November 29, 2007

Mr. Ronnie L. Gardner, Manager
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SUBJECT: FINAL SAFETY EVALUATION REPORT FOR ANP-10268(P), REVISION 0, "U.S. EVOLUTIONARY PRESSURIZED WATER REACTOR (EPR) SEVERE ACCIDENT EVALUATION TOPICAL REPORT" (TAC NO. MD3830)

Dear Mr. Gardner:

By letter dated October 31, 2006, (NRC's ADAMS Accession Number ML063100154), as supplemented by letters dated July 13, 2007 (ML071990057), and August 29, 2007 (ML072490436