydrogen presents a combustion hazard, particularly in the containment where mixture with oxygen is expected. Eventually, a molten corium pool will form inside the core; the pool then expands towards the heavy reflector and the lower core support plate. As the event progresses, intact fuel elements surrounding the core are eventually destroyed.

Since the melt is primarily oxidic, its contact with the heavy reflector does not lead to instant failure but to a slow, crust-limited heat-up. Due to its large mass, and correspondingly high heat capacity, the heavy reflector and lower support plate act as a temporary internal crucible, retaining the core within its boundary. As a consequence, it is expected that this intermediate molten pool will already contain a large fraction of the core. Melt-through of the heavy reflector, driven by natural convection, is expected to occur in the upper region of the molten pool. During melt relocation into the lower plenum the continued heating within the core coupled with the out-flowing melt is expected to widen the initial hole, allowing more core melt to relocate.

As a result of the contact with the residual water in the lower head, the released melt may form a partially fragmented debris bed or encrusted molten pool, or both. After evaporation of the residual water, a secondary molten pool forms within the lower plenum. The lower support plate will then be heated from both sides, by convection from above and by thermal radiation from below.

The two pools will evolve independently. Within the upper pool, remaining fuel and solid debris will heat-up. Newly created melt will exit through the existing hole in the heavy reflector and become incorporated into the lower pool. During this process the average temperature of the lower head will steadily increase, which leads to its deformation by thermal expansion and creep. Downward expansion of the lower head is ultimately limited by the concrete support structures provided at the bottom of the reactor cavity, the concrete enclosure surrounding the reactor pressure vessel (RPV). These structures preserve sufficient space for the outflow of melt and the later formation of a molten pool in the reactor cavity.

At some point, the RPV lower head fails thermally. Without lower head penetrations, this can begin as a local failure at a location in the upper part of the melt-contacted region. In this configuration, only part of the contained melt is released with the first pour. After this first relocation, further outflow into the reactor cavity depends on the development of the melt configuration within the RPV. Under the expected dry conditions, the lower head is subject to radiant heating from the surface of the molten pool and the surrounding aerosol-rich gas. This heat flux accelerates the global failure of the lower head and lower internals of the RPV. The chronology of events of this hypothetical severe accident is summarized in Table 19.2-1—Chronology of a Bounding Severe Accident through RPV Failure.

19.2.3.2.2 Ex-Vessel Melt Progression

After release from the RPV, a period of melt retention in the reactor cavity occurs followed by the spreading, flooding, quenching, and long-term cooling of the melt. This temporary retention phase accommodates the uncertainty associated with different RPV failure modes and release rates. The release of corium from the RPV into the reactor cavity will likely not take place in a single release, but over a period of time. Without a retention phase, the release of corium over an undefined period of time could result in potentially unfavorable conditions for subsequent melt spreading.

The melt retention phase is characterized by molten core-concrete interaction (MCCI) which ablates a layer of sacrificial concrete before releasing the corium into a lateral compartment for spreading. The surface of the reactor cavity that comes into contact with the corium is lined with a uniform thickness of sacrificial concrete. The sacrificial concrete is backed by a refractory layer of sintered zirconia bricks except for a rectangular melt plug at the center of the reactor cavity floor. The melt plug is composed of sacrificial concrete, with a thickness equal to the rest of the reactor cavity sacrificial concrete, but is backed by an aluminum gate atop a steel framework. This is the only part of the reactor cavity sacrificial concrete that is not backed by zirconia bricks and is therefore the defined failure area that allows the corium to be released into the spreading compartment.

The length of the temporary retention phase is driven by the release rate of corium from the RPV. There is a defined amount of concrete that is ablated during MCCI. The energy required for MCCI comes from the decay heat in the melt. For fast releases of corium from the RPV, there is an abundance of energy and MCCI proceeds quickly.

Alternatively, for slow releases of corium from the RPV, the decreased amount of energy in the melt reaching the reactor cavity causes MCCI to proceed at a slower rate. This leads to a longer retention phase which allows more of the melt to accumulate in the reactor cavity. The energy balance gives the melt retention phase a self-adjusting characteristic that decouples the spreading process from the uncertainties of the in-vessel phase of the severe accident.

In addition to melt accumulation, the retention period also gives the corium beneficial properties that contribute to successful melt spreading. The admixture of a defined amount of concrete into the corium equalizes the spectrum of possible melt states prior to spreading and generates more predictable melt properties. In addition, the temporary retention of the melt reduces the final temperature of the melt prior to spreading and the admixture of the concrete maintains the viscosity of the melt in a favorably low range.

After penetrating the sacrificial concrete, the melt heats and quickly fails the metallic gate and support structure. Upon failure, the gate opens a release path for the melt into the melt discharge channel which couples the reactor cavity to the spreading compartment. The flow of the hot melt through the gate erodes the surrounding material and expands the opening.

Following the failure of the melt plug, the melt flows through the discharge channel in a single pour. After passing the outlet of the melt discharge channel, the melt pours onto the floor of the spreading compartment where it is distributed over a large surface area (see Figure 19.2-3—Molten Debris Spreading Area). The floor and walls of the spreading compartment are composed of sacrificial concrete. Beneath the sacrificial concrete is a cast iron cooling structure which is flat on the side facing the spreading compartment and finned on the opposite side to enhance heat transfer. The layer of sacrificial concrete in the spreading compartment protects the cooling structure from the thermal loads of melt spreading and provides a time delay to allow the cooling structure to fill with water.

The spreading of the melt passively actuates spring-loaded valves that initiate a gravity-driven flow of cooling water from the IRWST. The incoming cooling water is distributed by means of a central supply duct underneath the spreading compartment which overflows and fills the cooling structure floor and walls. The water continues to rise up the walls of the cooling structure and pour onto the surface of the melt from the circumference. The rate of water ingress is limited to a maximum flow rate to avoid any energetic fuel-coolant interactions (FCI). Water overflow continues until the spreading compartment and IRWST water levels equalize, resulting in the submersion of the spreading compartment, transfer channel and a portion of the reactor cavity.

19.2.3.3 Severe Accident Mitigation Features

The U.S. EPR has design features to address a variety of severe accident challenges, including hydrogen generation and control, core debris coolability, high-pressure melt ejection (HPME), FCI, containment bypass, and equipment survivability. These features are described in detailed in Reference 1, Section 2.0. The following sections summarize that presentation. Performance analysis of these features is provided in Section 19.2.4.

19.2.3.3.1 External Reactor Vessel Cooling

The U.S. EPR severe accident features are focused on maintaining containment integrity through ex-vessel melt retention. Consequently, the U.S. EPR does not require external reactor vessel cooling to mitigate severe accidents.

19.2.3.3.2 Hydrogen Generation and Control

The generation of hydrogen can occur in the U.S. EPR during a severe accident due to oxidation on fuel rod surfaces, MCCI, and oxidation of the core support material. The largest contributor to hydrogen generation is the oxidation of the fuel rod cladding, which can vary depending on the timing of the melt progression. The CGCS and hydrogen monitoring system (HMS) are design features incorporated in the U.S. EPR to comply with the hydrogen generation and control requirements as follows:

- The CGCS provides for a mixed atmosphere within the containment (10 CFR 50.44(c)(1)).
- The CGCS limits the overall hydrogen concentration in containment to 10 percent by volume during and following an accident that results in a fuel cladding-coolant reaction involving 100 percent of the cladding surrounding the active fuel region. (10 CFR 50.44(c)(2)).
- The CGCS remains functional during and after exposure to the accident environmental conditions (10 CFR 50.44(c)(3)).
- The HMS continuously measures the hydrogen concentration in containment during and after the accident, and remains functional during and after exposure to the accident environmental conditions (10 CFR 50.44(c)(4)(ii)).

The CGCS is divided into two subsystems corresponding to their operational functions:

- Hydrogen reduction system.
- Hydrogen mixing and distribution system.

The hydrogen reduction system (HRS) consists of 41 large and six small passive autocatalytic recombiners (PAR) installed in various parts of the containment. The PARs are arranged inside the equipment rooms to support global convection within the containment, and thereby homogenize the atmosphere and reduce local peak hydrogen concentrations. Recombiners are also included in the containment dome to cope with stratification and to improve depletion after atmospheric homogenization.

The hydrogen mixing and distribution system is designed so that adequate communication exists throughout the containment to facilitate atmospheric mixing. Several of the equipment rooms surrounding the RCS are isolated from the rest of the containment during normal operation. In the event of an accident, communication is established between these normally segregated compartments, thereby eliminating potential dead-end compartments where non-condensable gases can accumulate. This ability to transform the containment into a single convective volume is supported by a series of mixing dampers and blowout panels.

The HMS monitors the hydrogen and steam concentration. The HMS provides information to the main control room on the hydrogen concentration, and its distribution within containment. The measuring points are arranged in different compartments of the containment to monitor the time dependence of the hydrogen distribution during a severe accident. The hydrogen concentration is monitored in the upper dome, steam generator (SG) compartments, pressurizer (PZR) compartment, and annular rooms.

The CGCS and HMS are described in detail in Section 6.2.5.

19.2.3.3.3 Core Debris Coolability

In the unlikely event of a severe accident in which the core melts through the reactor vessel, it is possible that the containment could be breached if the molten core is not sufficiently cooled. The CMSS and the SAHRS are U.S. EPR design features that address the issue of core debris coolability.

19.2.3.3.3.1 Core Melt Stabilization System

Melt retention within the RPV is not a design goal for the U.S. EPR. Rather, the U.S. EPR is equipped with an ex-vessel system to accommodate molten debris, including the entire core inventory and reactor internals. The goal of this system is to eliminate the potential for containment failure by any means derived from the core melt, including the interaction between core melt and the containment structure and the effects of melt cooling (i.e., over-pressurization of containment). When the molten debris reaches its final destination in the spreading room, is being cooled by water from the IRWST, and is no longer a threat to containment integrity, the core melt is considered “stabilized.” This condition is attained through the combined effects of the following portions of the CMSS:

- Reactor cavity.
- Melt plug.
- Melt discharge channel.
- Spreading area and cooling structure.

The reactor cavity utilizes a combination of sacrificial concrete and a protective layer of refractory material to provide a stage of temporary melt retention. The melt plug and gate are located in the reactor cavity and support the melt retention design by providing a defined failure location. The melt discharge channel utilizes a steel duct lined with refractory material to direct the conditioned melt from the reactor cavity to the lateral spreading compartment. The spreading area consists of a dedicated cooling structure lined with sacrificial concrete to promote stabilization of molten debris. The general configuration of the CMSS is shown in Figure 19.2-1.

Reactor Cavity

The reactor cavity refers to the region between the RPV and the surrounding structural concrete closest to the lower head. Following RPV failure, the reactor cavity receives core melt. The initial conditions for core melt in the reactor cavity are determined by the course of in-vessel core degradation, relocation and quenching, and finally by the sequence of melt release after failure of the lower head. All of these processes involve a degree of uncertainty. To make the U.S. EPR melt stabilization concept tolerant of such uncertainties, the reactor cavity is used to provide a period of temporary melt retention. This period of temporary retention addresses the fact that the release of molten material from the vessel will, most likely, not occur in one pour, but over a period of time.

Temporary retention is provided by a layer of sacrificial material that is penetrated by the melt before it can escape from the cavity. The corresponding delay, which is determined by the time needed to penetrate the sacrificial layer and to destroy the metallic gate, allows, for the more likely scenarios, for practically the entire core inventory to be collected in the cavity prior to spreading and stabilization – even in case of an incomplete first release of melt from the RPV.

The sacrificial layer consists of a 19.7 in layer of siliceous concrete with high iron-oxide content. The sacrificial concrete within the reactor cavity serves to equalize the spectrum of potential melt states by homogenizing the thermo-chemical conditions of the melt release from the vessel. Therefore, the retention phase serves to condition the melt so that the spreading process and subsequent measures are independent of the uncertainties associated with in-vessel melt progression and RPV failure mode. The advantages of the high iron-oxide content of the reactor cavity concrete are that it oxidizes remaining zirconium and uranium within the melt that can attack the zirconia bricks, thus protecting the structural concrete of the cavity. High iron-oxide concrete also leads to a low melt temperature and viscosity for spreading. A high SiO_2 composition also benefits the process through the formation of silicates that lower the radionuclide release from the corium pool.

The sacrificial concrete layer is backed with a refractory material that confines the melt and insulates the RPV support structure in case of a local penetration of the sacrificial concrete. The refractory material consists of zirconia bricks, which have a low thermal conductivity and a mechanical strength greater than concrete. This protective layer “guides” the melt towards the metallic gate of the melt plug.

Melt Plug and Gate

The upper part of the melt plug is a layer of sacrificial concrete with the same composition as the sacrificial layer within the cavity. However, this layer of concrete is not backed by refractory blocks but by an aluminum plate (referred to as the gate) atop a steel framework. At the end of the retention phase, the melt plug and gate are designed to fail open with sufficient cross-section to achieve a complete and rapid relocation of the accumulated melt into the lateral discharge channel leading to the spreading compartment.

The concrete cover of the plug is an integral part of the sacrificial layer in the cavity and has the same thickness of 19.7 in. Due to the large diameter of the cavity, the ablation front is expected to be relatively even and the entire surface of the gate is expected to be fully uncovered within a short time.

Once the molten debris comes into contact with the gate, the intensity of the convection within the molten pool is expected to quickly destroy the gate. The outflow of melt is limited by the residual concrete layer. The resulting rate of melt discharge after opening the full cross-section of the residual melt plug is substantially greater than that necessary to provide adequate spreading in the spreading compartment. If the gate initially failed over less than its full cross-section, the diameter of the opening would steadily increase due to the heat transfer from the flowing melt. Hole-widening effects make the discharge process self-adjusting; for a small initial opening, the duration of the discharge and the time of interaction will be correspondingly longer.

Melt Discharge Channel

Following the failure of the cavity retention gate, the melt flows through the transfer channel in a single pour. After passing the outlet of the melt discharge channel, the melt flows over the surface of the spreading compartment.

The melt discharge channel consists of a steel structure that is embedded in the structural concrete of the containment. The bottom, side walls and top of this structure are layered with refractory material. This protective layer of zirconia bricks has a low thermal conductivity and eliminates the possibility of blockages forming as a consequence of melt freezing.

Spreading Area and Cooling Structure

As previously discussed, the CMSS is designed for passive transport of molten debris through the discharge channel and into the spreading compartment. The spreading area is an approximately 1872 ft^2 horizontal concrete surface over which the molten debris disperses. Spreading increases the surface-to-volume ratio of the molten debris to allow effective stabilization via subsequent cooling. The spreading area is located in the lower portion of the containment and is surrounded by the IRWST. The configuration of the spreading area surface is shown in Figure 19.2-3.

The spreading compartment design prevents accumulation of a large amount of water so that molten debris spreads under dry conditions. The spreading compartment is a dead-end room and is isolated from the rest of containment by flood and splash walls. These features prevent the direct inflow of water from sprays, leaks, or pipe breaks. Only a limited amount of condensate may form inside the room. Though dry conditions are not required for successful spreading, they make the distribution more predictable and reduce the potential for fuel coolant interactions.

The spreading area is essentially a shallow crucible within which molten debris can be stabilized. A layer of sacrificial concrete within the spreading compartment covers a dedicated cooling structure used to cool the molten debris on all sides with water from the IRWST. This dedicated cooling structure consists of a number of cast iron cooling elements that line the floor and side walls of the spreading compartment. To enhance heat transfer, the horizontal and vertical plates have fins that form rectangular cooling channels. The sacrificial concrete layer protects the cooling structure against thermal loads resulting from melt spreading. It also delays melt contact with the metallic cooling structure so that the cooling elements will be flooded with water from the IRWST prior to the initial contact between them and the molten debris. The structural elements are joined using flexible connections so that the cooling structure withstands expansion and deformation.

The arrival of the melt into the spreading compartment triggers the opening of spring-loaded valves that initiate the gravity-driven flow of water from the IRWST into the spreading compartment. Initially, a cable holds each spring-loaded valve closed. Within the spreading compartment the cable is attached to a thermally sensitive initiator, consisting of a material of low melting point. When the initiator is destroyed during contact with molten debris, the cable will allow the spring-loaded actuator to open the flooding valve and allow water to flow from the IRWST.

The water first fills the central supply duct underneath the spreading area. From there, it enters the horizontal cooling channels and then fills the space behind the sidewall cooling structure. Finally, the water pours onto the surface of the melt and overflow will continue until the hydrostatic pressure in the IRWST and the spreading room is equal. Both the spreading room and the IRWST are open to the containment atmosphere with sufficient area of communication so there is no buildup of pressure as steam is generated in the spreading room. In parallel with the inflow of water, the melt interacts with the sacrificial concrete covering the horizontal and vertical cooling plates. The resulting delay allows the walls of the cooling structure to be cooled on the outside prior to the first contact with the molten corium.

19.2.3.3.3.2 Severe Accident Heat Removal System

The SAHRS works along with the CMSS to cool the molten debris. The SAHRS is a dedicated thermal-fluid system used to remove the heat generated in the containment during a severe accident. The SAHRS has four modes of operation, each playing a role in containment heat removal and controlling the environmental conditions within the containment so that its fission product retention function is maintained. These modes of SAHRS operation include:

- Passive cooling of molten debris.
- Active spray for environmental control of the containment atmosphere.
- Active recirculation cooling of the molten debris and containment atmosphere.
- Active backflush of the SAHRS pump suction strainer.

The SAHRS equipment is located in Safeguard Building 4, and includes:

- A suction line from the IRWST.
- Containment isolation valves.

- A recirculation pump.
- A heat exchanger for containment heat rejection.
- Discharge line to a containment spray header, the spreading room, and sump screen.
- Support from a dedicated cooling chain via plant auxiliary systems.

The SAHRS heat exchangers transfer the residual heat from the containment to the ultimate heat sink via dedicated portions of component cooling water system (CCWS) and essential service water system (ESWS) trains. During operation, the three possible flow paths downstream of the pump and the heat exchanger are:

- To a containment spray system with a ring header and spray nozzles.
- To the spreading area of the CMSS.
- To a sump screen flushing device which is used to remove accumulated debris.

The general configuration of the SAHRS is shown in Figure 19.2-2, with key design parameters provided in Table 19.2-2—SAHRS Design and Operating Parameters.

Passive Cooling of Molten Debris

In this mode the SAHRS provides water to the cooling structure surrounding the spreading compartment. Once molten debris is within the spreading compartment, water from the IRWST passively starts to fill the cooling structure. This dedicated flooding line is equipped with a flow limiter downstream of the IRWST outlet, which limits the flow such that its subsequent complete vaporization does not present a containment overpressurization challenge. This passive flow of water fills the cooling structure within five minutes. Water then overflows into the spreading compartment until it is hydrostatically balanced with water from the IRWST. This flooding submerges the spreading area and transfer channel, as well as a portion of the reactor cavity, thereby cooling any residual debris in those areas.

Operating in this passive mode, IRWST water supplied by the SAHRS boils off and is released into the free volume of the containment through the steam chimney directly above the spreading compartment (see Figure 19.2-1). As this process continues, the temperature and pressure within the containment steadily increase; however, the U.S. EPR containment is designed with sufficient free volume and structural heat sinks that atmospheric conditions of the containment do not approach design limits for several hours following the onset of core damage.

Active Containment Spray

The U.S. EPR containment has sufficient capacity to allow a grace period of several hours before operator action is needed to prevent the pressure and temperature within the containment from exceeding design limits. When operating in the containment spray mode, the SAHRS takes suction from the IRWST; coolant then flows through a heat exchanger outside containment prior to being routed back to the spray headers located in the upper volume of the containment. The spray water condenses atmospheric steam, thereby reducing containment pressure and temperature. The resulting condensate flows back into the IRWST for continued recirculation.

To initiate the SAHRS containment spray mode, the operator will have to perform the following steps:

1. Start the dedicated cooling train (i.e. start the essential service water and component cooling water system dedicated to the SAHRS).

2. Activate the motor-operated valves that allow the IRWST water to flow to the SAHRS pump.

3. Open the valve allowing water coming from the IRWST to flow directly to the spray header nozzles located in the reactor building dome.

4. Start the SAHRS pump.

Given the deliberate steps required to actuate the SAHRS, inadvertent actuation of the SAHRS is not a credible event.

Active Recirculation Cooling

As a core melt accident progresses, it can become necessary to use the recirculation function of the SAHRS to further control the environmental conditions within the containment. As previously discussed, the containment spray can be used to condense atmospheric steam with the condensate returning to the IRWST where it can be used as additional inventory for continued passive cooling of the molten debris. Once the containment spray has sufficiently reduced containment pressure, the SAHRS can be switched to a long-term recirculation mode where the SAHRS feeds water directly into the spreading area. As a result, the water pool in the cooling channels and on top of the melt becomes subcooled. Decay heat is now removed from the melt by single-phase flow, instead of by evaporation; and containment pressure is reduced.

In this mode of operation, the water level in the spreading compartment will rise to the top of the steam outlet chimney, overflow onto the containment floor and drain back into the IRWST where it can be recirculated back into the spreading area cooling system. Because the spreading compartment and the reactor cavity are connected through the gate and transfer channel, water will also enter the reactor cavity and submerge the vessel up to the level of the RCS piping. This establishes long-term cooling of any debris that has remained within the transfer channel, the reactor cavity, or the vessel itself.

Sump Strainer Backflush

The final mode of operation of the SAHRS is to provide a backflushing function for sump strainer. Operation in this mode serves to dislodge any debris from the sump strainers that might compromise the ability of the SAHRS to draw water from the IRWST. Only a fraction of the nominal SAHRS flow is used for backflushing;

therefore, the system can operate in this mode while continuing operation in another containment cooling mode.

SAHRS Dedicated Cooling Chain

To support the active heat removal modes of the SAHRS, portions of the CCWS and ESWS are used to form a dedicated cooling chain to transfer heat to the ultimate heat sink. This cooling chain is dedicated to severe accident operation and is not used to support normal plant operations or mitigate the effects of a design basis event. The SAHRS, the CCWS and the ESWS are designed to receive power from either the normal offsite grid, the emergency diesel generators (EDG) or the SBODGs.

The CCWS train consists of a pump located upstream of a dedicated heat exchanger, a surge tank connected to the pump suction line and a demineralized water supply line with a pressurizing pump. This portion of the cooling chain feeds water to the shell side of the SAHRS heat exchanger where containment heat is removed and discharged through the tube side of the CCWS heat exchanger interfacing with ESWS. The CCWS train is pressurized above the interfacing SAHRS to prevent contamination of the cooling chain by leakage of radioactive water through the SAHRS heat exchanger.

19.2.3.3.3.3 Summary

The U.S. EPR addresses the regulatory expectations of Reference 4 for core debris coolability as follows:

- The CMSS provides a large 1872 ft^2 area spreading surface to enhance debris spreading.
- The CMSS and SAHRS provide a means to cool the molten debris through both an early low-flow passive flooding phase and a long-term active flooding phase.
- The CMSS provides protection for the containment liner and other structural members by employing both an active basemat cooling system in the spreading compartment and layers of sacrificial and protective concrete in both the reactor cavity and spreading compartments.

Performance analysis is presented in Section 19.2.4 that demonstrate how the CMSS and SAHRS maintain the environmental conditions (pressure and temperature with uncertainties) resulting from severe accidents.

19.2.3.3.4 High-Pressure Melt Ejection

HPME is a postulated mechanism for the release of finely dispersed core debris into the containment atmosphere, corresponding to a rapid blowdown of the RCS. HPME results in rapid heat transfer between core debris and the containment atmosphere, potential hydrogen combustion, oxidation of metallic aerosols, and overpressurization of the containment. The resulting direct containment heating (DCH) has been assessed as a means of early containment failure because the stored energy of the debris is enough to cause containment overpressurization if a large quantity of the core inventory participates.

HPME and the associated DCH are not considered relevant severe accident phenomena for the U.S. EPR. The U.S. EPR design includes features that make the risk from HPME negligible for the more likely severe accident scenarios. The key feature is the PDS; however, low core power density, and a tortuous pathway from the reactor cavity to the upper containment, contribute to preventing or mitigating the potential consequences of high pressure melt ejection.

19.2.3.3.4.1 RCS Depressurization for Severe Accidents

RPV failure under high internal pressure is of importance to severe accident risk from HPME resulting in DCH. Even though such a failure is physically unlikely, an objective of the U.S. EPR severe accident response strategy is to convert high pressure core melt sequences into low pressure sequences with high reliability so that a high pressure vessel breach can be practically excluded. For the U.S. EPR, this is achieved through two dedicated severe accident depressurization valve trains, part of the PDS. Each of these PDS valve trains consists of a direct current (DC) powered depressurization valve in series with an isolation valve connected to the pressurizer, as shown in Figure 5.1-4.

The PDS valves are independent of the pressurizer safety relief valves (PSRV), safety-related components that provide RCS relief for overpressurization events. Both the PDS valves and the PSRVs discharge to the pressurizer relief tank (PRT). Each depressurization train has a discharge capacity of approximately 550 lb/s of saturated steam. Even though these valve trains are used exclusively for HPME prevention, in particular, and severe accident mitigation, in general, a 2 x 100 percent design philosophy is followed to provide a performance margin.

With the declaration of a severe accident (i.e., core exit temperature greater than 1200°F), the operator will actuate the PDS. As a consequence of RCS depressurization, loads anticipated within the reactor cavity (i.e., corresponding to the pre-rupture RCS pressure) will be below 275 psig. To address the possibility of loads approaching this value, the reactor cavity includes a set of walls aligned radially from the melt plug. These walls are designed to limit the downward expansion of the lower head resulting from contact with a molten pool and to provide protection for the reactor cavity integrity in the event of an abrupt vessel failure that results in a large section of the lower head falling into the reactor cavity.

19.2.3.3.4.2 Resistance to Core Melt Dispersal

The design of the reactor cavity significantly reduces the potential risk of HPME. The U.S. EPR reactor cavity is configured with several flow resistive obstacles to create a tortuous pathway from the reactor cavity to the upper containment. With each turn along this pathway, a significant amount of debris is expected to de-entrain onto compartment walls. The tortuous pathway to the containment atmosphere prevents the in-containment aerosol dispersal and long residence time required for HPME to occur, which eliminates the potential for early containment failure due to DCH by ejected core debris. Nonetheless, located in the cavity above the level of the protective layer are a number of ventilation outlet nozzles connected by a cylindrical ventilation channel. The combined cross-section of these openings is approximately 10.7 ft^2. Under the conditions resulting from a failure of the RPV under elevated pressure, a fraction of the dispersed melt could theoretically enter the ventilation duct; however, this fraction of melt should be negligible and the ventilation duct represents a tortuous path through which molten core debris must travel in order to reach the upper containment.

19.2.3.3.4.3 Summary

The U.S. EPR addresses the regulatory expectations of Reference 4 for HPME as follows:

- The PDS provide the capability to reliably depressurize the RCS after loss of decay heat removal.
- The reactor cavity and the U.S. EPR containment design, in general, provide a tortuous pathway to contain ejected core debris and prevent DCH.

19.2.3.3.5 Fuel-Coolant Interaction

Fuel-coolant interaction (FCI) is a process by which molten fuel transfers its thermal energy to the surrounding coolant, leading to break-up of corium with possible formation of a coolable debris bed or potential evolution to an energetic steam explosion. Two modes of contact between the molten corium and coolant are considered:

- A pouring contact mode, where corium is poured into a pool of water. This mode could conceivably occur within the RPV when corium relocates into the water-filled lower head of the vessel.
- An injection or stratified contact mode, where a pool of corium is flooded by water. This mode can occur within the RPV as a consequence of re-flood of the RPV, or later, during either molten pool formation inside the lower head or the designed flooding of the melt in the spreading area.

Based on the extensive research into FCI phenomena for PWR designs (summarized in Reference 1, Section 5.3.2), the likelihood of an FCI-induced steam explosion has been evaluated as negligible for both in-vessel and ex-vessel situations.

19.2.3.3.5.1 In-Vessel Fuel-Coolant Interaction

It has been hypothesized that a large in-vessel steam explosion could be sufficiently energetic to cause a breach of the reactor vessel, including a breach resulting in containment-failing missiles (e.g., the alpha mode of containment failure). This was initially identified in the Reactor Safety Study known as WASH-1400 (Reference 5). A reactor vessel breach could completely alter the course of the accident by causing the immediate ejection of fuel and fission products from the reactor vessel. A containment failing missile would essentially lead to simultaneous uncontrolled venting of the containment to the environment.

There have been several efforts utilizing expert elicitation to quantify the likelihood of an energetic steam explosion that fails the vessel and leads to an alpha mode containment failure – in particular, the second NRC-sponsored Steam Explosion Review Group in 1995 (Reference 6) and the Westinghouse AP600 application of the Risk Oriented Accident Analysis Methodology (Reference 7). The conclusion drawn from both studies was that in-vessel steam-explosion-induced lower head failure is physically unreasonable, effectively resolving the alpha-mode containment failure issue for pressurized water reactors.

19.2.3.3.5.2 Ex-Vessel Fuel-Coolant Interaction

The potential for ex-vessel FCI is minimized by designing the reactor cavity to avoid water accumulation during a LOCA. The only credible means of accumulating water in the reactor cavity is via a rupture of the large bore piping around its connection to the nozzle region of the RPV. The RPV nozzles are integral, forged pieces of the RPV and the large bore piping consists of integral, forged pieces. The only weld that exists is the connection of the large bore piping to the RPV nozzles. The probability is judged to be negligible of a LOCA initiator occurring within the weld that would result in an appreciable accumulation of water in the reactor cavity that would in turn progress to a severe accident. Therefore, at the time of RPV failure, corium which is discharged into the reactor cavity does not come into contact with a large amount of water.

The U.S. EPR design provides for a dry spreading area; only thin water films may develop because of steam condensation. Subsequent flooding of the corium for cooling and stabilization is performed only at a low flow. Therefore, ex-vessel melt-water interaction is considered in the case of melt quenching. During the initial quench of the melt, there is a pressure transient within the containment. Due to the large volume and heat capacity of the U.S. EPR containment, this pressure transient remains below the containment design pressure and considerably below its ultimate strength. Once the melt is initially quenched, a crust or viscous layer forms at the surface of the melt thereby reducing the intensity of energetic phenomena.

19.2.3.3.6 Containment Bypass

A containment bypass accident is one in which the fission products are released directly to the environment from the reactor coolant system. Such events are typically the leading contributor to risk in a nuclear power plant. The containment bypass accident class consists of two types of accident sequences: steam generator tube ruptures and interfacing systems loss-of-coolant accidents.

19.2.3.3.6.1 Steam Generator Tube Rupture

A steam generator tube rupture (SGTR) is a failure of one or more steam generator tubes resulting in the release of coolant from the reactor coolant system to the secondary system. The U.S. EPR employs a highly reliable strategy to reduce the reactor coolant system pressure and mitigate SGTRs. The SGTR mitigation concept is based on having the medium head safety injection (MHSI) pump delivery shutoff head at a value less than the setpoints for the steam generator safety valves in order to minimize potential radioactive releases. Partial secondary side cooldown is started automatically on low-low pressurizer level (MHSI actuation signal). This cooldown is needed to bring the RCS pressure below the main steam safety valve response threshold and enable injection from the MHSI system.

Prevention against overfilling of the affected SG and consequential prevention of liquid release to the environment is a design requirement for the safety systems and the steam generator, including situations with MHSI actuation.

Isolation of the affected SG, that is, isolating all feedwater supply (including emergency feedwater) and closing the main steam isolation valve (MSIV) and the main steam relief valve (MSRV), occurs automatically on a steam generator high level signal coincident with the end of partial cooldown. The subsequent plant cooldown to residual heat removal system operation is accomplished using the remaining intact loops.

No operator actions are required to mitigate the accident; and the secondary system remains sealed against releases to the environment after the relief valve or its block valve is closed. To create a containment bypass release pathway from a steam generator tube rupture, the accident scenario must include multiple system failures such that the steam generator tube rupture is not mitigated, and the secondary system pressure increases enough to open a safety valve. The safety valve must fail to reseat, and thereby provide a containment bypass pathway for the loss of coolant and for the possible release of fission products to the environment. As a consequence, the likelihood of a SGTR progressing to containment bypass has been significantly reduced in the U.S. EPR thus, the SGTR is not considered among the most likely of initiating events leading to a severe accident.

19.2.3.3.6.2 Intersystem Loss of Coolant Accident

Given the importance of maintaining the reactor coolant pressure boundary, isolation valves are designed and installed per safety-grade codes and standards, as described in Section 19.2.2.5, ISLOCA preventative measures. The U.S. EPR employs engineered ISLOCA mitigation through redundancy and separation. To be specific:

- Lines that originate in the reactor vessel or the containment are designed with a dual barrier protection that is generally obtained by redundant isolation valves.
- Lines that are considered non-essential in mitigating an accident isolate automatically in response to diverse isolation signals.
- Lines which may be useful in mitigating an accident have means to detect leakage or breaks and may be isolated should this occur.

19.2.3.3.7 Equipment Survivability

The structures, systems, and components (SSC) that are required for severe accident response are designed to withstand the severe accident environments they would experience in postulated accident scenarios, for the duration of the period in which they are needed, including the effects of pressure, temperature, and radiation. The U.S. EPR approach to equipment survivability considers the following:

- Identification of the SSC required for severe accident response from the more likely initiating events leading to core damage and reactor vessel failure.
- Functional performance criteria for each SSC, (e.g., mission times).
- Environmental conditions (e.g., pressure, temperature, radiation) which components necessary to manage and monitor the progress of a severe accident need to withstand.

19.2.3.3.7.1 Equipment and Instrumentation Necessary to Survive

Systems specifically designed for the environmental conditions anticipated during a severe accident within the RCS and the containment:

- Primary depressurization system (PDS) valves.
- Core melt stabilization system (CMSS).
- Combustible gas control system (CGCS).
- Severe accident heat removal system (SAHRS).

The PDS, CMSS, and CGCS components are located inside the containment and therefore are qualified for local ambient conditions, namely pressure, temperature, humidity and radiation. While the SAHRS is used to limit the pressure and temperature inside the containment, its main components, namely the heat exchanger and pump, are not located inside the containment. These components only need to be qualified for elevated temperature and radiation doses inside the compartments in the Safeguard Building where they are located. Containment isolation valves, containment penetrations, air locks, hatches and gaskets, are required to maintain their leak tightness during a severe accident. This equipment is qualified for elevated pressure and temperature. Table 19.2-3—Severe Accident Instrumentation and Equipment, summarizes all instrumentation and equipment necessary to monitor the severe accident progression and to allow for operator action.

19.2.3.3.7.2 Severe Accident Environmental Conditions

Environmental conditions for equipment survivability in a severe accident are quantified through the performance analysis described in Section 19.2.4.4.5. This analysis provides a realistic assessment of equipment stresses.

19.2.3.3.7.3 Basis for Acceptability

While severe accident equipment does not necessarily have to meet rigorous codes, standards, or procedures as typically specified for licensing design basis, the performance analysis given in Section 19.2.4.4.5 coupled, as necessary, with applicable equipment testing provides reasonable assurance that the equipment can perform its identified function during severe accident conditions. Of particular importance are those SSCs expected to directly inform the operator of critical measures requiring operator action and those SSCs expected to respond to operator action. Those SSCs are included in Table 19.2-3 associated with RCS depressurization, SAHRS operation, and annulus ventilation.

For much of the early phases of a U.S. EPR severe accident, event progression is passive. Specifically, this is from the onset of the severe accident (i.e., core outlet temperature exceeds 1200 F) until SAHRS actuation. Prior to this period, in-vessel conditions are accurately represented in the main control room. Most of these instruments and controls support design basis functions, and therefore are designed to meet the applicable code or standard defining equipment qualification. The measurement of the core outlet temperature is a highly reliable part of the operational incore instrumentation and includes 12 wide-range thermocouples evenly distributed across all four I&C divisions that take data from all four sectors of the core.

Following the actuation of RCS depressurization, the PAR performance is paramount. The AREVA PAR design has received extensive testing for a broad range of pressure, temperature, humidity, aerosol and radiation conditions.

The SAHRS system relies on conventional pump and spray technology with a long history of reliable performance. In addition, the SAHRS backflush capability assures its performance in the event of debris-blockage in the IRWST sump region.

19.2.3.3.8 Containment Venting

The U.S. EPR has not been designed with a dedicated severe accident containment vent system. Specific containment overpressure protection is provided through its large size and strength and through the availability of 47 PARs and the SAHRS for the removal of hydrogen and steam, respectively, the principal contributors to high containment pressure during a severe accident. The functions of these systems are described in Section 19.2.3.3.2.

19.2.4 Containment Performance Capability

19.2.4.1 Introduction

AREVA NP has developed a methodology (Reference 1) designed to confirm the adequacy of the U.S. EPR to address severe accident-related safety concerns. The principal issues relating to containment performance are hydrogen control, core debris coolability, high pressure melt ejection, fuel-coolant interactions and equipment survivability. This section describes the containment performance analysis for the U.S. EPR that meets the regulatory goals. Specifically addressed is the deterministic containment goal from Reference 4 which states:

"The containment should maintain its role as a reliable, leak-tight barrier (for example: by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products."

19.2.4.2 Analytical Methodology

The generic severe accident evaluation methodology involves:

- Identification of safety goals.
- Documentation of severe accident engineering activities addressing related issues.
- Derivation of a calculation matrix addressing severe accident uncertainties.
- Presentation of analysis results based on the derived calculation matrix.

The methodology for modeling severe accident phenomena is documented in AREVA NP's severe accident safety issue evaluation topical report (Reference 1) and involves a three-step approach:

1. Evaluation of those scenarios considered relevant or least improbable (Relevant Scenarios).
2. An uncertainty analysis exploring a range of values affecting the phenomena of interest (Uncertainty Analysis).
3. Additional parametric, sensitivity, or confirmatory analyses, as needed (Supplemental Analyses).

To simulate integral plant response, the U.S. EPR is modeled using the computer code MAAP4.0.7 (Reference 2). MAAP4 can simulate the response of light water reactor power plants during severe accidents sequences, including actions taken as part of the accident management. The code quantitatively predicts the evolution of a severe accident starting from full power conditions given a set of system faults and initiating events through key phenomena, such as core melt, reactor vessel failure and containment failure. Furthermore, models are included in the code to represent the actions that could stop the accident by in-vessel cooling, external cooling of the reactor pressure vessel or cooling the debris in the containment. MAAP4.0.7 contains new algorithms to model the severe accident phenomena and response functions unique to the U.S. EPR.

19.2.4.2.1 Regulatory Considerations

10 CFR 50.44 stipulates, among other things, that the amount of hydrogen to consider be equivalent to the hydrogen produced from the oxidation of 100 percent of the fuel cladding surrounding the active fuel with water. The ideal method to account for all of this hydrogen would be to have MAAP force all of the fuel cladding to interact with coolant to produce the required amount of hydrogen. Unfortunately, such an option is not available. Therefore an alternate method was employed to simulate this 100 percent clad-coolant hydrogen reaction. This alternate method supplements the clad-coolant reaction with the additional hydrogen that is produced from the MCCI. Physically, the hydrogen production due to MCCI occurs at temperatures above the auto-ignition temperature and therefore burns as it is generated. Ventilation duct outlets, located above the maximum possible melt level within the reactor cavity,

provide a supply of oxygen that mixes with high temperature hydrogen, resulting in a standing flame during MCCI. In MAAP4 calculations, however, the temperature of auto-ignition was raised to an unobtainable value. The effect is that the hydrogen produced through MCCI is added to the inventory produced from the clad-coolant reaction. The results of the uncertainty analysis show that the hydrogen added from the MCCI is more than enough to substitute for the portion of the fuel clad that does not react in-vessel.

19.2.4.2.2 Relevant Scenarios

The first step in developing the calculation matrix is the identification of the set of relevant scenarios. The relevant scenarios are identified by incorporating risk/consequence information from probabilistic risk assessment (PRA) to select those events that are more likely to lead to core damage and have the potential to challenge containment integrity.

The verification goal for these cases is to show that the severe accident measures of the U.S. EPR function as designed and the leak-tightness and operability of the containment system is maintained. Large uncertainties exist as to what initiating events and coincident occurrences lead to a severe accident. For this reason, relevant scenarios are defined as those having a Core Damage Frequency (CDF) greater than 1.0E-08/yr.

Relevant scenarios are derived using results from Level 1 PRA. This is done by identifying those initiating events whose CDF exceeds 1.0E-08/yr and identifying a corresponding Core Damage End State (CDES). CDES are used by PRA to link the Level 1 core damage event trees to the Level 2 containment event trees. This is done by bringing together core damage sequences with similar characteristics, and using those sequences as the initiating event for examining severe accident mitigation and containment failure probability.

The relevant scenarios evaluated from this process are:

- Loss of offsite power with Seal LOCA.
- Loss of offsite power with a low pressure end state.
- Loss of offsite power with a high pressure end state.
- Loss of balance of plant.
- Small LOCA.

19.2.4.2.3 Uncertainty Analysis

An objective of the uncertainty analysis is to analyze the range of conditions over which severe accidents are most likely to occur and to capture the full event progression for which the U.S. EPR severe accident response features were designed. These include core damage, reactor vessel failure, and melt relocation into the containment. The severe accident uncertainty analysis provides broad insight into the performance of the U.S. EPR severe accident response features.

Seventeen processes and phenomena associated with the more likely U.S. EPR severe accidents have been identified along with their individual uncertainty ranges. A total of 27 different MAAP4 parameters are associated with the 17 processes and phenomena (Reference 8). Each of these parameters has a range identified in which the parameter could possibly vary. It is this set of 27 parameters that comprises the severe accident uncertainty domain for the U.S. EPR. The transient type is included among the sampled uncertainty concentrations and is sampled according to the predicted frequency.

To evaluate this uncertainty domain, a non-parametric statistical approach has been adopted. The process involves "Monte Carlo"-like simulations using the MAAP4.0.7 computer code and the U.S. EPR plant model. For each execution of the MAAP4.0.7 code, each of the important phenomena and plant process parameters being treated statistically are randomly sampled based on a previously determined probability distribution. Each execution of MAAP4.07 can be viewed as the performance of an experiment with the experimental parameters being the important phenomena and plant process parameters. The result produced from each experiment can be any calculated measure such as hydrogen concentration, containment pressure, and fission product mass. This process can treat a large number of uncertainties simultaneously, far more than could be reasonably considered with response surface techniques. Unlike response surface methods, which often produce probability distributions for conditions not necessarily corresponding to the real case, this Monte Carlo method propagates input and model uncertainties at the point being analyzed.

Based on the results of a suite of 59 sample calculations, the uncertainty domain of any particular performance metric of interest is quantified. The selection of 59 samples is based on the work of Wilks (Reference 9). Following this non-parametric approach, when 59 observations are drawn from an arbitrary, random distribution of outcomes, it has been shown that the largest value is the limit such that with 95 percent confidence, at least 95 percent of all possible observations from that distribution will be less than the resulting largest value; that is, this result is the 95/95 tolerance limit. The 95/95 benchmark is assumed to be an adequate reflection of the total tolerance limit of any particular performance metric used to demonstrate the U.S. EPR severe accident response features.

19.2.4.2.4 Supplemental Analyses

Additional standalone MAAP4 analyses were performed to evaluate certain specific questions regarding the design of the U.S. EPR. These include the SAHRS performance analysis, which used a relevant scenario case with the highest containment pressure. This bounding case was permuted several times to demonstrate the acceptable performance of the SAHRS. Some analyses require analytical tools unavailable to MAAP 4.0.7 or require other methods. As needed, additional analyses are used outside of the uncertainty analysis to supplement the findings of the uncertainty analysis.

The method used to analyze core debris coolability required the use of features not available in MAAP4.0.7. As necessary, the additional codes MELTSPREAD-1 (Reference 10) and WALTER (Reference 1) were used to model certain aspects of the decomposition and subsequent solidification of the corium-concrete mixture. Inputs required for both of these codes were derived from the bounding values presented in the Uncertainty Analysis (i.e., MAAP4) results.

19.2.4.2.5 Combustible Gas Control

The analysis of the combustible gas control system is based upon the results of the uncertainty analysis. There are, however, some unique methods employed in examining the uncertainty analysis results as they pertain to the control of combustible gases.

Given the conditions necessary for hydrogen combustion, the analysis requires knowledge of the hydrogen and steam concentration in the containment as a function of time and necessitates a numerical analysis which models the containment environment. The approach is to model the containment with MAAP Version 4.0.7 to integrate the in-vessel and ex-vessel melt retention analysis.

The initial conditions for the accident release scenarios were obtained from in-vessel analyses. The initial conditions for a release scenario, such as the magnitude, location, and time of release, can have a significant impact on the accident progression. Thus, an array of initial conditions was analyzed. Upon the occurrence of a severe accident, hydrogen begins to form inside of the reactor vessel with a production spike occurring at the time of the relocation of the core. Since the accident is likely to be caused by a loss of coolant there will be opportunities for the hydrogen to exit the RCS into the containment prior to vessel breach.

The highest concentration of hydrogen in the containment is likely to occur at the time and place of release. The most unfavorable scenarios are: highest amount of hydrogen in the containment, highest concentration of hydrogen, and highest ratio of hydrogen to steam. Ignition sources and locations can play a significant role in the accident progression as well. From the simulated accident scenarios, the pressure and temperature response of the containment is tracked. The analysis assumes that combustion can occur at any given time, given the appropriate atmospheric conditions.

19.2.4.2.5.1 Deflagration

The pressure resulting from deflagration represents the primary challenge to containment integrity. The nature of the combustion front has a dynamic influence on combustion-induced pressure loads. For slow deflagration, however, this pressure is bounded by the adiabatic isochoric complete combustion (AICC) pressure. The AICC pressure is the maximum pressure that can result from a laminar combustion, and is what would result if the combustion process were to undergo complete combustion in a constant volume (isochoric) and there was no heat transfer to the outside volume (adiabatic). However, this pressure could hardly be reached in a realistic containment because:

- As flame velocity is low, heat can be transferred to the structures, to inert gases, to steam, and also to droplets (departure from adiabatic condition).

- If hydrogen concentration is below eight percent by volume, combustion is not complete. The complex structure of the large containment also leads to incomplete combustion (departure from the completeness condition).

19.2.4.2.5.2 Detonation and Flame Acceleration

The loads resulting from flame acceleration may not be bounded by AICC pressure (Reference 11). Evidence that may suggest otherwise is inconclusive. Therefore, evidence is presented to demonstrate that the occurrence of flame acceleration, and thereby deflagration-to-detonation transition (DDT), is highly unlikely.

Experimental results show that for flame acceleration, and consequently DDT, the most important parameter is the expansion ratio. This property will be related to the sigma (σ) criterion.

The σ criterion relates the expansion ratio σ (density of the gas before combustion divided by the density of the gas after non-isochoric combustion) to a limit value obtained through experimentation. Combustion risk is analyzed using the σ criterion, which states that there is no risk of flame acceleration as long as the expansion ratio σ remains below an experimentally based limit (σ^*). For regions with $\sigma > \sigma^*$, FA cannot be excluded by this procedure alone but does not necessarily occur. The geometrical conditions must also favor flame acceleration (obstacles, no venting volume). The σ criterion is a necessary condition for flame acceleration, but not necessarily sufficient by itself. Experimental results show that for lean hydrogen mixtures the critical sigma values (σ^*) depend upon the Zeldovich number. The sigma index is the sigma value normalized to the critical sigma value, defined as σ/σ^*. For scenarios where the sigma index is greater than 1.0, flame acceleration cannot be excluded.

19.2.4.3 Assumptions

MAAP version 4.0.7 was used to simulate U.S. EPR integral system performance during a severe accident. Analyses using MELTSPREAD and WALTER code were performed to supplement the integral system results.

19.2.4.4 Severe Accident Evaluations

This section describes the containment performance analyses for the U.S. EPR that meets the goals set forth by Reference 3 and Reference 4 with respect to the following principal issues:

- Combustible gas control.

- Core debris coolability.

- High pressure melt ejection (HPME).

- Fuel-coolant interactions (FCI).

- Equipment Survivability.

19.2.4.4.1 Combustible Gas Control

Figures cited in this section presenting "Tolerance Limit" plots illustrate the minimum, median and maximum result from all sample calculations at any given time.

During a design-basis LBLOCA or a severe accident, hydrogen can be produced within the reactor pressure vessel by zirconium fuel cladding reacting with water, by radiolysis of water, or by the corrosion of structural support steel within the vessel. For a severe accident in which the reactor pressure vessel has failed, hydrogen can also be produced as a product of several chemical reactions associated with MCCI. As a consequence, this combustible gas can be introduced into the containment. The CGCS is provided to limit the hydrogen concentration in the containment so that containment integrity is not endangered as a result of either overpressurization or combustion.

19.2.4.4.1.1 Design Evaluation

The design of the U.S. EPR CGCS considers the peak hydrogen concentrations expected from the more likely severe accident scenarios. In these analyses, 10CFR50.44 requires that 100 percent of the fuel cladding reacts with water. Although hydrogen production due to radiolysis and corrosion occurs, the cladding reaction with water and MCCI dominates the production of hydrogen. The function of the CGCS is to lower hydrogen concentration through the recombination with oxygen, thus, reducing the likelihood of combustion event, particularly those leading to deflagration or detonation.

This analysis demonstrates the following:

- Hydrogen is distributed in the containment and removed for load minimization.
- The global hydrogen concentration in the containment atmosphere does not exceed 10 percent by volume.
- The hydrogen concentration is reduced to levels below 4 percent by volume 12 hours after the onset of a severe accident.
- The containment can withstand (retain its integrity) a global deflagration, based on the amount of hydrogen and resulting pressure (AICC) at the time of ignition.
- There is no risk of flame acceleration or deflagration-to-detonation transition (DDT).

19.2.4.4.1.2 Hydrogen Production

During a severe accident in the U.S. EPR, hydrogen can appear in the containment from both in-vessel and ex-vessel sources. In-vessel hydrogen production is due to zirconium or steel oxidation. The zirconium oxidation is limited by the availability of steam and zirconium, and by the steam diffusion process into the clad surface. Steel oxidation in the primary circuit will impact the core barrel, internals, core support plate and upper core plate, and the metallic phases in the corium. Radiolysis is also possible; however, the hydrogen generation rates by this process are relatively small in comparison. Ex-vessel hydrogen production is the consequence of the oxidation of metals by steam or water that are released during the interaction. The metals Zr, Si, Cr, and Fe will oxidize. Zr, Cr, and Fe come from the molten core, molten parts of the RPV and the concrete rebar. Si (and some SiO) is produced by initial reduction of SiO_2 present in the concrete by Zr. The main effect of SiO_2 is to delay the hydrogen production.

Because of the large uncertainties associated with the availability of steam and water to interact with high temperature metals, 10 CFR 50.44 requires that analyses assume 100 percent oxidation of cladding surrounding active fuel. The containment performance analyses address this requirement by realistic simulation of hydrogen generation processes, penalizing assumptions influence these processes, and precluding auto-ignition in the reactor cavity.

Auto-ignition occurs when metal-water contact temperatures exceed 1430°F and a sufficient concentration of oxygen is present, conditions consistent with the reactor cavity following reactor pressure vessel failure. Realistic simulation of MCCI processes in the U.S. EPR show that about 1300 lbms of hydrogen is produced. As such, the minimum hydrogen mass in containment equating to 100 percent cladding oxidation is found by assuming 100 percent in-vessel hydrogen generation, i.e. 3300 lbms, minus 1300 lbms or about 2000 lbms. Hydrogen generation by this method is highly dependent on the uncertainties associated with in-vessel progression. Because core melt is sufficiently rich in metals capable of reacting during the ex-vessel phase, the 1300 lbms approximates the total hydrogen generation uncertainty. Figure 19.2-4—Tolerance Limit Plot of Hydrogen, shows the range of hydrogen production observed in the uncertainty analysis, verifying that the 2000 lbm threshold is exceeded by the minimum observed in the analyses. The median result is very near 3300 lbms of hydrogen mass, nearly exactly the 100 percent oxidation of fuel cladding surrounding the active fuel, and the maximum value is significantly above the requirement of 10 CFR 50.44.

19.2.4.4.1.3 Hydrogen Distribution

The issue of hydrogen distribution is the transport of hydrogen from production sources (i.e., the reactor core and MCCI) to locations in which concentrations can result in combustible configurations. As a very light element, hydrogen easily diffuses through heavier gaseous substances. In spaces without inherent convection currents, hydrogen may stratify, consolidating in high concentrations that pose a combustion risk. An inherent mitigating consideration is that steam, either from a large break or the pressurizer relief valves, reduces the combustion potential in two ways: by enhancing the homogenization of hydrogen and thus reducing the peak hydrogen concentrations, and by reducing the flammability through higher steam volume concentrations.

The release of hydrogen into the containment is predominant in the spreading room and chimney, the reactor pit, and the equipment rooms (pumps and steam generators). Excluding the previous compartments Figure 19.2-5—Hydrogen Concentrations through the U.S. EPR Containment, reveals that the hydrogen concentrations are close to each other and behave very similarly, as would be expected for a well-mixed containment atmosphere. Each trace appearing in Figure 19.2-5 represents a different compartment hydrogen concentration result. The observable differences correspond to the relationship of those compartments to the locations in which hydrogen originally appears. The small variation demonstrates the desired occurrence of global convection and resolves the concern of possible secluded recesses of high concentrations of trapped hydrogen.

19.2.4.4.1.4 Hydrogen Combustion

The combustion mechanism for hydrogen can be classified into two regimes, deflagration and detonation. A deflagration is a laminar combustion process where the flame speed, or the combustion front, is sub-sonic. These can be further divided into slow deflagration and fast deflagration. Slow deflagrations are typically classified with a flame speed below 330 ft/s. Fast deflagration is produced as a result of flame acceleration, which is also the driving mechanism for detonation. A detonation is a combustion process where the flame speed is sonic or supersonic.

Hydrogen combustion can have two damaging effects on the containment and equipment, those resulting from either pressure or temperature. The primary function of the CGCS is to minimize the threat of combustion by maintaining the global concentration of hydrogen below 10 percent by volume, as required by 10 CFR 50.44. This is accomplished through global convection and the distribution of the PARs (which itself aids in global convection). Figure 19.2-6—Tolerance Limit Plot of Hydrogen Concentration, shows that the global hydrogen concentration did not reach or exceed 10 percent by volume for any of the scenarios.

Containment structural integrity must be maintained per 10 CFR 50.44. Thus, the containment response was monitored to ensure that the pressure loads resulting from the accumulation and combustion of hydrogen did not exceed the containment ultimate capacity pressure limit. To provide reasonable assurance that structural integrity was not compromised, the containment was qualified with regard to two phenomena: (1) global hydrogen deflagration and (2) flame acceleration.

With regard to global deflagration, the AICC pressure was used as a bounding value for the pressure that would result should a single large deflagration occur. From Figure 19.2-7—Tolerance Limit Plot of Containment AICC Pressure, the global maximum AICC pressure is 105 psia, for all the uncertainty cases. This does not exceed the containment ultimate capacity pressure of approximately 119 psig (see Section 3.8.1.4.11).

The pressure loads from a detonation or flame acceleration are the results of dynamic pressure caused by the combustion front. These loads resulting from flame acceleration may not be bounded by AICC pressure. To eliminate the possibility of flame acceleration, the sigma index was calculated for every compartment and every case. The reactor cavity and the spreading room were excluded from this comparison because of the presence of the corium, which would be expected to auto-ignite the hydrogen in those compartments. In approximately 15 percent of the simulations, conservatism associated with hydrogen production in excess of the 10 CFR 50.44 requirement was brought into improved alignment with the regulation (although still conservative) to eliminate this penalty. Figure 19.2-8—Tolerance Limit Plot of Sigma Index for the Pump/SG Compartment, shows the sigma index for the pump/SG compartment (representative of the worst location) for all cases.

The results show that the sigma index does not exceed 1.0 for any scenario where the hydrogen produced is less than or equal to that resulting from 100 percent oxidation of the ziralloy cladding; therefore, the risk of flame acceleration or DDT in the U.S. EPR containment is negligible.

It should also be noted that for the most penalizing global deflagration and flame acceleration estimated from this analysis, an ignition source would need to be both timed at the worst possible moment and that the ignition source be located spatially in the right place, both highly unlikely events.

19.2.4.4.1.5 Hydrogen Recombination

PARs are simple devices, consisting of catalyst surfaces arranged in an open-ended enclosure. In the presence of hydrogen (with available oxygen), a catalytic reaction occurs spontaneously at the catalyst surfaces and the heat of reaction produces natural convection flow through the enclosure, exhausting the warm, humid hydrogen-depleted air and drawing fresh gas from below. PARs work both individually, as a remover of free hydrogen in the containment, and collectively, to drive atmospheric circulation in the containment, thus, encouraging the homogenization of hydrogen.

The primary concern with hydrogen concentrations is for early containment damage or failure, which can result in a large radioactive release. As such, mitigation of the long term accumulation of hydrogen is necessary. To remove any threat of combustion, hydrogen concentrations must be reduced to levels below four percent by volume (in dry atmosphere). Part of the design basis of the CGCS is to reduce the concentration of hydrogen to levels below four percent by volume 12 hours after the onset of a severe accident.

Figure 19.2-9—Hydrogen Concentration at Significant Events, shows the hydrogen concentration at key events during the accident sequence for the uncertainty cases. It shows that the hydrogen concentration at the time of SAHRS actuation (for the uncertainty scenarios SAHRS actuation was initiated 12 hours after the onset of a severe accident) is always less than four percent.

19.2.4.4.2 Core Debris Coolability

The U.S. EPR design includes provisions for the retention and long-term stabilization of the molten core inside the containment. The mitigation scheme presupposes a depressurization of the RCS prior to the formation of a molten pool within the lower plenum of the RPV. After RPV failure, the molten corium first accumulates in the reactor cavity and later relocates, in one event, into a lateral compartment. Spreading of the melt is followed by flooding, quenching and sustained cooling of the corium.

The assessment of core debris coolability begins with a characterization of the main processes involved in this sequence, namely:

- Temporary melt retention and conditioning in the reactor cavity.
- Gate failure and relocation of the accumulated melt into the spreading area.
- Melt spreading.
- Passive flooding, quenching and long-term heat removal of the spread melt.

Part of the ex-vessel severe accident mitigation strategy is that the consequences of MCCI contribute to the transformation of the melt into a stable configuration. In this two-stage stabilization process, retention and spreading, MCCI is not only unavoidable; but, it is actually incorporated into the U.S. EPR solution for severe accident mitigation. Thus, the phenomena associated with MCCI are treated separately in this section to support the degree of topical coverage appropriate for a thorough characterization of the U.S. EPR severe accident mitigation strategy.

19.2.4.4.2.1 Temporary Melt Retention in the Reactor Cavity

A phase of temporary melt retention in the reactor cavity makes the relocation of the melt into the spreading compartment independent of the release sequence from the RPV and of the corresponding state of the melt (characterized by temperature and composition). Temporary melt retention is provided by sacrificial and protective layers in the reactor cavity. The sacrificial layer is intended to delay melt progression in the vertical direction and contact with the melt gate until effectively the entire corium inventory has been released from the RPV. During this time, the protective layer will restrict the radial penetration into the load-bearing RPV support structure. This effectively decouples the subsequent stabilization measures from the inherent uncertainties associated with in-vessel melt pool formation and RPV failure.

For melt accumulation to be successful, the following targets must be sufficiently fulfilled before the melt pool comes into contact with the melt gate:

- The MCCI in the reactor cavity is still on-going at the time when the lower head plus lower support plate has failed and, thus, the residual in-vessel melt is released into the reactor cavity.
- All incorporated material has been diluted in the MCCI-pool and conditions that favor melt spreading are achieved.

In considering the former, a performance metric was defined as the difference between the time at which the lower core support plate fails and the time at which the gate fails. The uncertainty analysis results show that this value is always positive with a margin exceeding a few hours and, therefore, the lower core support plate always fails before the gate fails. This holds true for a broad range of melt states, reactor vessel failure modes, and melt release sequences.

The tolerance to such uncertainties is indicative of the U.S. EPR “self-adjusting” characteristic of MCCI in the reactor cavity. This is further illustrated by examining the trending between the time of RPV failure and the duration of the retention period. Figure 19.2-10—Time of RPV Rupture versus Duration of Retention Period, illustrates that as RPV failure occurs later in the transient, then the retention period generally becomes longer. Neglecting differences in corium mass emanating from the RPV failure, this trend results because the later the RPV failure occurs, the lower the residual decay heat power of the melt when it enters the reactor cavity. With the lower decay heat level, the retention phase becomes longer as reactor cavity ablation is slowed.

Given the inherent characteristics of the MCCI to adjust the ablation front progression to the amount and energy content of the melt, the performed analysis demonstrates that the core melt inventory will be collected in the reactor cavity sufficiently before the melt comes into contact with the melt gate independent of the underlying scenario and reactor pressure vessel release modes. An important consequence of the MCCI process is that the composition of the melt at the end of the retention period is found to be well conditioned for spreading. This is assessed from the oxide/metal ratio and the relatively low volumetric solid fraction and viscosity (inferred from the corium temperature) at the time of gate failure.

This unification of melt accumulation and conditioning is attributed to the geometrical constraint, established by the refractory layer. This spatially restricts the extent of melt front progression in the radial and axial directions and effectively predefines the amount of incorporated concrete at the end of the retention period. In this sense, temporary retention makes all subsequent measures independent of initiating scenarios as well as from the inherent uncertainties related to in-vessel melt progression and RPV failure.

19.2.4.4.2.2 Gate Failure

The phase of temporary melt retention in the reactor cavity ends when the melt gate fails and allows melt to relocate into the spreading compartment. Once the sacrificial concrete layer at the bottom of the reactor cavity is ablated, the melt comes in contact with the lower, metallic part of the melt plug, the so-called “gate”. The gate is mechanically supported by a dedicated structure that transfers the loads into the bottom of the transfer channel. The upper side of the gate is structured to provide a tight connection with the overlying concrete layer.

As the result of the heat flux entering its upper surface after contact, the gate heats up. Due to the limited capacity for heat loss from the lower surface, a final thermal-mechanical failure of the gate and the subsequent outflow of the accumulated melt into the spreading area are inevitable. For any reasonably expected initial contact pattern, the remaining concrete layer atop the gate will be completely ablated at the time the gate fails locally. This establishes favorable conditions for a widening of the initial hole as a result of the high heat fluxes to the periphery induced by the out-flowing melt and the focusing effect of the reactor cavity protective layer. Hole-widening effects make the discharge process self-adjusting – for a small initial opening, the duration of the discharge and with it, the time of interaction, will be correspondingly longer.

At the time the gate is exposed to the melt, the following kinds of melt layering in the reactor cavity (from top to bottom) are possible:

A. Three-layer system (a metallic/oxide mixture or slag, metallic corium, oxide corium).

B. Two-layer system (mixed oxide corium/slag, metallic corium).

C. Completely mixed melt with the metal dispersed within a mixed oxide corium.

In the first case, which corresponds to the situation before the oxide-over-metal layer inversion, the gate will first come in contact with the oxidic melt. In the second case, the gate will first come in contact with the metallic melt. In the third case, the mixed system is expected to behave like an oxide melt. However, the possibility of complete mixing is still uncertain, considering the differences in liquidus temperature and density between the oxide and metallic melt fractions and the related tendencies to separate from each other and to form an oxide crust at the interface. Therefore, two cases were analyzed: contact between the gate and metallic melt and contact between the gate and oxide melt. To determine the bounding duration of gate integrity, these two cases were evaluated using the lowest melt temperature, lowest decay heat level and lowest melt thickness found in the analyzed cases.

Figure 19.2-11—Temperature Profiles within the Gate after Contact with an Oxidic Melt, shows the resulting WALTER analysis with contact between the gate and the oxide melt. As a consequence of the lower heat fluxes involved, the gate heat-up takes between six and seven minutes to completely fail the gate.

Figure 19.2-12—Temperature Profiles within the Gate after Contact with a Metallic Melt, shows the resulting WALTER analysis with contact between the gate and the metallic melt. As a consequence of the higher heat fluxes involved, the resulting gate heat-up is fast, taking between 26 and 27 seconds to completely fail the gate.

The performed WALTER analysis of the melt gate shows that when the melt gate is contacted by the melt, the melt gate fails within a period from about 30 seconds to about seven minutes, with the more likely scenarios being nearer to the 30 second result. As a consequence of this rather short period, the melt state will not be impacted and the conditioned melt will flow unobstructed into the transfer channel.

19.2.4.4.2.3 Melt Spreading

After passing the gate, the melt flows through the transfer channel and pours onto the concrete-covered surface of the spreading compartment. Due to its large cross-section of about 10.76 ft^2 and its ceramic structure, the transfer channel does not have any retarding effect on this flow. The melt spreads under almost dry conditions because the spreading compartment design does not allow the possibility of a direct inflow of water from sprays or leaks. Only a limited amount of condensate may form inside the room. Though dry conditions are not required for a successful spreading, they make the distribution more predictable and eliminate the potential for fuel coolant interactions.

In this analysis, the conditioned corium and concrete mixture is transferred to and evenly distributed throughout the spreading compartment. Using the conditioned melt as described in Section 19.2.4.4.2.1, the MELTSPREAD analysis tool was used to determine the topography of the metallic and oxide layers of the melt in the spreading compartment. Sensitivity studies examined combinations of large and small melt masses and large and small gate failure areas. Results were not sensitive to melt mass; however, gate failure area does impact the spreading process. Figure 19.2-13—Corium Height as a Function of Distance from Melt Channel (100% Gate Failure), presents corium height as a function of distance from the transfer channel for a large melt pour with 100 percent of the melt plug failing. Figure 19.2-14—Corium Height as a Function of Distance from Melt Channel (10% Gate Failure) presents corium height as a function of distance from the transfer channel for a large melt pour with only 10 percent of the melt plug failing. Each of these figures shows the contour of the top of the oxide and metallic layers in the spreading compartment at 120 seconds, 300 seconds and 600 seconds into the transient. For the 100 percent gate failure cases, based upon the relative flatness of the top of the oxide and metallic layers shown in these figures, the corium and concrete mixture is evenly distributed throughout the spreading compartment as early as 120 seconds into the spreading transient. For the 10 percent gate failure cases, based upon the comparisons in the overall level of the top of the oxide layer shown in these figures, even though standing waves still exist, the corium and concrete mixture is distributed throughout entire surface area of the spreading compartment also as early as 120 seconds into the spreading transient.

All of the melt spreading transients performed show that within 10 minutes the top surface of the melt has effectively lost any presence of large standing waves. At two minutes, even before large standing waves have been eliminated, the corium and concrete mixture is evenly distributed throughout the spreading compartment indicated by the relative heights of the top of the oxide and metallic layers.

The size of the opening in the gate has a major effect on spreading. However, even with a 10 percent opening, the corium still spreads throughout the entire spreading area in two minutes and has leveled-out (no large standing waves) by 10 minutes.

19.2.4.4.2.4 Melt Flooding, Quenching and Long-Term Stabilization

The arrival of the melt in the spreading compartment triggers the opening of valves that initiate the gravity-driven flow of water from the IRWST into the spreading compartment. The water first fills the central supply duct underneath the spreading compartment. From there, it enters the horizontal cooling channels and then submerges the space behind the sidewall cooling structure. Finally the water pours onto the surface of the melt, which causes local quenching, solidification, and fragmentation. Overflow will continue until the hydrostatic pressure level between the IRWST and the spreading room is balanced.

In parallel with the progressing inflow of water, the spread melt interacts with the sacrificial concrete (i.e., MCCI) that covers the horizontal and vertical cooling plates. The concrete layer allows the spreading compartment to be cooled on the outside and on the melt surface by flooding before the melt contacts the cooling structure.

Containment overpressurization and basemat protection represent the credible containment failure modes of interest during this phase of a severe accident. These concerns are addressed through the steady heat removal from the melt pool moderated by passive means.

Containment Pressure Response

The containment pressure response during this period is primarily governed by heat transfer in the spreading compartment. As a consequence of flooding the spreading compartment, containment pressure increases proportionally to the heat transfer from the melt to the water. Initially, the direct contact of the melt with the flooding water converts all the water to steam resulting in a significant pressure excursion. Crust formation reduces the rate of steam generation; and, subsequently, the pressure rise. Eventually, heat transfer drops below that necessary to vaporize all the water entering the spreading compartment, and a water pool forms. As the liquid levels in the spreading compartment and the IRWST equalize, compartment flooding ends. The water pool temperature will rise to saturation; however, steam generation will occur at a much slower rate driven by the decay level, a level adequately mitigated by the U.S. EPR SAHRS.

Figure 19.2-15—Containment Pressure Following Gate Failure, shows the response of containment pressure following gate failure. This plot shows a peak appearing shortly after gate failure that corresponds to the moment when passive flooding coolant contacts the melt. A second peak follows corresponding to steady-state steaming that occurs following the water fill-up phase. The worst case in the uncertainty analysis has a maximum pressure in the containment of 74 psia. This is well below the containment ultimate pressure of approximately 119 psig (refer to Section 3.8.1.4.11).

Basemat Protection

Once the sacrificial concrete cover is ablated, the melt contacts the cooling structure. The thickness of the concrete at the bottom and walls allows the cooling structure to be completely flooded by the time melt contacts the cooling structure on the outside. During this stage of melt stabilization, the metallic melt is in contact with the cooling plate, and the decay power is mainly released in the oxide melt, which is located on top of the metallic melt. The principal concern with regard to basemat protection is the capacity of the cooling channels to remove the power it receives from the surface contact with the melt. Test program assessment of this design has confirmed that the cooling structure can manage heat loads in excess of 11.15 kW/ft^2.

The expected heat load has been determined using bounding melt composition results from the MAAP4-based uncertainty analysis as inputs in the transient one-dimensional heat conduction calculation of the WALTER code. As shown in Figure 19.2-16—Variation of Power into Cooling Channels, the maximum amount of power going into the cooling channels is 7.4 kW/ft^2, which occurs at approximately two hours into the transient.

The postulated bounding scenario presented here results in the temperature profiles within the melt and cooling plate shown in Figure 19.2-17—Temperature Profiles within the Cooling Plate (< 6 Hours), and Figure 19.2-18—Temperature Profiles within the Cooling Plate (>6 Hours, < 30 Days), as calculated by WALTER. The left and right sides relate to the upper surface of the spread melt and the water-contacted bottom of the cooling structure, respectively. The oxide melt, which is initially liquid, steadily cools down. Crusts on the top and bottom of the melt form first.

After about two days, the melt is practically solid. This corresponds to the parabolic temperature profile. The metallic melt underneath the oxide starts to freeze from the bottom immediately after contact with the cooling structure. The maximum temperature at the upper surface of the structure is 2037°F which is lower than the melting point of the structure (2141°F). Inside the bulk of the cooling structure there is a quasi-linear temperature profile.

Considering the results presented above where the worst-case scenario has margin between the temperature of the cooling structure and its respective melting point, the cooling structure is not in jeopardy of failing. In fact, the cooling structure does not ablate at all. As Figure 19.2-16 shows, the downward heat fluxes into the cooling structure always remains below 11.15 kW/ft^2. It can be concluded that the size and composition of the cooling structure provides sufficient margin to withstand any reasonably expected transient and steady-state thermal loads from the melt.

As it was shown in the previous analysis, the highest heat fluxes, and consequently the highest temperatures throughout the cooling plate, occur within the first day. In the long-term, the requirements on bottom cooling of the cooling plate will steadily decrease. After some time (potentially weeks or months), all heat can extracted from the melt through its upper surface.

19.2.4.4.3 High-pressure Melt Ejection

High-pressure core melt ejection is prevented by two manually-operated valve trains that are part of the PDS. Each of these valve trains consists of a DC-powered PDS valve in series with an isolation valve. Even though these valve trains are used exclusively for severe accident mitigation, a 2 x 100 percent design philosophy is followed to provide a performance margin. In addition, the consequences from a postulated HPME are mitigated by the containment layout which provides a tortuous pathway to the upper compartment, and no direct pathway for the impingement of debris on the containment shell.

The purpose of the HPME analysis is to evaluate the U.S. EPR response to the more likely severe accident scenarios. A successful response is characterized by meeting the following targets:

- The U.S. EPR maintains the capability to be depressurized after loss of decay heat removal.
- The reactor cavity contains ejected core debris and prevents it from impinging on the containment boundary.

19.2.4.4.3.1 RCS Pressure at RPV Failure

A reliable PDS is demonstrated by assessing the expected domain of RCS pressures at the time of reactor pressure vessel failure for the more likely severe accident scenarios. Given the design load on the reactor cavity of 290 psia, a successful depressurization is considered to be an RCS pressure below that level at the time of RPV failure. The robustness of this action has been demonstrated in this analysis by incorporating a degraded response by the operator. The prescribed response requires that at the moment in which a severe accident has been detected, corresponding to a core outlet temperature reading of 1200°F, the operator opens the PDS valves. Uncertainty in this setpoint was considered in the evaluation up to 1832°F coinciding with an additional delay of up to 15 minutes.

Following actuation of the PDS, the RCS begins to depressurize. The rate of depressurization is governed by the flashing of RCS inventory and the sizing of the PDS valves. For the domain of more likely severe accident scenarios, a minimum of 1.5 hours is available. Even with the degraded operator response time, the amount of time available for the PDS to reduce the RCS pressure is significant. While reliable on-time depressurization is expected by operator action, the analysis defines a margin for the success of the mitigation regarding the vessel failure pressure.

Figure 19.2-19—RCS Pressure at RPV Failure, presents the results from the severe accident uncertainty analysis for RCS pressure at the time of RPV failure. In all instances, the RCS pressure is below 203 psia.

19.2.4.4.3.2 Containment of Ejected Core Debris

While the use of the PDS valves is expected to eliminate the potential ejection of core debris, the reactor cavity is nonetheless designed to minimize core debris impingement on the containment boundary. The configuration of the reactor with the reactor cavity requires core debris from a failed RPV to follow a difficult path to reach the containment boundary. Without lower head penetrations, the expected lower head failure mode corresponding to the expected relocated debris state would be a local side wall failure near the oxide/metal melt pool interface. As a result, the majority of the entrained metallic melt would immediately de-entrain on the reactor cavity walls. The tortuous pathway to the upper containment atmosphere would prevent the necessary in-containment aerosol dispersal and long residence time required.

19.2.4.4.3.3 Containment Failure Probability

Tutu, Ginsberg, and Fintrok (Reference 12) identified an RCS pressure of 145 psia at RPV failure as the threshold under which HPME is extremely remote. The study presented in Section 19.2.4.4.3.1 produced a maximum RCS pressure at RPV failure that exceeds that threshold. As such, the possibility of containment failure by HPME cannot be eliminated solely by assessment of initial conditions prior to RPV failure. To further characterize the impact of HPME in the U.S. EPR, a separate probabilistic evaluation examines containment failure probability based on the methodology developed in Pilch, et al. (Reference 13).

Given the specific aims of Reference 13 (referred to here as the ‘NUREG model’) and that it did not specifically assess the U.S. EPR design (since the analysis was aimed at already operating power plants), the NUREG model results can only be used as an input to the evaluation. Mayer, et al. (Reference 14) provides another input since it reports on debris dispersion for the U.S. EPR geometry. By combining the load model of Reference 13 with the dispersion information of Reference 14, a credible estimate of the U.S. EPR specific situation with respect to DCH can be made.

The key measure in this analysis is the relative pressure increase. A regression model was defined such that it calculated the pressure increase due to DCH rather than the final absolute pressure. In so doing, it becomes possible to input U.S. EPR specific baseline pressures, taken from MAAP analyses, into the model. A regression model was generated by tabulating the following quantities for the 28 U.S. pressurized water reactors included in the evaluation:

- Zircaloy mass (total in core).
- Steel mass in lower plenum at vessel failure.
- Retention factor for lower sub-compartments.
- UO_2 mass (total in core).
- Cavity dispersion fraction.
- Coherence multiplier.
- Containment volume.

The regression model was validated against the NUREG model and found to be in close agreement. The retention factor for lower sub-compartments and the cavity dispersion fraction were based on experimental results reported in Reference 14, while the remaining parameters were supplied by MAAP4 calculations. The results from the analysis (refer to Section 19.1.4.2.1.2) conclude that the probability of DCH-induced containment failure as a result of HPME is very small. (< 1.0E-03).

19.2.4.4.3.4 Summary

For an HPME event to occur, the precursors (large core melt masses, high pressure, entrainment potential, affinity to form particulates, aerosol dispersal, long residence in containment atmosphere) all must be present; however, the assessment of more likely severe accident scenarios reveals that this condition is not possible in the U.S. EPR. The PDS valves can function under both ideal and degraded conditions to successfully reduce RCS pressure to benign levels. While the approach to severe accident mitigation in the U.S. EPR does not attempt to minimize the potential for large masses of molten core material to form following the onset of core damage, the other factors are difficult and unlikely to obtain.

19.2.4.4.4 Fuel–Coolant Interactions

As discussed in Section 19.2.3.3.5, the U.S. EPR addresses the impact of fuel-coolant interactions for the more likely scenarios by acknowledging the conclusions of various test programs and of expert groups established to review this phenomenon. Nonetheless, the principal reactants, water and high temperature metallic and oxide masses, are present and configurations can be imagined that result in FCI-developed containment loads that challenge containment integrity (i.e., in-vessel scenarios) or cavity structure loads that can damage the U.S. EPR CMSS effectiveness (i.e., ex-vessel scenarios). Because of the large uncertainties associated with low probability events such as FCI, the likelihood of containment failure by steam explosion cannot be resolved through traditional deterministic analysis; rather, a probabilistic evaluation was developed.

For the probabilistic evaluation, separate analyses were performed assessing containment or cavity structure failure probability from either in-vessel or ex-vessel scenarios (see Section 19.1.4.2.1.2). The key measure in these analyses is load energy. This is a strong function of the amount of corium mass relocating to the lower head (or into an ex-vessel pool) and the fraction interacting with the available water, its temperature, and several “efficiency” terms. For the in-vessel scenario, this energy is assumed to be transferred to a slug of water which impacts the reactor vessel upper head and the containment or to the lower head which detaches and impacts the reactor cavity. For the ex-vessel scenario, this equates to a direct impulse loading on cavity structures.

The parameters addressed probabilistically were:

- The fraction of core which is involved in pre-mixing.
- The conversion ratio for thermal to mechanical energy.
- The fraction of the mechanical energy transmitted to a water slug (in-vessel only).
- Load bearing capacity of the upper head (in-vessel only).
- Relative frequency of occurrence of a steam explosion given a melt pour into coolant (pressure dependent).

The uncertainty distributions were developed based on expert opinion, such as that appearing in SERG-2 report (Reference 6), experiments, or analysis. Values for corium mass were taken from MAAP4 calculations.

The results from the analysis conclude that the probability of in-vessel steam explosions leading to containment by alpha-mode failure or reactor cavity damage by a lower head missiles was very small (< 1.0E-03). In addition, the ex-vessel scenario is not credible for containment failure because of its proximity to the containment boundary and that the probability of reactor cavity structure failure was also very small (< 1.0E-04).

19.2.4.4.5 Equipment Survivability

A severe accident involves a variety of phenomena that have the potential to generate considerable loads on the RCS, the containment, and its structural internals. Therefore, the U.S. EPR design has severe accident mitigation features, which exclude, reduce and maintain the severe accident loads below critical values to maintain the integrity of the containment. To function properly, these severe accident mitigation features must withstand the environmental conditions present during a severe accident. Equipment survivability, which is the ability to perform the intended function when needed, depends on the period of time necessary to operate and the local conditions during the time period.

Reference 3 and Reference 4 state that features provided solely for severe accident protection need not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirements of 10 CFR Part 50, Appendix B, or redundancy/ diversity requirements of 10 CFR Part 50, Appendix A. However, the guidelines go on to state that reasonable assurance must be provided that mitigation features will operate in the severe accident environment (e.g., pressure, temperature, radiation) for which they are designed and for the time span that the equipment is needed.

As outlined in Reference 1, severe accidents are monitored and mitigated with appropriate instrumentation and components in order to perform operator actions, survey the effectiveness of the installed mitigation measures, and survey the overall plant conditions including possible releases to the environment during a severe accident.

The equipment survivability analysis defines the following:

- Equipment required to operate during a severe accident.
- Length of time the equipment is needed.
- Local conditions in which the equipment must be able to operate.

The equipment survivability analysis uses results from the MAAP4-based uncertainty analysis to determine the local conditions that the equipment experience. The results of the uncertainty analysis provide a tolerance range of minimum and maximum values for the more credible severe accident scenarios.

The environmental conditions vary locally within the containment during a severe accident. Equipment type is grouped based on its location and the common environmental conditions that result. Each location category identifies the equipment required to operate in a severe accident, the time it is required to function and the local conditions that are likely to be experienced. The four location categories are:

- Incore/In-RCS.
- Inside containment.
- SAHRS compartments (in Safeguard Building 4).
- Reactor Building Annulus.

19.2.4.4.5.1 Incore/In-RCS

Any component located inside the core or RCS required during a severe accident needs to remain operable until RPV failure. The instrumentation required and relied upon for the in-vessel phase of the accident progression are:

- Wide-range core outlet thermocouples.
- RCS pressure sensors.

- PDS valves.

The pressure and temperature requirements are summarized in Figure 19.2-20—Bounding Incore Temperature and Pressure Development during a Severe Accident. The time between each of the events varies and is dependant on the scenario. As such, the given values are a composite of the minimum and maximum values at the particular event times as determined from the uncertainty analysis; that is, the time development between events is not linear.

The displayed measurement range of the wide-range core outlet temperature thermocouples will be at least up to 2282°F and their qualification up to 1832°F. With respect to pressure, the thermocouples will be qualified up to 2900 psia. As the severe accident progresses after depressurization of the RCS, the temperature at the location of the core outlet increases beyond the thermocouple qualification range, thus leading to their anticipated failure before RPV failure.

In order to have continuous information on RCS pressure, the pressure sensors should remain available until RPV failure and deliver correct values even if the hot leg and pressurizer gas temperatures have exceeded 2192°F.

The PDS valves are qualified for temperatures up to 1112°F for the reliable opening of the valves, which occurs only once at the 1200°F core outlet temperature signal. Because the PDS discharge capability is required until RPV failure, the PDS valves must remain open and their position reliably indicated up to pressurizer gas temperatures of 1832°F. On the outside, the PDS valves and their position sensors will be qualified to withstand the environmental conditions inside the containment during a severe accident.

19.2.4.4.5.2 Inside Containment

The following equipment and instrumentation are positioned inside containment and must withstand the conditions expected to occur during a severe accident:

- Containment isolation valves and position sensors.
- PDS valves and position sensors.
- Containment pressure sensors.
- H_2 Monitors.
- Hydrogen mixing dampers and position sensors, PARs, convection and rupture foils.
- IRWST water level and temperature.
- Dose rate measurement (i.e. gamma-sensitive detector).
- Severe accident sampling system.

- Thermocouples inside insulation liner to measure temperature of RPV lower head.
- Flooding valves and position sensors.
- Thermocouples in core catcher main cooling channel and steam chimney.
- Containment spray nozzles.

The time span for which the equipment is to remain operable (mission time) is also important. This is summarized in Figure 19.2-21—Course of Primary Events during a Severe Accident, which gives a schematic representation of the course of main events and maximum pressure and gas temperature during a severe accident. The graph is a composite of all 59 uncertainty analysis cases. As with Figure 19.2-20, the given values are a composite of the minimum and maximum values at the particular event times as determined from the uncertainty analysis; that is, the time development between events is not linear.

Pressure, Temperature, and Humidity within Containment

The pressure, temperature, and humidity time developments during a severe accident were evaluated from the MAAP4-based uncertainty analysis. The maximum "global" containment pressure and temperature that equipment and instrumentation may be exposed to during the progression of a severe accident are 76.9 psia and approximately 410°F, respectively. The maximum humidity inside the containment experienced by the equipment conservatively can be assumed to be 100 percent after the commencement of spraying. However, due to the existence of other gases inside containment, the steam concentration approaches a conservative value of 80 percent. Finally, the IRWST reaches a maximum temperature of 257°F. The SAHRS is conservatively designed for a maximum IRWST water temperature of 320°F.

For the relevant scenarios, which form the basis of the uncertainty analysis cases, localized hydrogen detonation and deflagration can be reliably excluded. However, because the highest AICC pressure and temperature resulting from representative and bounding scenarios is 105 psia and 1634°F, respectively, it is necessary to assess the equipment and instrumentation capabilities within this extended operational range. The AICC pressure is a purely theoretical value that cannot be reached, because actual combustion is neither adiabatic, isochoric, nor complete.

While all equipment and instrumentation inside containment may be exposed to such pressure and temperature spikes, only equipment relied upon to actively mitigate the consequences of hydrogen in the containment atmosphere is required to survive such occurrences per 10 CFR 50.44. Therefore, the hydrogen mixing dampers and PARs must be capable of surviving such short lived pressure and temperature spikes. During the recombination process, the PARs can experience localized temperatures that are well above 1832°F and must therefore be able to adapt to high temperatures. Based on the design, the PARs are not pressure retaining components and are open at the bottom and the top. Hence, the PARs are unaffected by localized pressure increase. Similar arguments can be made for the hydrogen mixing dampers and the rupture and convection foils. Because those open on pressure differential and, in the case of the convection foils on temperature differential, their operation is also not affected by localized pressure and temperature increase due to hydrogen combustion.

Radiation within Containment

A deterministic analysis of the direct dose radiation environment in the U.S. EPR buildings, as well as the submersion dose for accident conditions, was performed for the U.S. EPR. The analysis basis for the determination of the radiation exposure levels of severe accident equipment is a LOCA in one of the reactor coolant lines. In severe accident conditions the complete radioactive inventory of the core is released into the containment. The release rates for such a case are used to determine radioactivity concentrations for equipment survivability determinations. The analysis is based on realistic assumptions for partitioning of the fission product groups between sump water and containment atmosphere. Decay data are then used to convert these activities into beta and gamma source strengths. The radiation levels relevant for qualification inside the containment during a severe accident assume that the fraction of the core inventory released to the containment is distributed uniformly both in the atmosphere and on the walls. The corresponding cumulated dose values based on highly conservative assumptions (e.g., no wash out) and adjusted by the weighted average factor, and are presented below:

- Dose due to air-borne gamma radiation after 24 hours: 399 kGy.
- Dose due to air-borne gamma radiation after one year: 6670 kGy.
- Dose due to deposition gamma radiation after 24 hours: 385 kGy.
- Dose due to deposition gamma radiation after one year: 6815 kGy.
- Dose due to gamma radiation after one year due to the activity inventory of the IRWST (calculation dose point is at the water level): 4640 kGy.

The presented dose values above are maximum values based on conservative assumptions. The calculated dose for one year implies that for one year the conditions inside the containment are nearly constant. Only radioactive decay is considered as the effect leading to the decrease of the activity inventory. Other effects like transport processes of nuclides from the containment atmosphere into the IRWST water, where a part of radioactive nuclides would be retained, are not considered, meaning the local dose rates will be lower because of shielding by existing walls and structures.

Reactor Cavity

It has been postulated that the failure of the RPV and the subsequent dropping of mass onto the reactor cavity may cause structural damage, which could impair the ability to mitigate the effects of a severe accident. To demonstrate that the reactor cavity is designed to withstand the RPV failure and continue to function properly, a structural study of the reactor cavity was performed.

The objectives of this study were to demonstrate analytically that a RPV failure does not cause impairment of the melt retention capability, a loss of structural integrity, or breach of the liner. This study assessed the structural integrity of the reactor cavity walls and floor due to the mechanical and thermal loading caused by this accident. This study included:

- 290 psia overpressure (note that the results of the MAAP4-based uncertainty analysis showed the maximum RCS pressure at the moment of RPV failure to be less than 200 psia).
- The impact load of the detached RPV lower head plus contained melt on the bottom structure of the reactor cavity.
- Temperature transient due to heat diffusion through the protective layer.
- Thermal radiation from the surface of the molten pool.
- Convective heat from gas during MCCI.

Based on the numerical results, which include post-cracking regime in concrete and large deformations and plasticity in the impacting RPV shell and rebars, the objectives of the study were met with a significant margin to failure. The peak reactor cavity floor load due to missile impact is found to be only about 40 percent of the ultimate capacity of the concrete pads and about 20 percent of the punching shear capacity of the floor and basemat. Therefore, the melt plug integrity is not impaired. Furthermore, the kinetic energy of the lower RPV missile is only about five percent of the post-cracking energy absorbing capacity prior to liner rupture.

The maximum overpressure capacity is 420.5 psia. The crack patterns at 290 psia overpressure indicate through-cracking but no concrete crushing that diminishes the temporary retention and condition function of the reactor cavity. The tangential rebar stresses indicate stresses at yield level, while the vertical and radial stresses are within elastic limits. The crack patterns in the structural concrete due to the thermal transients indicate extensive cracking but no concrete crushing. The tangential and vertical rebar stresses are at or near yield level. The radial rebar stresses are elastic.

Because the reactor cavity is capable of withstanding the impact loads and thermal transients induced by the RPV failure, it is able to perform its intended function without impairment.

19.2.4.4.5.3 Safeguard Building – SAHRS Compartments

The equipment, instrumentation, and severe accident control functions belonging to this category are:

- Actuation of SAHRS and its associated systems (ESWS, CCWS, containment isolation valves).
- SAHRS heat exchanger.
- SAHRS inlet and outlet temperature sensors.
- SAHRS pump and associated pressure sensors.

- SAHRS volume flow rate sensor.
- SAHRS containment isolation and instrumentation valves.

With the exception of the spray system, the SAHRS equipment is located in Safeguard Building 4, which is not exposed to any severe accident-related conditions until the SAHRS is activated. After start of the SAHRS, the contaminated IRWST water flows through the system, having a maximum temperature of 257°F. The standard boron concentration in the IRWST is 1700 ppm ± 100ppm.

Based on the containment radiation analysis (Section 19.2.4.4.5.2), the cumulative dose in the SAHRS compartments is the result of:

- Gamma radiation dose due to a pipe of the SAHRS system in operation (water is assumed to be drawn from the IRWST).
- Gamma radiation dose in a SAHRS compartment caused by spread fluid on the floor due to an assumed pipe leakage.

The main results are:

- Dose due to gamma radiation from the pipe in 3.28 ft distance after one year: 131 kGy.
- Dose due to gamma radiation in a SAHRS compartment caused by leaked fluid on the floor after 100 hours: 334 kGy.

In the case of the SAHRS room, the selection of a reference dose point 3.28ft away from the pipe takes into consideration the layout of the pipe branches and the room. The simultaneous radiation from two pipes, one on the pump suction side and one on the pump pressure side, is considered for the expected gamma dose. Inside the heat exchanger and valve room a similar situation exists with regard to the pipes running to and coming from the heat exchanger. Additionally, injection into the containment is possible via different paths depending on the SAHRS operating mode – spraying, active cooling of the spreading area, or back flushing.

Taking these facts into account, it is assumed that irradiation can take place from two sides. Qualification in these rooms doubles the dose from 130 kGy to 260 kGy. The dose caused by spilled fluid after a pipe break or leakage is not relevant for equipment qualification of SAHRS components because such an event will probably result in a long-term inoperability of the affected SARHS components.

19.2.4.4.5.4 Reactor Building Annulus

The instrumentation belonging to this category are:

- Annulus pressure.
- Dose rate downstream of annulus ventilation system filters.
- Volume flow rate downstream of annulus ventilation system filters.

The annulus is designed against a maximum overpressure of 16 psia, which is limited by the strength of the doors. The temperature within the annulus does not exceed initial conditions significantly as a result of a severe accident. The cumulated dose inside the annulus is assessed to be 144 Gy after 24 hours and 3168 kGy after one year.

19.2.4.4.5.5 Equipment

Table 19.2-3 provides a listing of severe accident instrumentation and equipment necessary to monitor the severe accident progression and to allow for operator action.

19.2.4.4.5.6 Summary

In the event of a severe accident, environmental conditions inside the RCS, the containment, the SAHRS compartments, and the annulus can be harsher than during a design basis accident. The instrumentation and equipment identified herein are relied upon to mitigate the consequences of a severe accident during these beyond design-basis accidents. The environmental conditions are used to specify the conditions in which the equipment is required to operate. By using equipment that is qualified for use in these beyond design basis accident conditions, the U.S. EPR reliably minimizes the consequences of a severe accident and prevents containment failure.

19.2.4.5 Conditional Containment Failure Probability

The conditional containment failure probability is presented in Section 19.1.4.2.2.1.

19.2.4.6 Summary

The containment performance analysis shows that the containment maintains its role as a reliable, leak-tight barrier for at least 24 hours following the onset of core damage for the following severe accident challenges.

Hydrogen levels are kept sufficiently low to preclude containment failure by global deflagration and meets the 10 CFR 50.34(f)(2)(ix) requirement that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal water reaction.

The corium is reliably conditioned in the reactor cavity to promote spreadability in the spreading compartment after melt gate failure. The core melt stabilization system transfers the corium into a coolable geometry within the spreading compartment, thus providing sufficient removal of residual decay and long-term stabilization.

The U.S. EPR design, which incorporates several design features with enhanced preventive response to an HPME, precludes the potential mechanisms for HPME initiation and subsequent DCH.

Design characteristics of the U.S. EPR inherently impede the potential for steam explosion-induced containment failure because the necessary conditions required for steam explosions to exist are avoided.

Instrumentation and equipment that are relied upon to mitigate the consequences of a severe accident are qualified for use in beyond design basis accident environmental conditions.

19.2.5 Accident Management

The goal in managing an accident that exceeds the design basis is to return the plant to a controlled state in which the nuclear chain reaction is essentially terminated, continued fuel cooling is ensured and radioactive materials are confined. Accident management includes taking full opportunity of existing plant capabilities, if necessary going beyond the originally intended functions of some systems and using some temporary or ad hoc systems to achieve this goal. Accident management is responsive to the specific circumstances of the event, even though they might not have been anticipated. Advantage is taken of whatever time might be available between correct diagnosis of the symptoms and the impending release of fission products to the environment. For the diagnosis of beyond design basis events and the execution of accident management activities, somewhat longer periods than those for design basis accidents could be available to the operating staff.

Severe accident management encompasses those actions taken during the course of an accident by the plant operating and technical staff to:

- Prevent core damage.
- Terminate the progress of core damage if it begins and retain the core within the reactor vessel.
- Maintain containment integrity as long as possible.
- Minimize offsite releases.

In principle, severe accident management extends the defense-in-depth philosophy to the plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel damage regimes, using existing plant equipment, operator skills, and personnel creativity to terminate severe accidents and limit potential offsite consequences.

The SAMGs address the recognized need to provide nuclear power plant technical staff with structured guidance for response to a potential severe accident condition involving core damage and potential release of fission products to the environment. AREVA NP has developed a new approach to SAMGs in a project called Operating Strategies for Severe Accidents (OSSA). The OSSA framework makes maximum use of the lessons learned to date in the field of severe accidents and incorporates a number of new features which simplify and streamline the guidance material while maintaining comprehensive guidance for response to any severe accident.

The purpose of this section is to describe the OSSA framework for the U.S. EPR SAMGs. The high-level actions that would need to be taken to mitigate severe accidents are described in the context of the unique severe accident design features of the U.S. EPR. The potential challenges that need to be addressed by the technical support center team and the OSSA diagnostic tool used to mitigate these challenges are described.

As stated in FSAR Section 19.1.2.2, the COL applicant will review final plant-specific EOPs and SAMGs to confirm that the assumptions used in the PRA and severe accident analyses remain valid.

19.2.5.1 Accident Management through Design

Severe accident management in the U.S. EPR begins with several design elements specifically addressing the stated objectives of maintaining fuel, RPV, and containment integrity while minimizing radiological releases. These design elements have been described in Section 19.2.2 and Section 19.2.3.

19.2.5.2 OSSA Directed Actions

The ultimate goal for the OSSA is to provide mitigation strategies to cover all potential events that lead to core melt and to stop or reduce the releases of fission products to the environment.

Considering containment challenges rather than accident scenarios promotes protection of the containment as priority in every case regardless of the accident sequence. The OSSA considers a broad range of sequences, even if not analyzed or quantified through the PRA Level 2 or through the supporting safety studies. For the severe accident sequences occurring in the Fuel Building, building failure is not a concern due to the leakage rate and high degree of permeability of the structure. In this case, the building-defined challenges are the phenomena that can lead directly to large radioactive releases.

The OSSA diagnostic is developed based on a list of possible challenges to severe accident mitigation and the corresponding instrumentation used to assess safety margins. Subsequently, appropriate actions can be derived, predicated on which challenges may be present at a given moment. The technical bases for such actions can be developed in most instances through deterministic process studies using the U.S. EPR MAAP4.0.7 model. By using the framework defined in this section, the SAMGs are developed which include specific U.S. EPR diagnostics, definition of necessary associated instrumentation, and the development of high level and in-depth mitigation strategies to be used by the technical support center during a postulated severe accident. Table 19.2-4—SAMG Technical Basis–Mapping Challenge Mechanism to Operator Action, provides several examples mapping challenge mechanisms to operator action.

19.2.5.3 Interface with Emergency Procedures

The prevention of core damage is considered to be within the domain of EOPs. The review of existing approaches, together with the consideration of advanced reactor design measures, led to the choice of a ‘standalone’ severe accident management guidance. Once a specified set of plant conditions is met, use of EOPs is abandoned and control switches to the OSSA. For the U.S. EPR, it has also led to the decision to treat primary system depressurization as an immediate action. However, since primary system depressurization performs both preventive and mitigative functions, a consideration is needed which ensures that if the preventive function of depressurization is successful, control remains in the EOPs.

19.2.5.4 Measurable Safety Objectives

Equipment, instrumentation and diagnostic aids are available to plant operators, who may at some time be faced with the need to control the course and consequences of an accident beyond the design basis. In OSSA, the diagnostic tool recognizes the presence of systems and instrumentation designed and qualified for the mitigation and monitoring of severe accidents. A severe accident sequence in which the dedicated severe accident measures perform as designed is described as following the “mitigation path.” Qualified severe accident instrumentation is sufficient to monitor plant conditions and determine whether the accident is on the mitigation path through explicit measures of safety margins on the various fission product barriers. While the accident remains on the mitigation path there is, by design, no challenge to the ultimate fission product barrier. However, certain manual actions are taken so that the event remains on the mitigation path. Also, strategies which could arrest the core melt and terminate the event before vessel breach are evaluated and implemented if available and if the evaluation process leads to a recommendation to implement the strategy.

The development of abnormal plant behavior following equipment malfunction or operator error could be rapid in some circumstances. The operating staff would then have to diagnose the cause quickly and plan appropriate corrective action. Equipment is provided especially to assist in this. It comprises instrumentation reading out in the control room, environmentally qualified and capable of providing the information needed to recognize abnormal conditions, to correct faults and to determine the effects of corrective action. Examples of instrumentation provided specifically for accident management are coolant inventory trending systems, monitors for very high containment pressure, hydrogen monitors and monitors of activity in primary coolant. Table 19.2-3 presents a list of the severe accident instrumentation and equipment.

19.2.5.5 RPV Integrity Management

The U.S. EPR employs an ex-vessel severe accident mitigation strategy. As such, it has been designed to preserve the containment as the ultimate fission product barrier for scenarios in which a full core and associated structure has been released from the RPV. Along the path leading to the declaration of a severe accident, core coolability will steadily degrade. With the declaration of a severe accident (i.e., core exit temperature greater than 1200°F), application of the EOPs have failed for the plant at the current state. Opening of the PDS valves provides a time window in which the introduction of core cooling can continue by employing all means available to the operators; however, at the moment in which the core exit temperatures are above 1200°F and RCS pressure is been fully reduced, the core coolability function has been lost. At that moment, the severe accident is mitigated through the ex-vessel strategy outlined in Section 19.2.3.2.2.

19.2.5.6 Post-RPV Failure – Short-Term Response

The U.S. EPR has been designed with severe accident mitigation features specifically addressing ex-vessel stabilization of core melt. These features are passive in nature for 12 hours following the onset of a severe accident and the operator response for initiating RCS depressurization. By the end of the 12 hour period, the melt is expected to have been transferred into the spreading room where water passively delivered from the IRWST will reside in a pool above the spread melt.

Two objectives are met during this period: decay heat is being removed and silicates within the concrete have mixed with the heavy fuel oxides to encase fission products as the melt stabilizes.

19.2.5.7 Post RPV Failure – Long-Term Response

Beyond this 12 hour period, the event moves into the long-term cooling phase which involves active operator response beginning with the actuation of the SAHRS. During this event phase the spreading compartment, adjoining chimney vent, and reactor cavity are flooded. The flooding response serves to both remove decay heat and contain fission products. The SAHRS also controls containment pressure through condensation of resident water vapor.

The melt is considered to be stabilized when it is determined that the corium is no longer a threat to containment integrity as evaluated through various containment compartment pressure and temperature measurements. Further remediation of the severe accident then abides by the ALARA principles to minimize radiological consequences to both plant personnel and the surrounding environment.

19.2.6 Consideration of Potential Design Improvements under 10 CFR 50.34(f)

19.2.6.1 Introduction

The purpose of the severe accident mitigation design alternatives (SAMDA) analysis is to review and evaluate plant design alternatives that could significantly reduce the radiological risk from a core damage event resulting from a postulated severe accident. Plant changes are evaluated that would reduce the likelihood of a core damage event and that could mitigate the consequences should such an accident occur. This section summarizes the principle conclusions presented in Reference 15, AREVA NP Environmental Report Standard Design Certification. The U.S. EPR Design Certification Environmental Report details the costs and benefits of severe accident mitigation design alternatives, and the associated bases for not incorporating severe accident mitigation design alternatives in the design to be certified.

19.2.6.2 Estimate of Risk for Design

As presented in Section 19.1.8, the U.S. EPR CDF is well below the USNRC goal of 1.0E-04 per reactor-year.

19.2.6.3 Identification of Potential Design Improvements

The candidate SAMDAs are defined as potential design improvements to the U.S. EPR plant design that have the potential to prevent core damage and prevent significant releases from containment. The comprehensive list of candidate SAMDAs were developed for the U.S. EPR by reviewing industry documents and considering plant-specific enhancements not considered in published industry documents. Since the U.S. EPR is an evolutionary pressurized water reactor (PWR), particular interest was paid to existing SAMDA candidates for PWRs. Therefore, the primary industry documentation supporting the development of U.S. EPR candidate SAMDAs is as follows.

- Severe Accident Mitigation Alternatives (SAMA) Analysis, NEI Guidance Document (Reference 16).
- NUREG/BR-0184. “Regulatory Analysis Technical Evaluation Handbook” (Reference 17).

The top 100 Level 1 cutsets, representing approximately 33 percent of the total CDF for the U.S. EPR, were evaluated to identify plant-specific modifications for inclusion in the comprehensive list of candidate SAMDAs. The individual cutsets below this point each individually contribute less than 0.11 percent to the total CDF. Therefore, these cutsets have little influence on the CDF and are not likely contributors for identification of cost beneficial enhancements to the U.S. EPR.

An extensive evaluation of the top 100 cutsets was completed in order to make sure all possible design alternatives for the U.S. EPR were addressed. Through the evaluation numerous U.S. EPR specific operator actions and hardware-based SAMDAs were developed. Several generic SAMDA candidates from NEI 05-01 (Reference 16) and NUREG/BR-0184 (Reference 17) were determined to be applicable to the U.S. EPR through the evaluation of the PRA Level 1 cutsets. Therefore, these SAMDA candidates were not duplicated in the analysis.

As stated in FSAR Section 19.1.2.2, the COL applicant will review the plant-specific PRA results and cutsets to confirm that the conclusions of the SAMDA remain valid.

Since most of the SAMDAs were derived from the Reference 16 NEI guidance document, they include a wide variety of potential enhancements that may or may not be applicable to the U.S. EPR. In addition, several candidate SAMDAs initially considered may or may not have already been included in the design of the U.S. EPR. Prior to initial consideration, many of the candidate SAMDAs were not examined for applicability to the U.S. EPR. Therefore, each of these SAMDAs were screened by using one of the seven categories discussed below. These categories are the suggested categories from Reference 16.

- Not Applicable: The candidate SAMDAs were identified to determine which are definitively not applicable to the U.S. EPR. Potential enhancements that are not considered applicable to U.S. EPR are those developed for systems specifically associated with boiling water reactors (BWR) or associated with specific PWR equipment that is not in the U.S. EPR design. For example, the candidate

SAMDAs that address pneumatic MSRVs are not applicable due to the fact that the U.S. EPR MSRVs are motor driven. It should be noted that simply because a modification was intended for a BWR, ice condenser containment, or other system that is not applicable to the U.S. EPR, each SAMDA was thoroughly reviewed to make sure that every potential modification similar in intent, and applicable to the U.S. EPR design, was identified.

- Already Implemented: The candidate SAMDAs not dispositioned as being Not Applicable were reviewed to make sure that the U.S. EPR design does not already include features recommended by a particular SAMDA. It also may have been found that the intent of a particular SAMDA was fulfilled by another design feature or modification. In these cases the candidate SAMDAs are already implemented in the U.S. EPR plant design. If a SAMDA candidate has already been implemented at the plant, it is not retained. For example, the U.S. EPR has 47 PARs installed throughout containment, which passively actuate when a threshold hydrogen concentration is reached. This satisfies the SAMDA that calls for the addition of a passive hydrogen control system.

- Combined: If one SAMDA candidate is similar in nature to another SAMDA candidate, and can be combined with said candidate to develop a more comprehensive or plant-specific SAMDA candidate, only the combined SAMDA candidate is retained for screening. For example, “installation of a independent active or passive high pressure injection system” and “provide an additional high pressure injection pump with independent diesel” provide similar risk-reduction benefits. Therefore, these SAMDAs are evaluated in conjunction with each other.

- Excessive Implementation Cost: The methodology described in Reference 16 and Reference 17 was applied for the calculation of the maximum benefit obtained by the elimination of all severe accident risk and serves as the cost for screening out potential plant modifications. For the U.S. EPR the maximum benefit was calculated to be $50,642. A rounded maximum benefit of $51,000 was chosen for the U.S. EPR. SAMDA candidates that exceed the maximum benefit of $51,000, even without an implementation cost estimate, incur an excessive implementation cost and are not retained.

  – For example, the cost of installing an additional, buried off-site power source would exceed the maximum benefit just discussed and would not require further analysis. Consideration should be given to lower cost alternatives, such as temporary connections using commercial grade equipment (i.e., portable generators and temporary cross-ties), procedure enhancements and training enhancements that could offer a potential risk reduction at a fraction of the cost of safety-related modifications.

- Very Low Benefit: If a SAMDA is related to a non-risk-significant system for which change in reliability is known to have negligible impact on the risk profile, it is deemed to have a very low benefit and is not retained. There are two ways to determine the risk impact for the U.S. EPR:

  – A PRA Level 1 importance list is used to determine if a given system is risk significant for the U.S. EPR. If a SAMDA candidate is associated with a system that is not included on the importance list, it can be concluded that the design alternative would have a negligible impact on the risk profile, and it is not retained.

– If a SAMDA candidate can be shown to have a minimal impact on CDF, it is not retained.

- Not Required for Design Certification: Evaluation of any potential procedural or surveillance action SAMDA enhancements are not appropriate until the plant design is finalized and the plant procedures are being developed. If a SAMDA candidate is related to any of these enhancements, it is not retained for this analysis.

- Considered for Further Evaluation: Following the screening process, if a particular SAMDA is not categorized by any of the preceding categories, the SAMDA is considered for further evaluation and is subject to a cost-benefit analysis.

19.2.6.4 Risk Reduction Potential of Design Improvements

A total of 167 SAMDAs developed from industry and U.S. EPR documents were evaluated in this analysis.

- Twenty-one candidate SAMDAs were Not Applicable to the U.S. EPR design.

- Sixty-seven candidate SAMDAs were Already Implemented in the U.S. EPR design either as suggested in the SAMDA or an equivalent replacement that fulfilled the intent of the SAMDA. These are summarized in Table 19.2-5—SAMDA Candidates – Already Implemented.

- Four candidate SAMDAs were Combined with another SAMDA because they had the same intent.

- Twenty-three candidate SAMDAs were categorized as Excessive Implementation Cost.

- One candidate SAMDA was categorized as Very Low Benefit.

- Fifty-one candidate SAMDAs were categorized as Not Required for Design Certification because they were related to procedural and surveillances actions.

- None of the SAMDA candidates were categorized as Considered for Further Evaluation.

19.2.6.5 Cost Impacts of Candidate Design Improvements

Several of the SAMDA candidates were evaluated to determine the cost impact of implementing a particular candidate. The implementation cost of candidates was determined by using the implementation cost provided in recent license renewal applications for the same SAMDA candidate. The implementation costs obtained were not modified for inflation.

19.2.6.6 Cost-Benefit Comparison

The SAMDA candidates placed in the Considered for Further Evaluation category require a cost benefit evaluation. No candidates were identified for this category; therefore, a cost-benefit evaluation was not required for the U.S. EPR design.

19.2.6.7 Conclusions

As demonstrated by PRA, the low probability of core damage events in the U.S. EPR coupled with reliable severe accident mitigation features provide significant protection to the public and the environment. A detailed analysis of specific severe accident mitigation design alternatives from previous industry studies and from U.S. EPR PRA insights was performed against broad acceptance criteria. None of the SAMDA candidates met the criteria; therefore, the overall conclusion is that no additional plant modifications are cost beneficial to implement due to the robust design of the U.S. EPR with respect to prevention and mitigation of severe accidents.

19.2.7 References

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Table 19.2-1—Chronology of a Bounding Severe Accident through RPV Failure

Event	Comment
Large Break LOCA and SBO	Assumed initiating event
Core Refill	Large amount of water is bounding for hydrogen generation
Injection Fails	Complete failure necessary for maximum fuel melt
Fuel Heat up, Failure, and Melting	N/A
Contact with Core Support Structure and Heavy Reflector	Slow melt of heavy reflector retains heat in the core region, enhancing transition to liquid phase
Melt Progression towards the Lower Head	Convective heat transfer enhances destruction of core and supports
Debris Formation from Contact with Water in the Lower Head	Crust/debris formation insulates molten pool from vessel wall, retention allows separation of oxide and metal
Molten Pool Formation	
Lower Head Failure	Flow of metallic melt from the RPV further degrades the RPV, spilling the entire core into the reactor cavity

Table 19.2-2—SAHRS Design and Operating Parameters

Parameter	Value
SAHRS Pump	
Type	Single-stage centrifugal
Nominal flow rate	232 lb/sec
Nominal discharge head	498 ft
Design pressure	436 psig
Design temperature	324°F
SAHRS Heat Exchanger (Tube Side)	
Nominal flow rate	208 lb/sec
Tube material	Austenitic stainless steel
Design pressure	436 psig
Design temperature	324°F
SAHRS Heat Exchanger (Shell Side)	
Nominal flow rate	307 lb/sec
Shell material	Ferritic steel
Design pressure	500 psig
Design temperature	215°F
Containment Spray Nozzles	
Number	75
Nominal flow rate	208 lb/sec
Design pressure	436 psig
Design temperature	324°F
Passive Outflow Restrictor	
Maximum Passive Flow Rate	220 lb/sec

Table 19.2-3—Severe Accident Instrumentation and Equipment
Sheet 1 of 4

Function	Measured quantity/ Activated device	Type of device	Location	12h Batt.	SBO
RCS Depress. & Monitoring					
	Core outlet temperature(COT)	Thermocouple	Incore fingers	x	x
	RCS pressure wide range	PDE	inlet RHR to RCS	x	x
	RCS pressure narrow range	PDE	inlet RHR to RCS	x	x
	PDS Actuation	Valve Actuator	PDS/PRZ	x	x
	PDS position	Position sensor	PDS/PRZ	x	x
Corium Position					
Threat of RPV meltthrough	RPV outside-wall temperature	Thermocouple	RPV insulation	x	x
Corium arrival in core catcher	Temperature in chimney above spreading area	Thermocouple	"chimney" above spreading area	x	x
Corium arrival in core catcher	Position of flooding valve	Position sensor	flooding valve - SAHRS dedicated valve compartment	x	x
Corium arrival in core catcher	Flow downstream of Passive flooding valve	Flow rate meter	SAHRS dedicated valve compartments	x	x
Threat of basemat penetration	Temperature in basemat main cooling channel, or failure of thermocouple	Thermocouple	main core catcher cooling channel	x	x
Hydrogen Mitigation & Monitoring					
Hydrogen Combustion	Local/global hydrogen concentration	Containment Atmosphere Monitoring System	Safeguard Buildings 1 and 4	x	x
	Local/global hydrogen concentration	Containment Atmosphere Monitoring System	dome, SG comp., PRZ comp., PRZ valve comp.	x	x
	Passive Autocatalytic Recombiners	PAR	Containment (Equipment Rooms)		
Compartment separation mixing dampers	Mixing Damper actuation	Damper Actuator	between IRWST and lower annular rooms	x	x

Table 19.2-3—Severe Accident Instrumentation and Equipment
Sheet 2 of 4

Function	Measured quantity/ Activated device	Type of device	Location	12h Batt.	SBO
	Mixing Damper postion	Position sensor	between IRWST and lower annular rooms	x	x
Containment Monitoring & Heat Removal					
	Containment pressure	PDE	two locations in containment	x	x
	Containment pressure (dedicated)	PDE	two locations in containment	x	x
SAHRS operation	SAHRS Pump	Pump Motor	SAHRS compart.(SB)		x
	Inlet T	PT type	SAHRS compart.(SB)		x
	Outlet T	PT type	SAHRS compart.(SB)		x
	Volume flowrate	Flow rate meter	SAHRS compart.(SB)		x
	SAHRS pressure (Pump Inlet and Outlet)	PDE	SAHRS compart.(SB)		x
	SAHRS spraying line isolation valve	Valve	SAHRS compart.(SB)	x	x
	SAHRS spraying line isolation valve position	Position sensor	SAHRS compart.(SB)	x	x
	SAHRS suction line valve	Valve	SAHRS compart.(SB)	x	x
	SAHRS suction line valve position	Position sensor	SAHRS compart.(SB)	x	x
	SAHRS cooling line isolation valve	Valve	SAHRS compart.(SB)	x	x
	SAHRS cooling line isolation valve position	Position sensor	SAHRS compart.(SB)	x	x
	SAHRS sump level	Sump level gauge	SAHRS compart.(SB)	x	x
	SAHRS KLC ventilation flaps	ventilation flap	SAHRS compart.(SB)		
	SAHRS KLC ventilation flap position	Position sensor	SAHRS compart.(SB)	x	x
	IRWST water level	Sump level gauge	IRWST	x	x
	IRWST temperature	Thermocouple	IRWST	x	x

**Table 19.2-3—Severe Accident Instrumentation and Equipment
Sheet 3 of 4**

Function	Measured quantity/ Activated device	Type of device	Location	12h Batt.	SBO
SAHRS backflushing	Pressure drop over sump screen filter	Pressure difference	IRWST SAHRS sump		x
	SAHRS flushing line isolation valve	Valve	SAHRS compart.(SB)	x	x
	SAHRS flushing line isolation valve position	Position sensor	SAHRS compart.(SB)	x	x
	SIS valve position for single train backflushing	Valve	JNG compart.(SB)		x
	SIS valve position for single train backflushing	Position sensor	JNG compart.(SB)		x
CCWS operation	CCWS Pump	Pump Motor	CCWS compart.(SB)		
	CCWS Pressurizing Pump	Pump Motor	CCWS compart.(SB)		x
	CCWS Demineralized Water Supply valve	Valve	CCWS compart.(SB)		x
	CCWS Sump level	Sump level gauge	CCWS compart.(SB)	x	x
	Inlet T (downstream of CCWS Pump)	PT type	CCWS compart.(SB)		x
	Outlet T (downstream of SAHRS Heat Exchanger)	PT type	CCWS compart.(SB)		x
	Volume flow rate (downstream CCWS Pump)	Flow rate meter	CCWS compart.(SB)		x
	T upstream Dedicated CCWS Heat Exchangers	PT type	CCWS compart.(SB)		x
	T upstream CCWS Pump	PT type	CCWS compart.(SB)		x
	Volume flow rate (downstream SAHRS Heat Exchanger)	Flow rate meter	CCWS compart.(SB)		x
	Water level - Surge Tank	Tank level gauge	CCWS compart.(SB)		x
	Pressure - Surge Tank	PDE	CCWS compart.(SB)		x
	Overpressure Protection – Surge Tank	Position sensor	CCWS compart.(SB)		x
	Overpressure Protection – Surge Tank	Position sensor	CCWS compart.(SB)		x

Table 19.2-3—Severe Accident Instrumentation and Equipment
Sheet 4 of 4

Function	Measured quantity/ Activated device	Type of device	Location	12h Batt.	SBO
	Pump Safety valve	Position sensor	CCWS compart.(SB)		x
ESWS					
	ESWS Pump	Pump Motor	ESWS		x
	Pressure drop over sump screen filter	PDE	ESWS		x
	Volume flow rate	Flow rate meter	ESWS		x
	Electrical and I&C Processing Units	Black Box	ESWS		x
Activity Release					
	Position of MSRIV	Position sensor	LAB MSRIV compart.	x	x
	Dose rate in containment	Gamma sensitive detect.	inside containment and outside equipment hatch		
	Dose rate downstream of filters	Gamma sensitive detect.	stack		
	Volume flowrate stack	Flow rate meter	stack		
	Dose rate in safeguard building	Gamma sensitive detect.	downstream of ventilation filter		
	Volume flowrate safeguard building ventilation	Flow rate meter	downstream of ventilation filter		
	Dose rate in the annulus	Gamma sensitive detect.	downstream of ventilation filter		
	Volume flowrate annulus ventilation	Flow rate meter	downstream of ventilation filter		
Annulus Ventilation Monitoring					
	Subatmospheric pressure	PDE	annulus		

Table 19.2-4—SAMG Technical Basis – Mapping Challenge Mechanism to Operator Action

Challenge Mechanism	Applicable Safety Function	Plant Parameters / Instrumentation Needed for Diagnostic	Applicable Immediate or Systematic Actions
Interfacing LOCA Containment Isolation Failure	Releases	Site dose Auxiliary / safeguards buildings radiation Containment isolation valve position	Containment isolation (immediate action)
Hydrogen deflagration in phase 2 (SG available) Hydrogen deflagration in phase 2 (no SG available) Ex-Vessel Hydrogen Deflagration	Containment	Containment pressure Containment hydrogen concentration	N/A
Induced SGTR	Releases	Site dose Steam system radiation Primary system pressure	Primary depressurization (systematic action, in EOPs prior to entry)
Basemat Penetration	Heat removal	Containment heat removal system temperatures and flows Core catcher thermocouples	N/A
SA Following Initiated SGTR (Containment Bypass)	Releases	Site dose Steam system radiation	N/A
Quenching	Containment	Containment pressure	N/A
DCH RPV Rocket	Releases	Site dose Annulus radiation Primary system pressure	Primary depressurization (systematic action, in EOPs prior to entry)

Table 19.2-5—SAMDA Candidates – Already Implemented
Sheet 1 of 3

SAMDA ID	Potential Enhancement
AC/DC-01	Provide additional DC battery capacity.
AC/DC-03	Add additional battery charger or portable, diesel-driven battery charger to existing DC system
AC/DC-04	Improve DC bus load shedding.
AC/DC-05	Provide DC bus cross-ties.
AC/DC-06	Provide additional DC power to the 120/240V vital AC system.
AC/DC-07	Add an automatic feature to transfer the 120V vital AC bus from normal to standby power.
AC/DC-09	Provide an additional diesel generator.
AC/DC-11	Improve 4.16kV bus cross-tie ability.
AC/DC-14	Install a gas turbine generator.
AC/DC-16	Improve uninterruptible power supplies.
AC/DC-24	Bury off-site power lines.
AT-01	Add an independent boron injection system.
AT-02	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.
AT-07	Install motor generator set trip breakers in control room.
AT-08	Provide capability to remove power from the bus powering the control rods.
CB-01	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.
CB-04	Install self-actuating containment isolation valves.
CB-10	Replace steam generators with a new design.
CB-12	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture.
CB-14	Provide improved instrumentation to detect steam generator tube ruptures, such as Nitrogen-16 monitors.
CB-16	Install a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources.
CB-20	Install relief valves in the component cooling water system
CC-01	Install an independent active or passive high pressure injection system.
CC-04	Add a diverse low pressure injection system.
CC-05	Provide capability for alternate injection via diesel-driven fire pump.
CC-06	Improve ECCS suction strainers.
CC-07	Add the ability to manually align emergency core cooling system recirculation.

Table 19.2-5—SAMDA Candidates – Already Implemented
Sheet 2 of 3

SAMDA ID	Potential Enhancement
CC-10	Provide an in-containment reactor water storage tank.
CC-15	Replace two of the four electric safety injection pumps with diesel-powered pumps.
CC-17	Create a reactor coolant depressurization system.
CC-21	Modify the containment sump strainers to prevent plugging
CP-01	Create a reactor cavity flooding system.
CP-03	Use the fire water system as a backup source for the containment spray system.
CP-07	Provide post-accident containment inerting capability.
CP-08	Create a large concrete crucible with heat removal potential to contain molten core debris.
CP-11	Increase depth of the concrete base mat or use an alternate concrete material to ensure melt-through does not occur.
CP-13	Construct a building to be connected to primary/secondary containment and maintained at a vacuum.
CP-17	Install automatic containment spray pump header throttle valves.
CP-20	Install a passive hydrogen control system.
CP-21	Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure.
CP-22	Install a secondary containment filtered ventilation
CW-01	Add redundant DC control power for SW pumps.
CW-02	Replace ECCS pump motors with air-cooled motors.
CW-04	Add a service water pump.
CW-05	Enhance the screen wash system.
CW-06	Cap downstream piping of normally closed component cooling water drain and vent valves.
CW-10	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.
CW-15	Use existing hydro test pump for reactor coolant pump seal injection.
CW-16	Install improved reactor coolant pump seals.
CW-17	Install an additional component cooling water pump.
EPR-01	Provide an additional safety chilled water system train.
EPR-05	Add redundant pressure sensors to the pressurizer and steam generator.
FR-03	Install additional transfer and isolation switches.
FR-05	Enhance control of combustibles and ignition.

Table 19.2-5—SAMDA Candidates – Already Implemented
Sheet 3 of 3

SAMDA ID	Potential Enhancement
FW-01	Install a digital feed water upgrade.
FW-02	Create ability for emergency connection of existing or new water sources to feedwater and condensate systems.
FW-04	Add a motor-driven feedwater pump.
FW-07	Install a new condensate storage tank (auxiliary feedwater storage tank).
FW-11	Use fire water system as a backup for steam generator inventory.
FW-15	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.
HV-01	Provide a redundant train or means of ventilation to the switchgear rooms.
HV-02	Add a diesel building high temperature alarm or redundant louver and thermostat.
HV-04	Add a switchgear room high temperature alarm.
HV-05	Create ability to switch emergency feedwater room fan power supply to station batteries in a station blackout.
SR-01	Increase seismic ruggedness of plant components.
SR-02	Provide additional restraints for CO2 tanks.
OT-01	Install digital large break LOCA protection system.

Figure 19.2-1—Core Melt Stabilization System

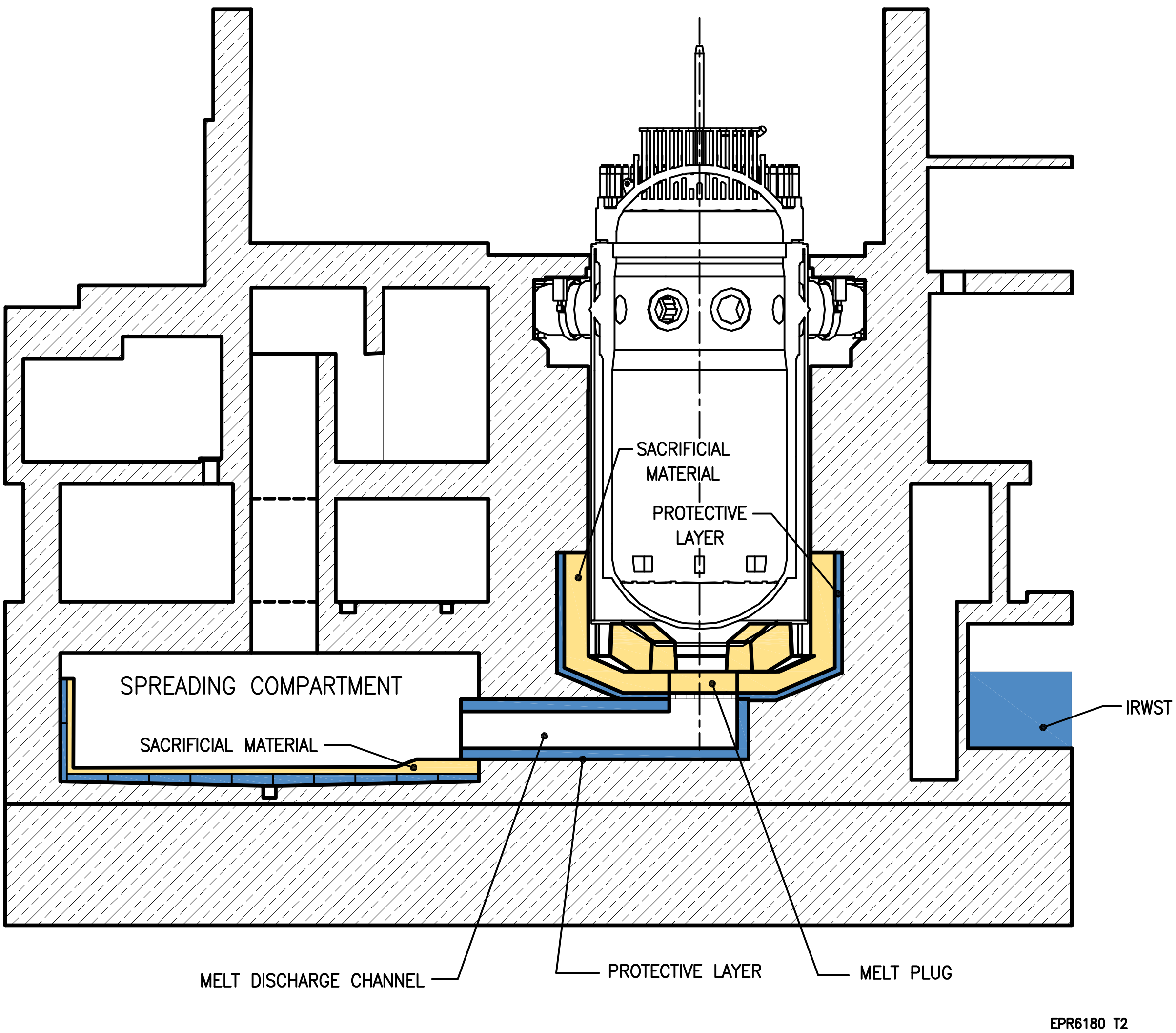

Figure 19.2-2—Severe Accident Heat Removal System

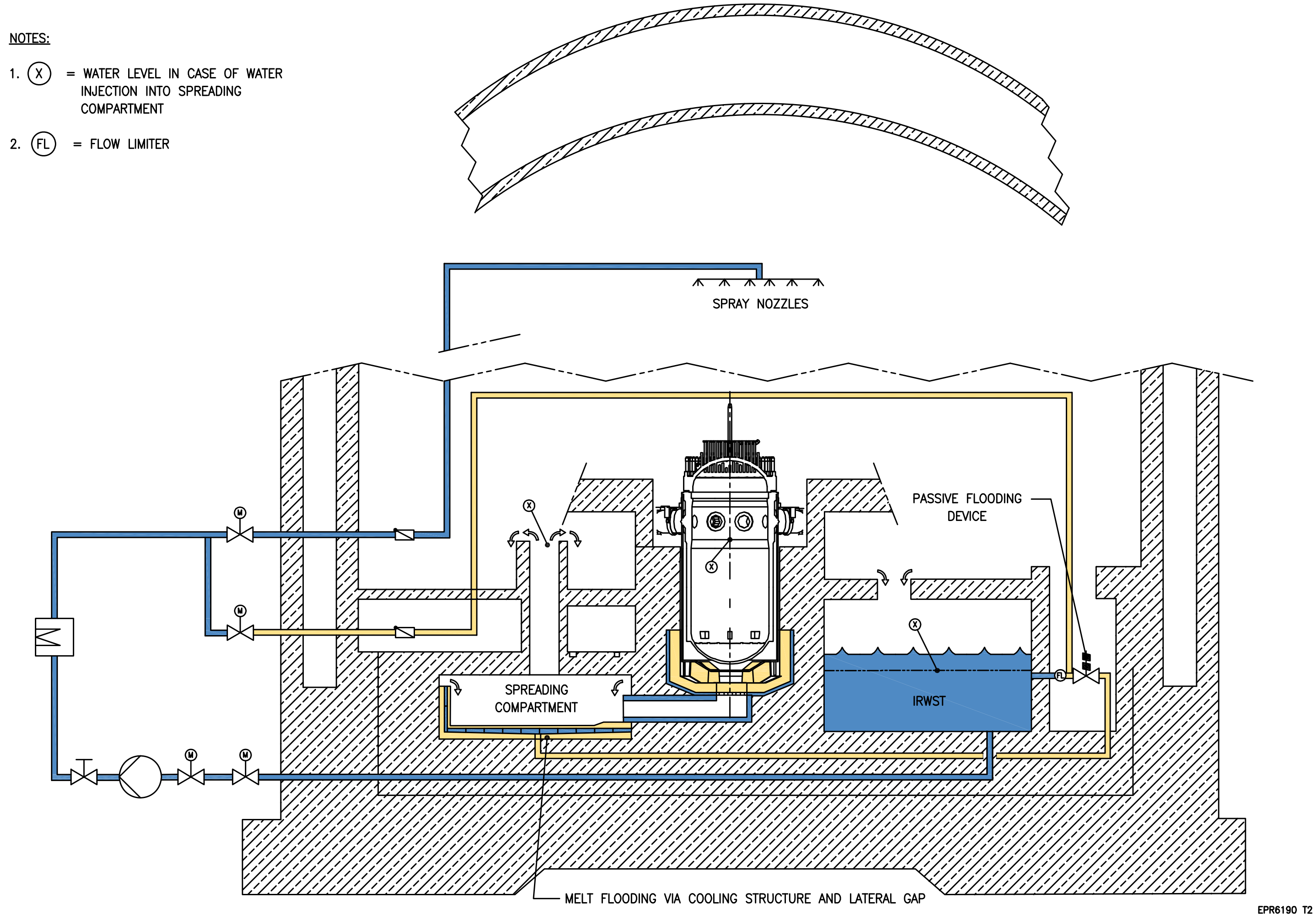

Figure 19.2-3—Molten Debris Spreading Area

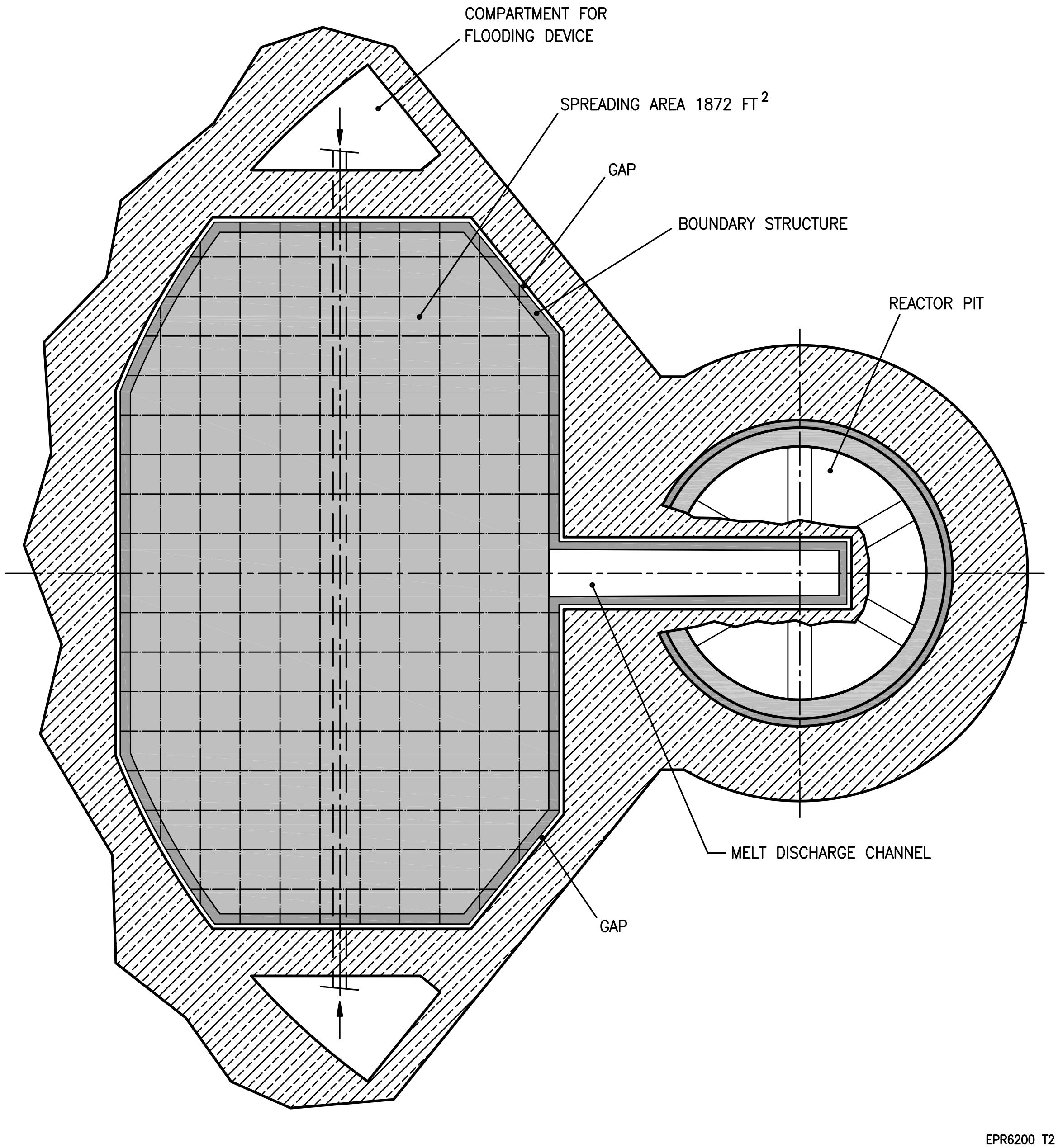

Figure 19.2-4—Tolerance Limit Plot of Hydrogen Production

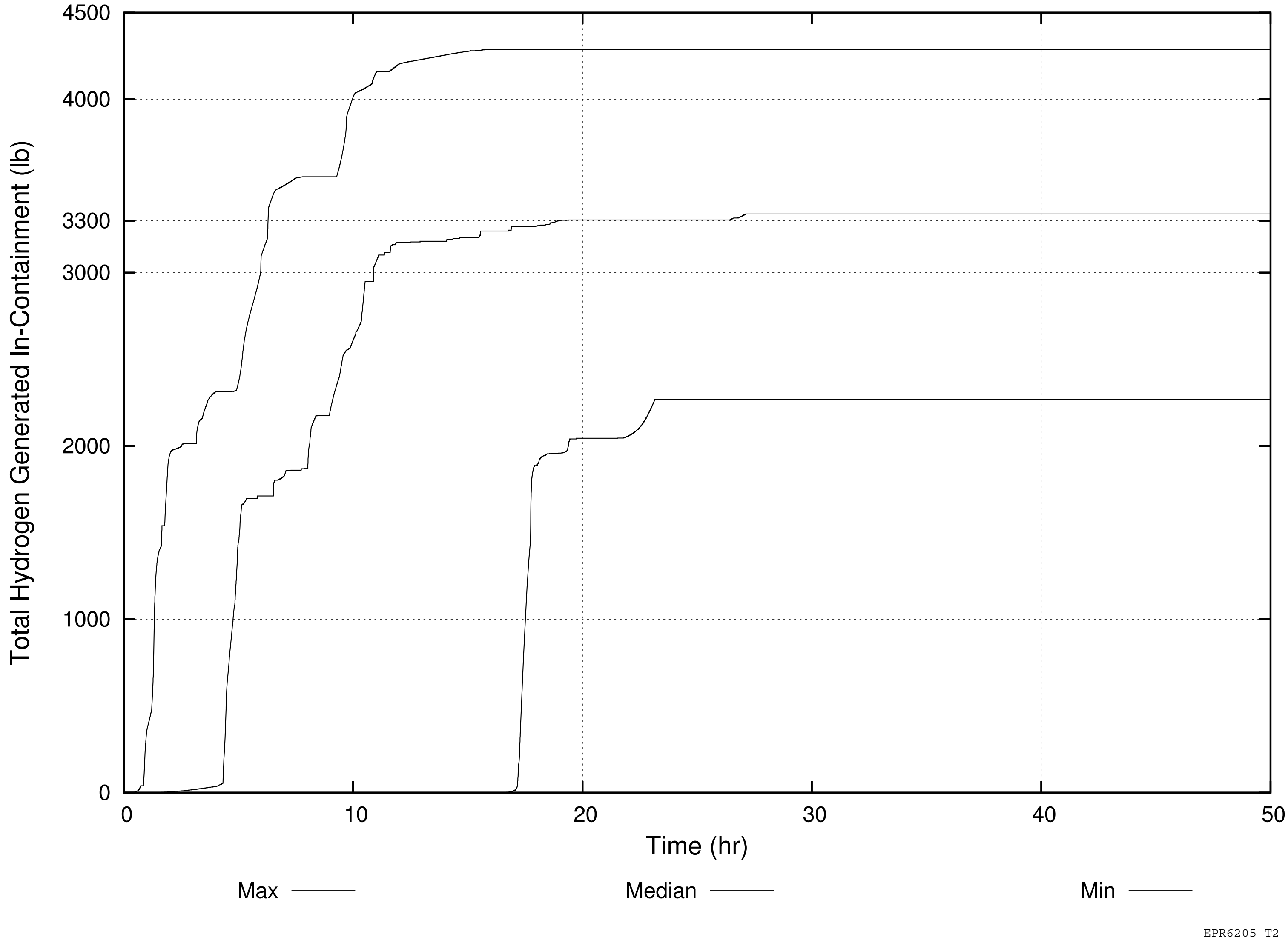

Figure 19.2-5—Hydrogen Concentrations through the U.S. EPR Containment

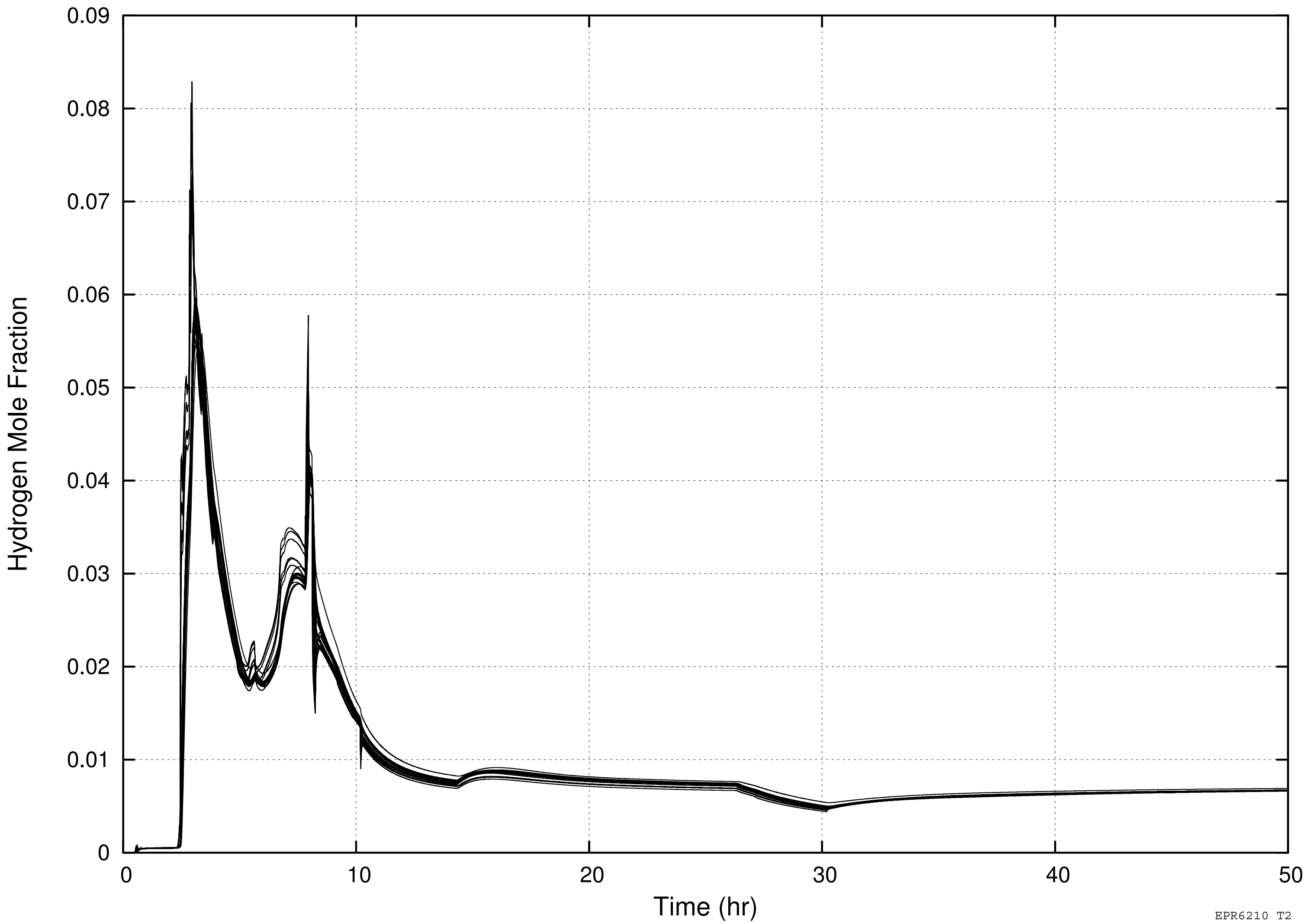

Figure 19.2-6—Tolerance Limit Plot of Hydrogen Concentration

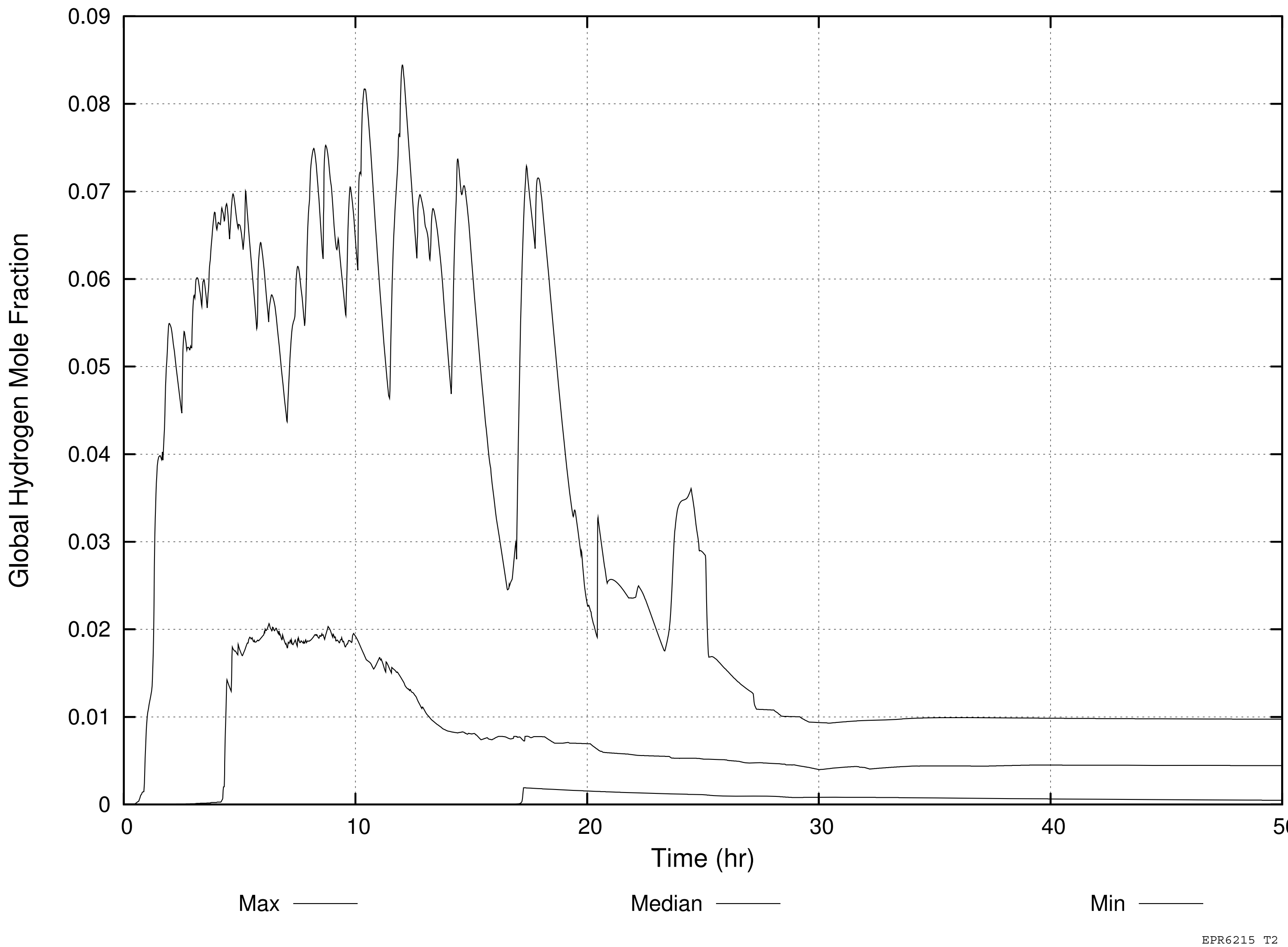

Figure 19.2-7—Tolerance Limit Plot of Containment AICC Pressure

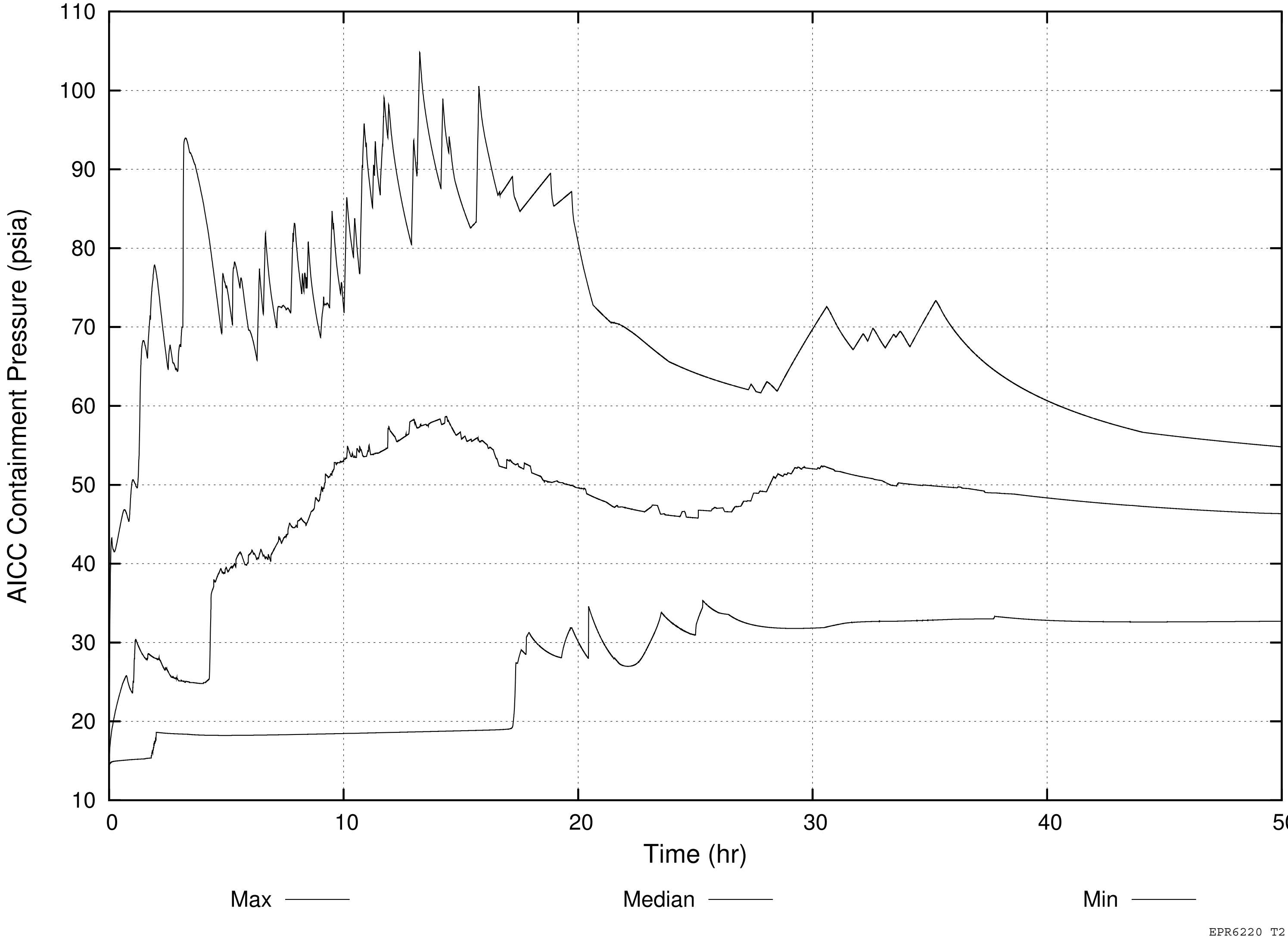

Figure 19.2-8—Tolerance Limit Plot of Sigma Index for the Pump/SG Compartment

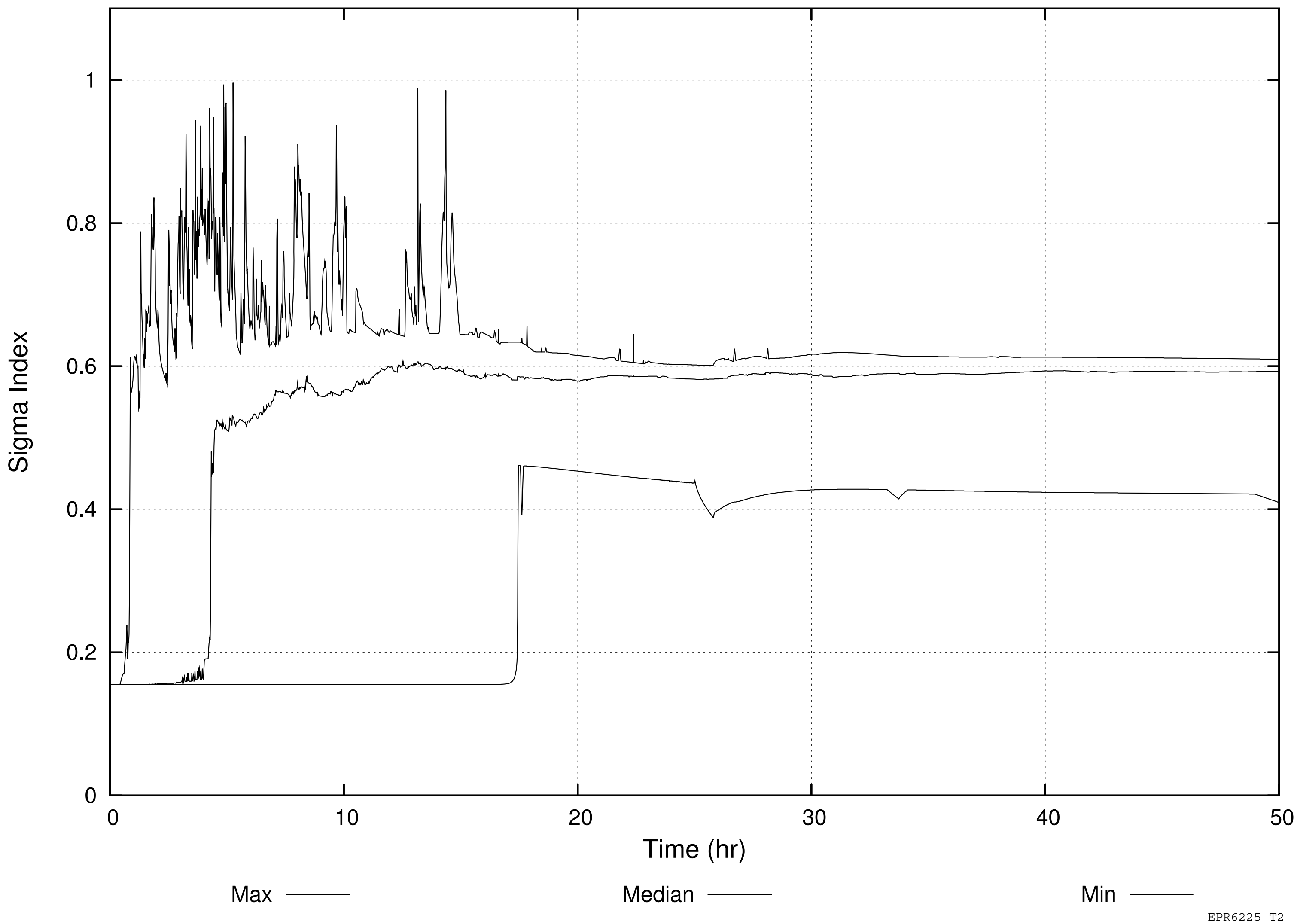

Figure 19.2-9—Hydrogen Concentration at Significant Events

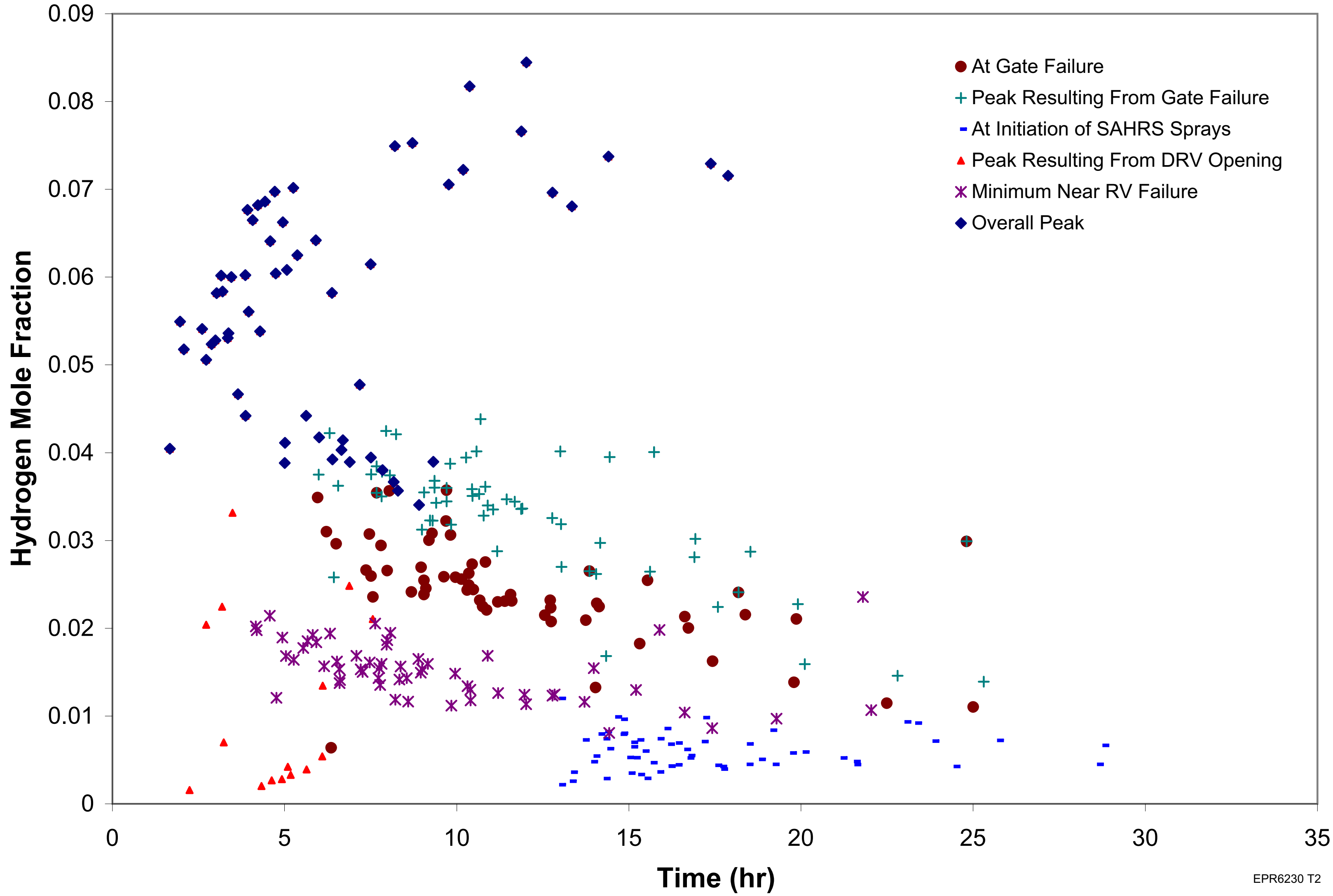

Figure 19.2-10—Time of RPV Rupture versus Duration of Retention Period

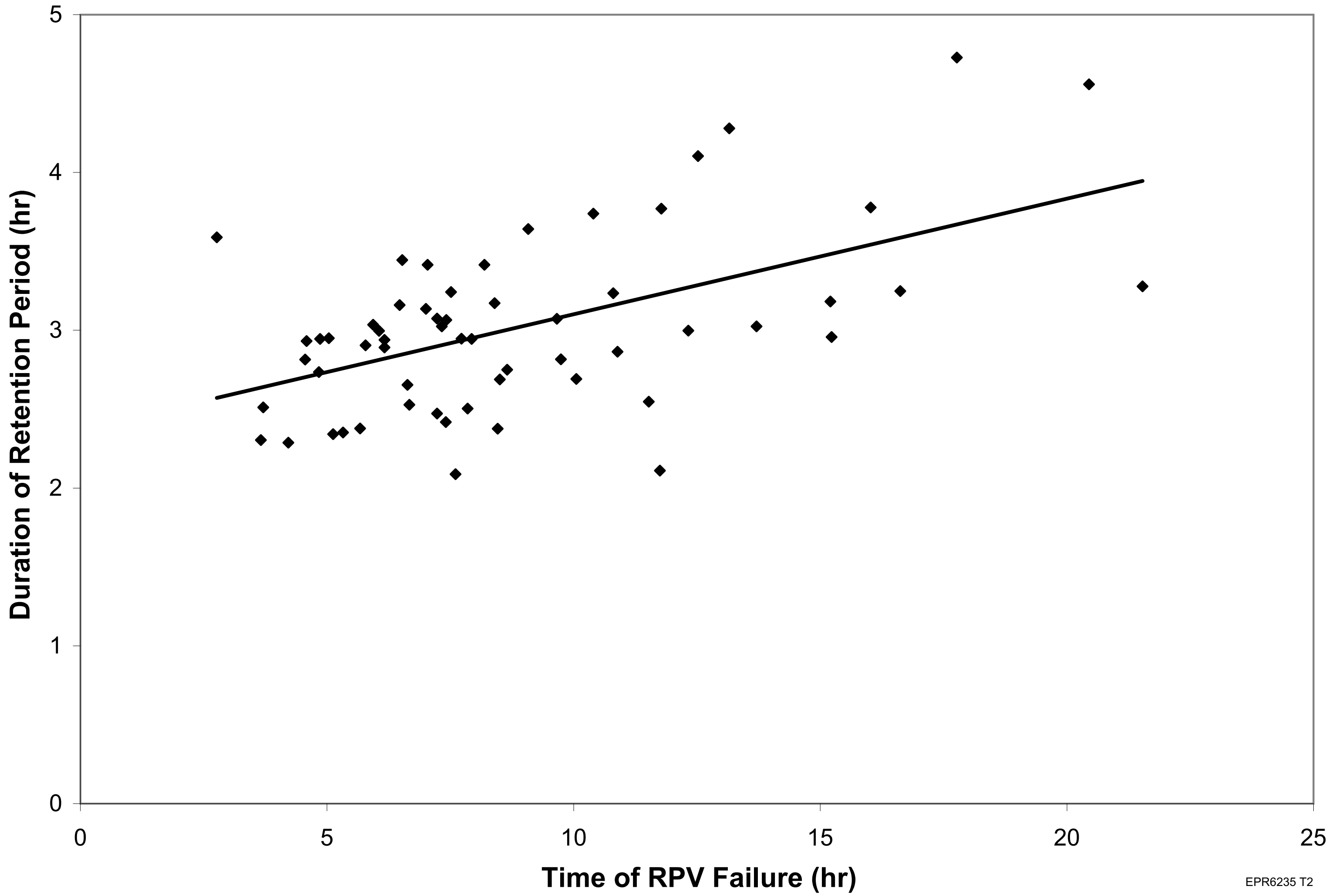

Figure 19.2-11—Temperature Profiles within the Gate after Contact with an Oxidic Melt

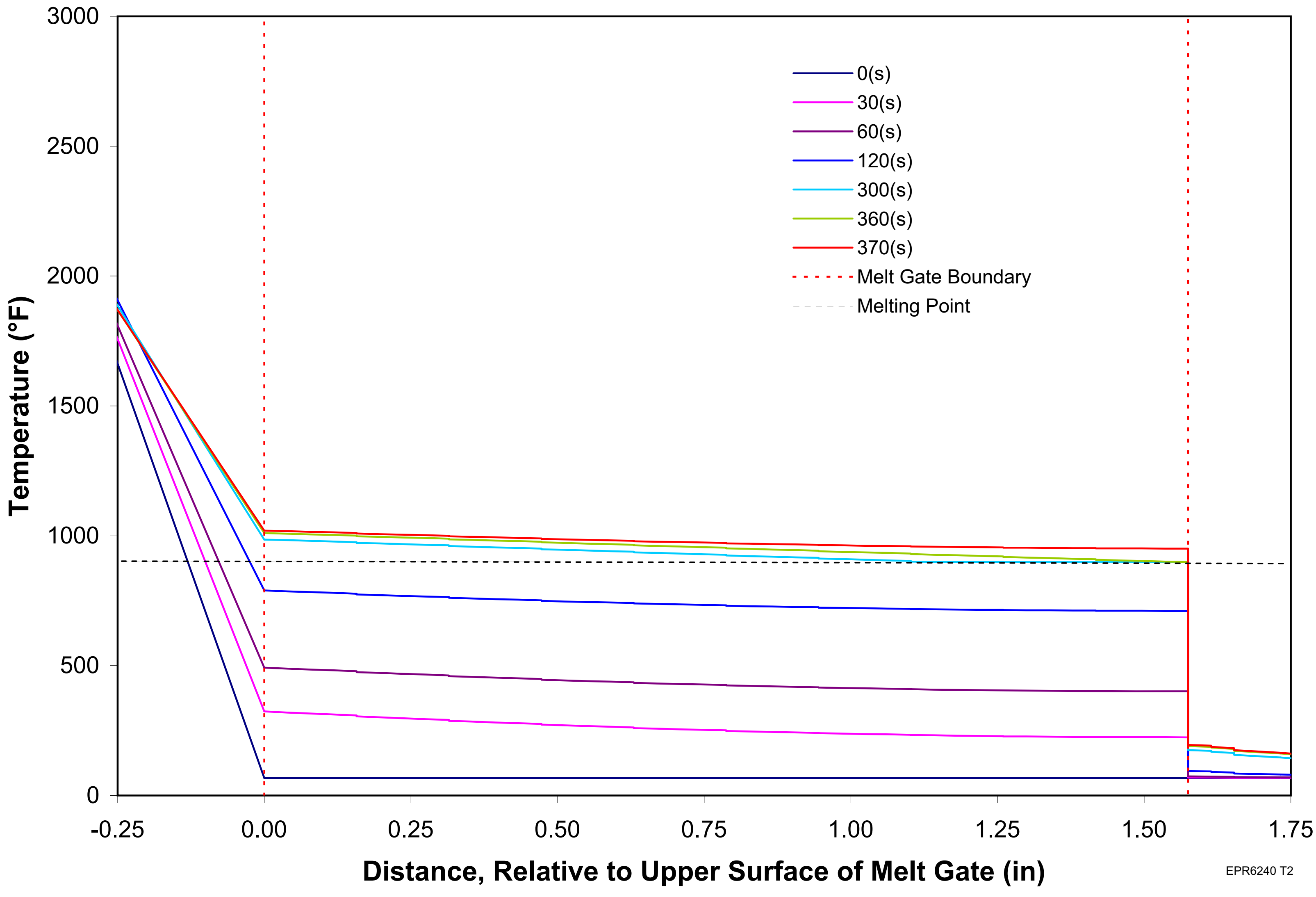

Figure 19.2-12—Temperature Profiles within the Gate after Contact with a Metallic Melt

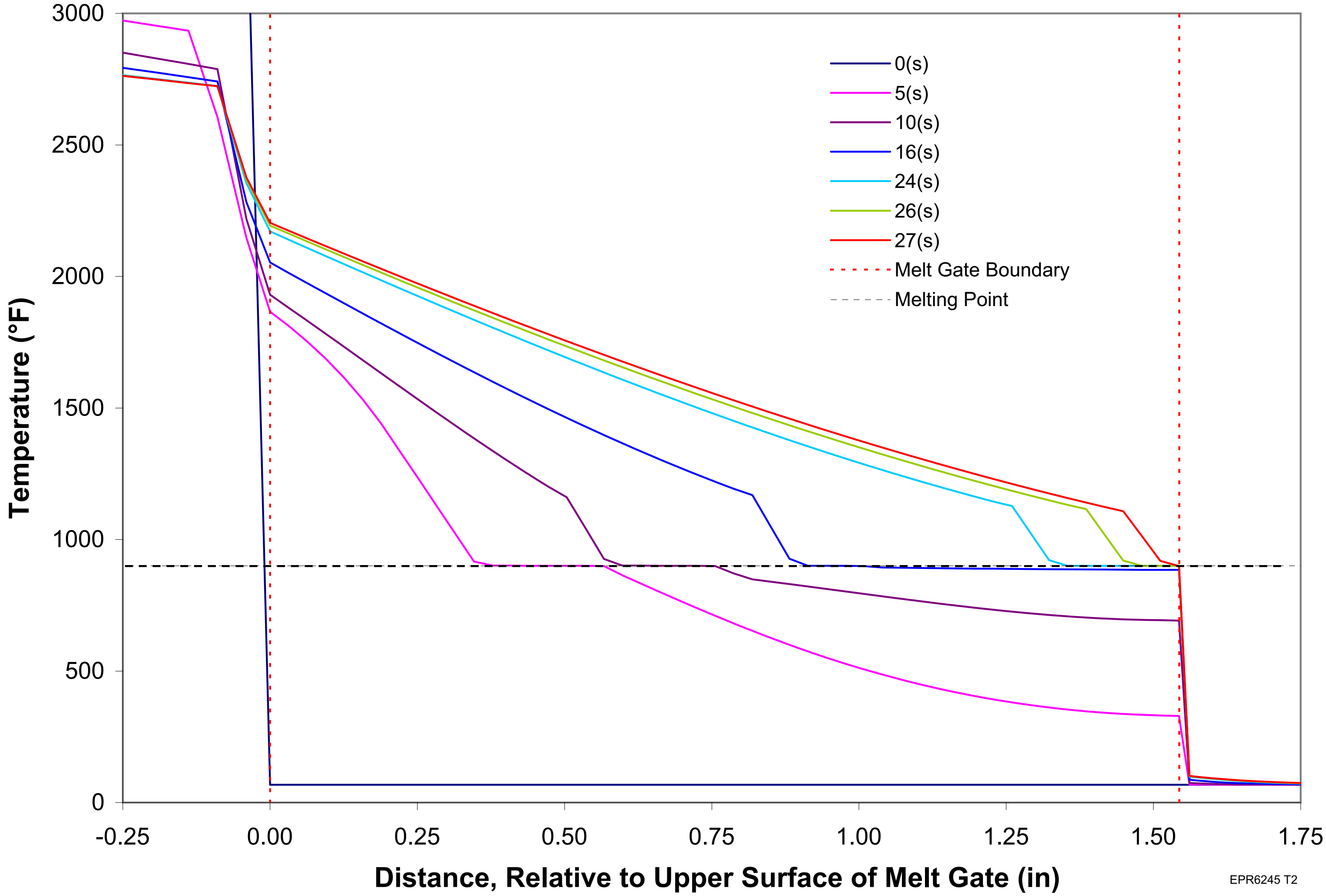

Figure 19.2-13—Corium Height as a Function of Distance from Melt Channel, 100% Gate Failure

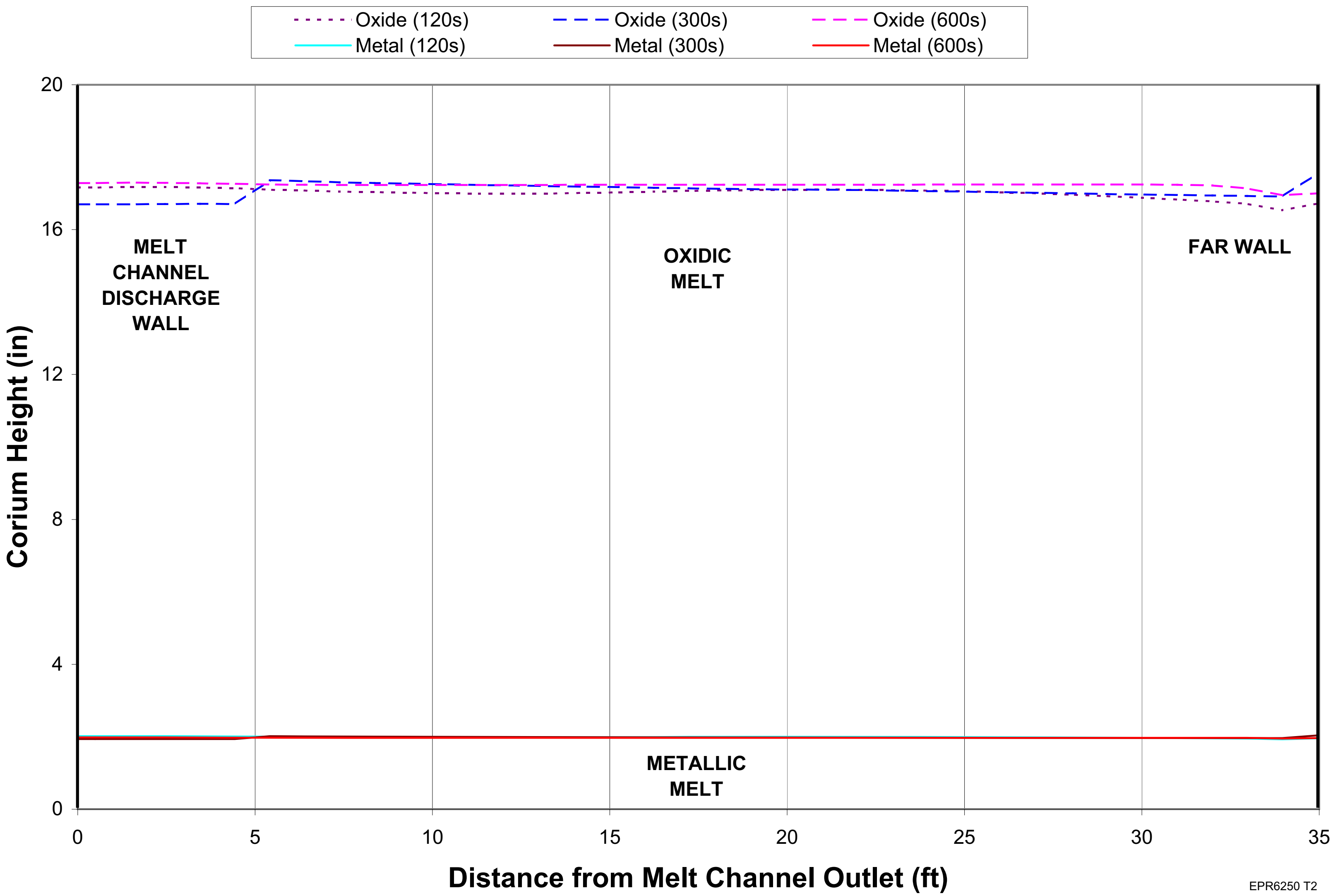

Figure 19.2-14—Corium Height as a Function of Distance from Melt Channel, 10% Gate Failure

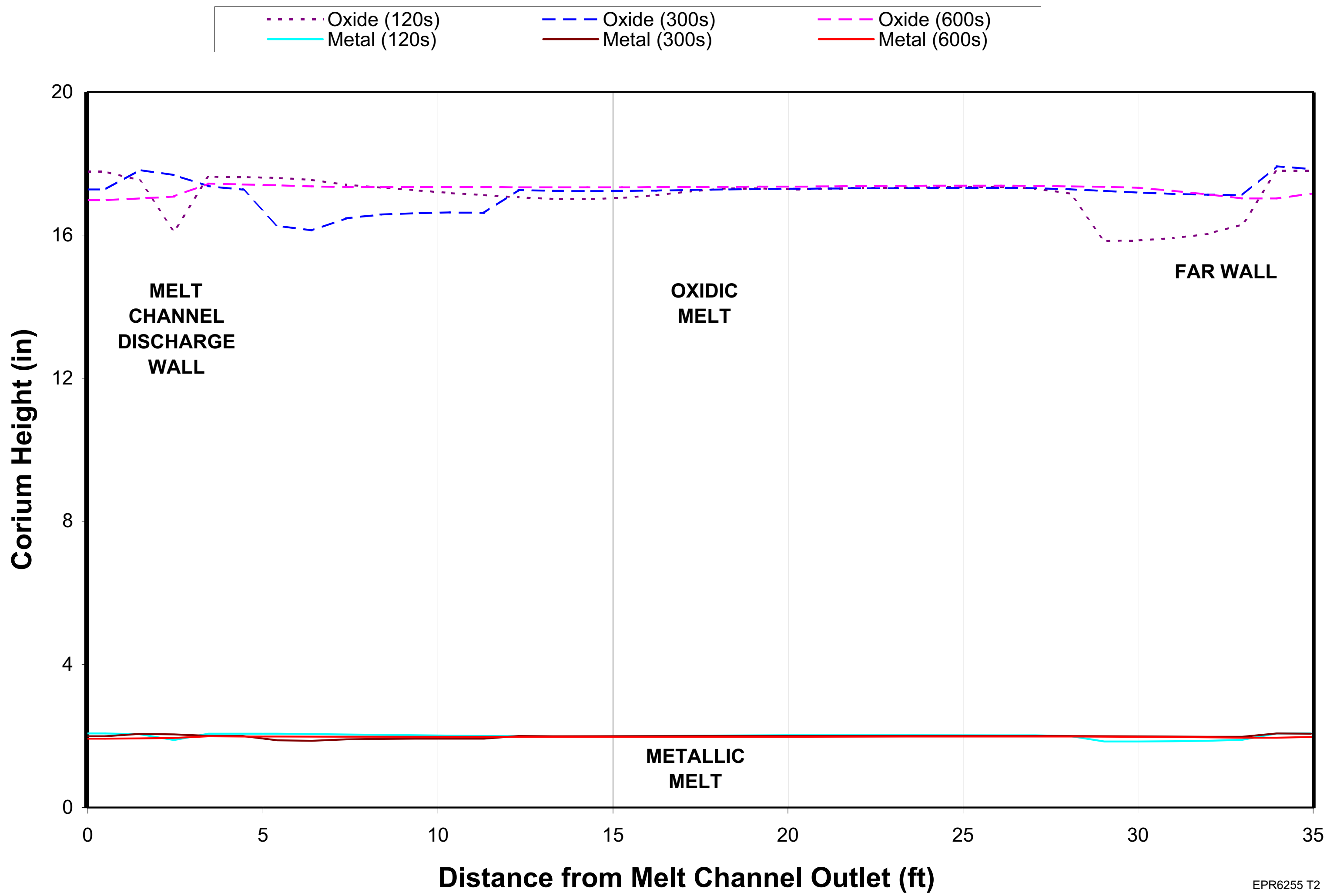

Figure 19.2-15—Containment Pressure following Gate Failure

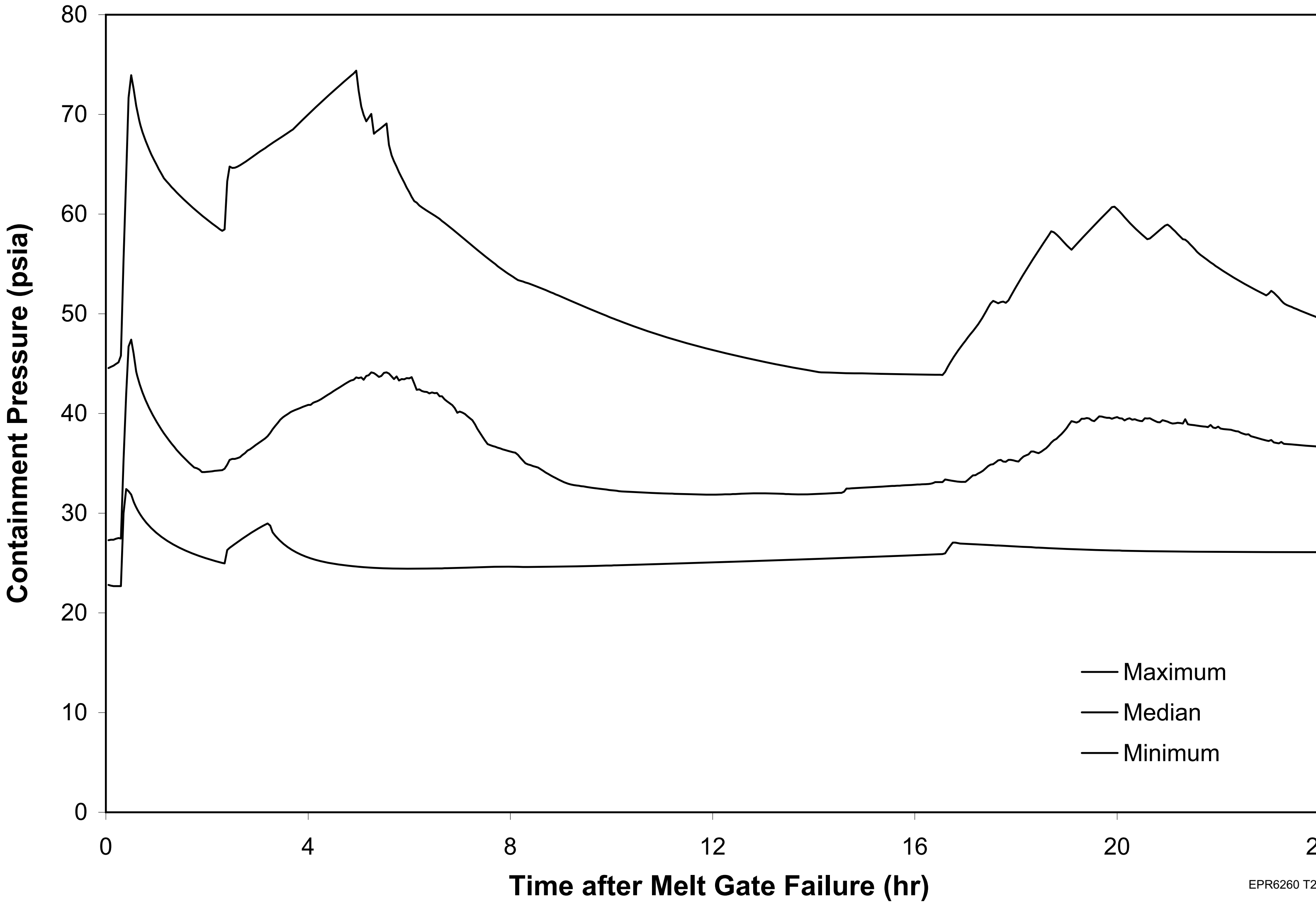

Figure 19.2-16—Variation of Power Into Cooling Channels (kW/ft²)

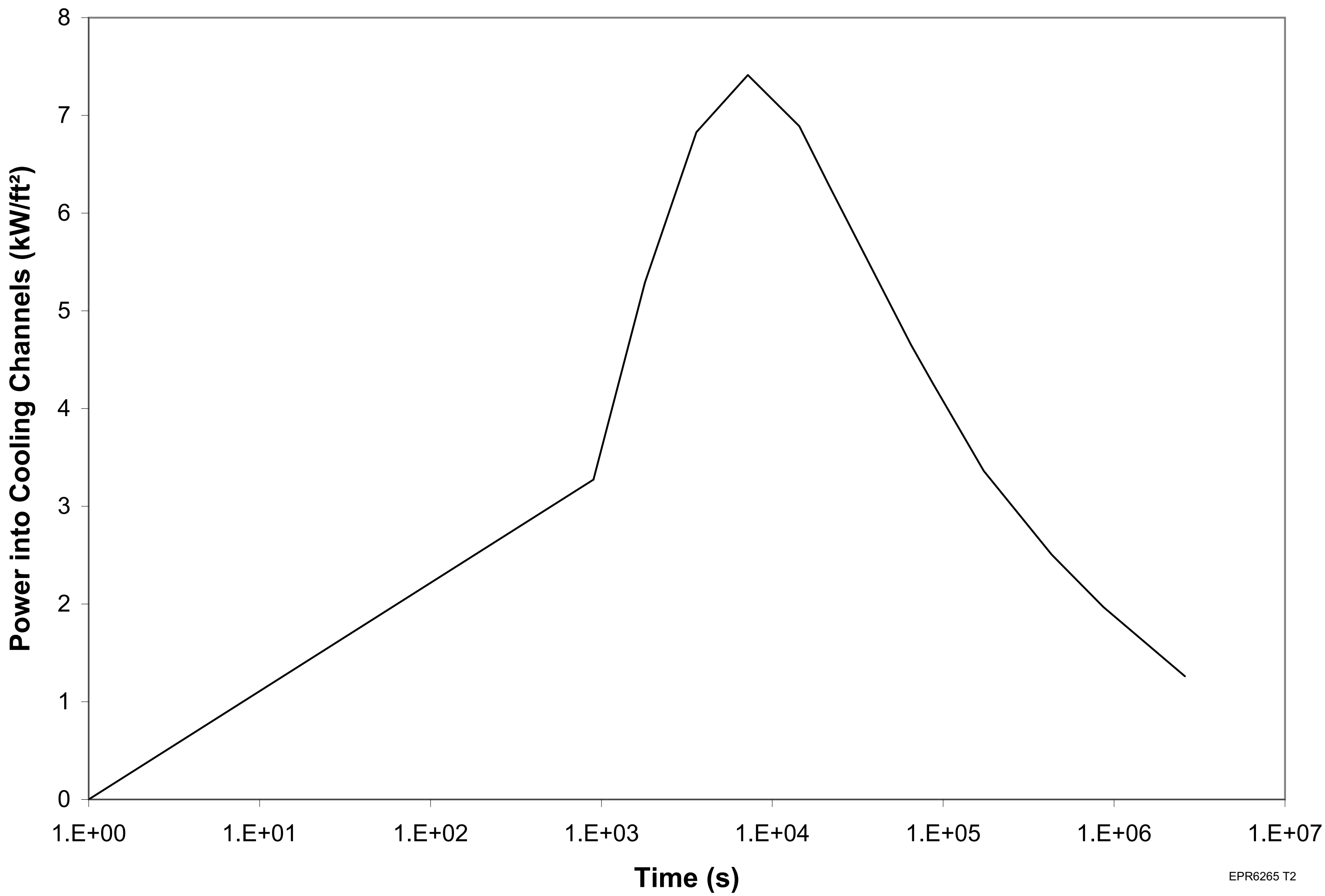

Figure 19.2-17—Temperature Profiles within the Cooling Plate (<6 hours)

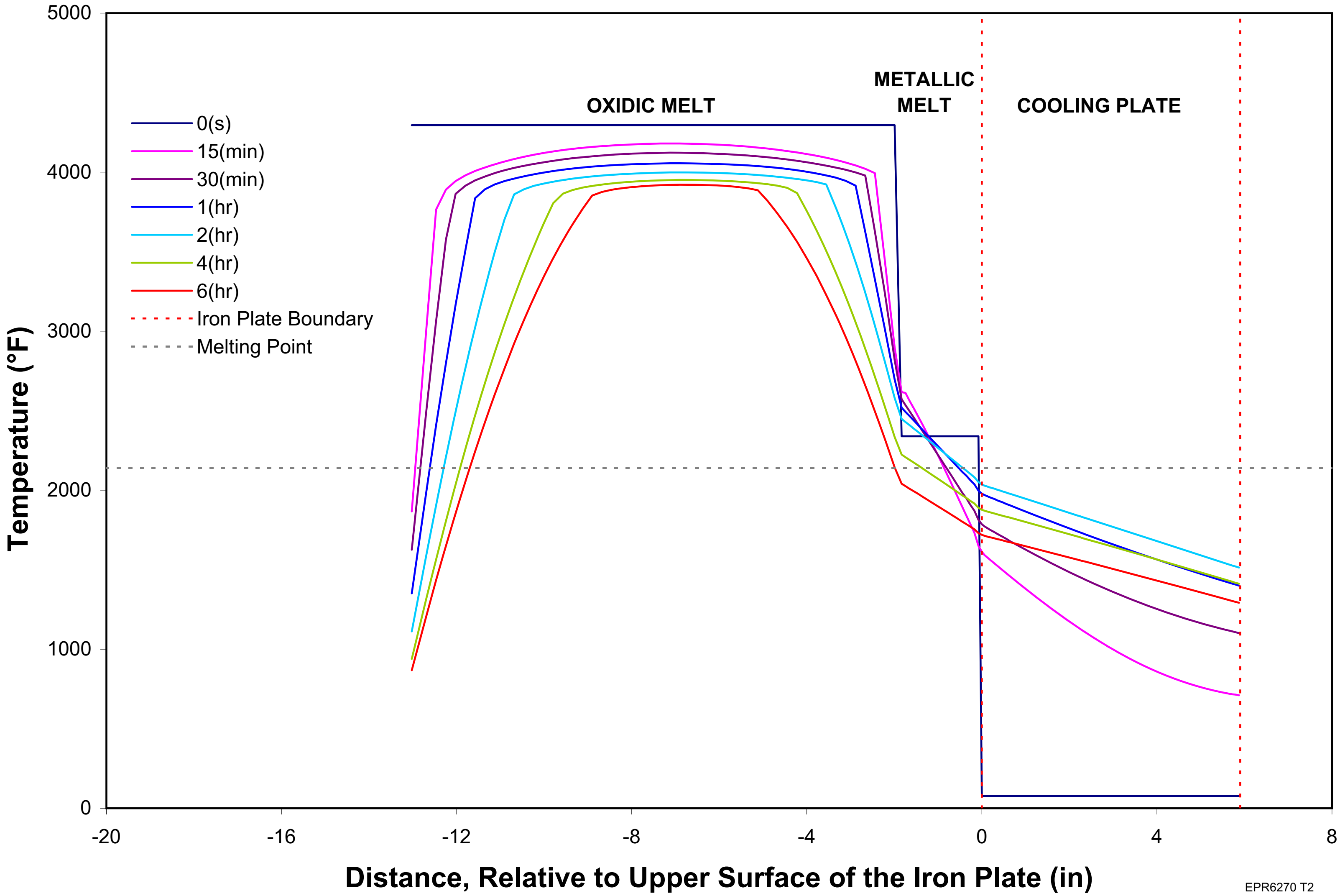

Figure 19.2-18—Temperature Profiles within the Cooling Plate (>6 hours, <30 days)

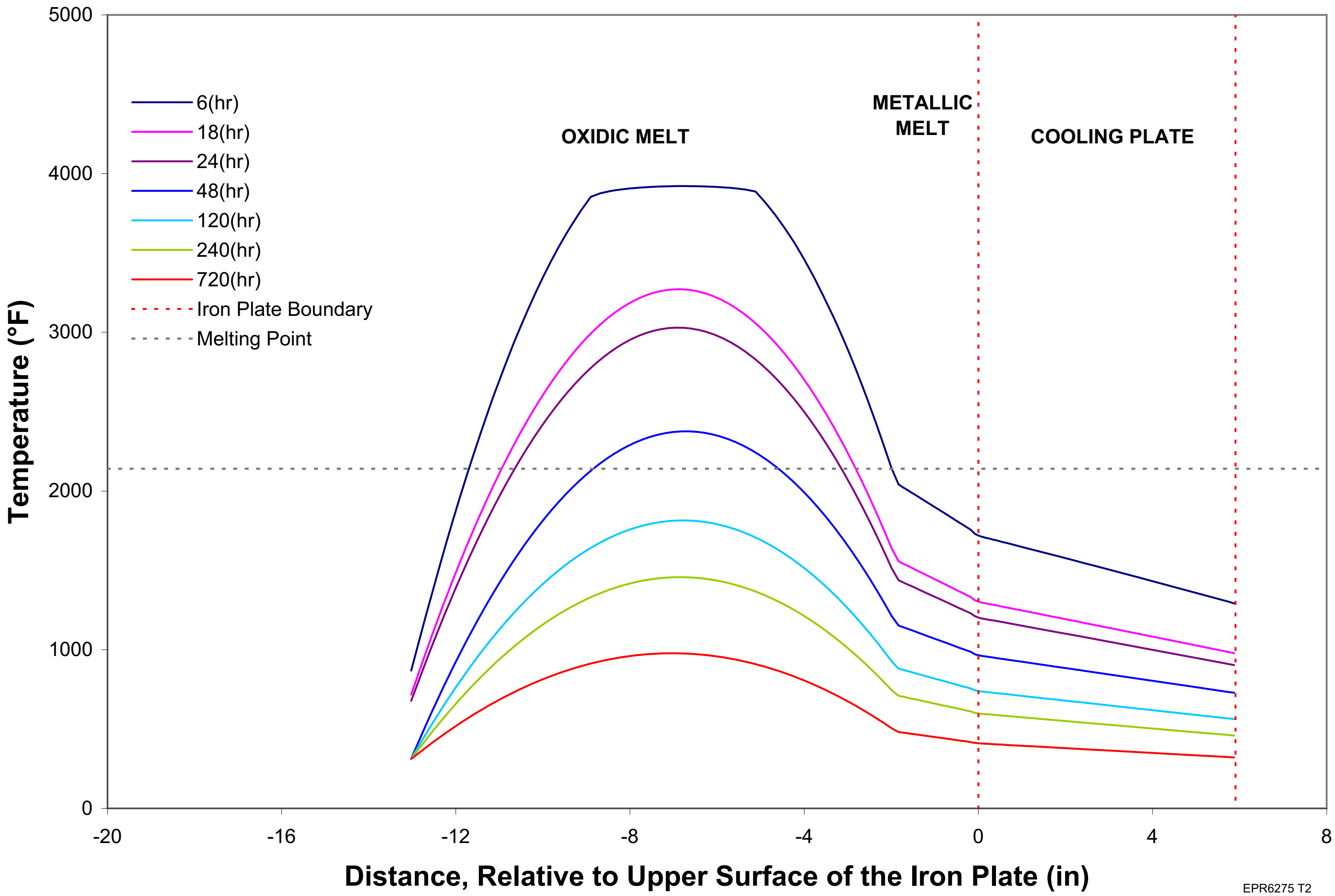

Figure 19.2-19—RCS Pressure at RPV Failure

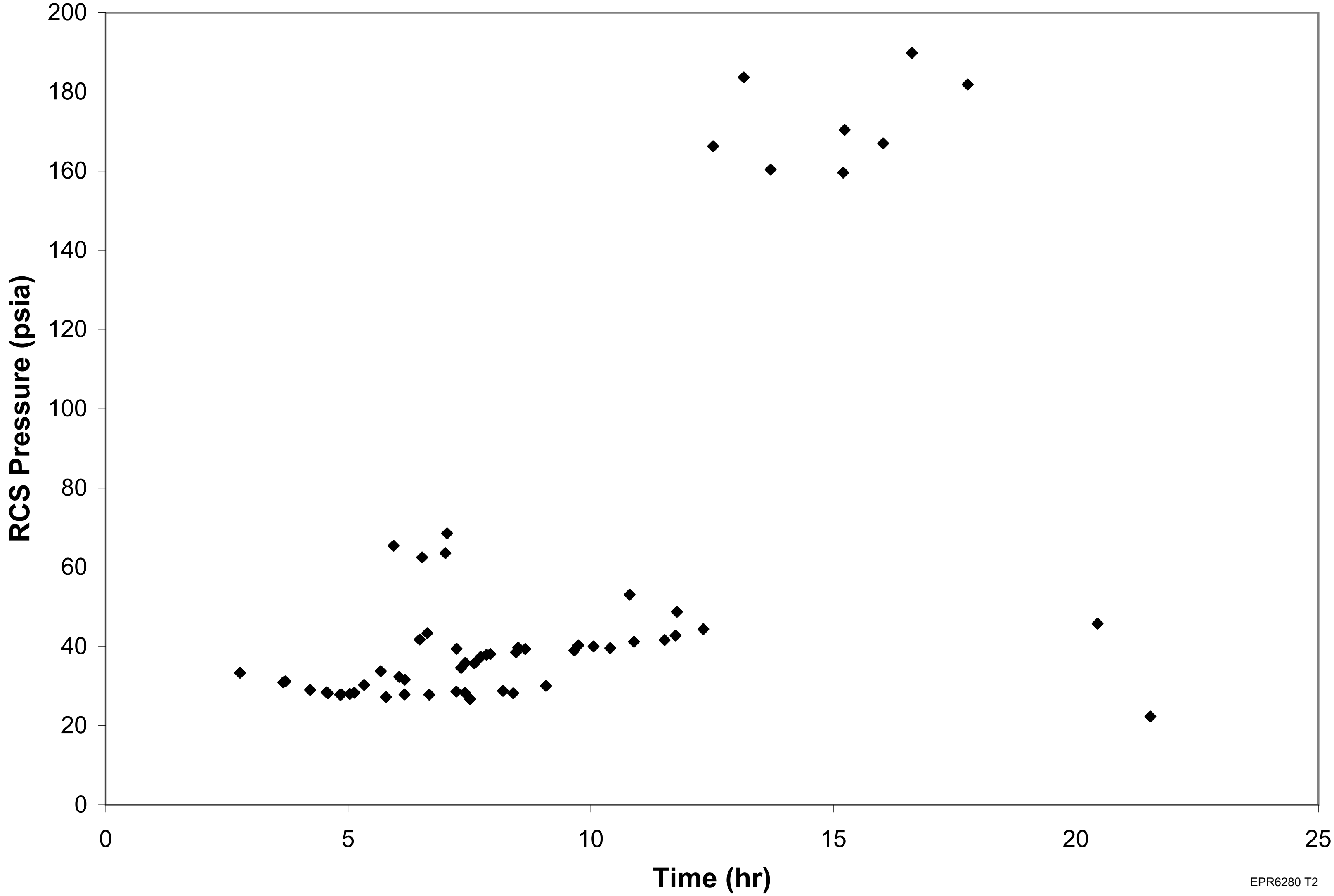

Figure 19.2-20—Bounding In-Core Temperature and Pressure Development during a Severe Accident

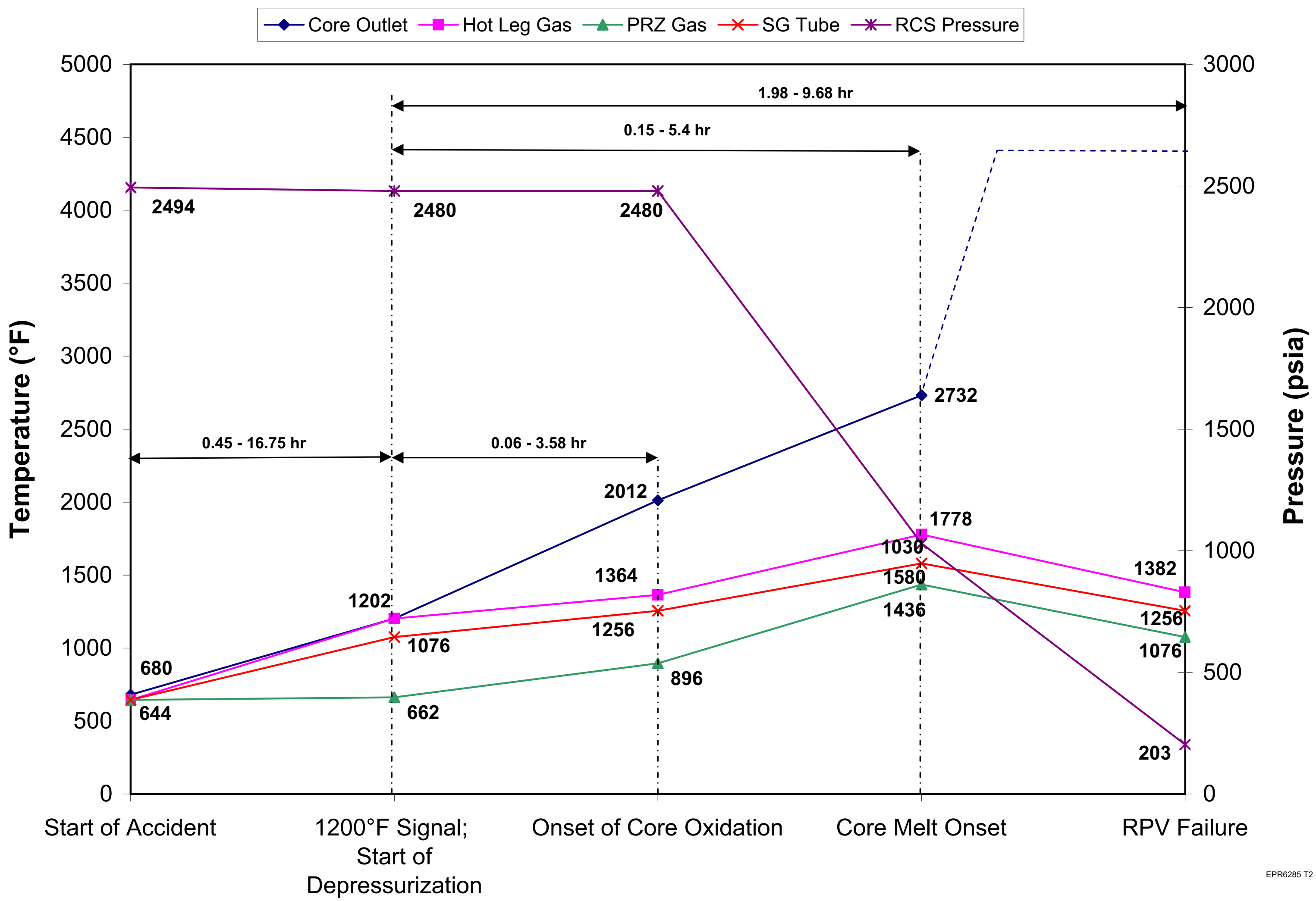

Figure 19.2-21—Course of Primary Events during a Severe Accident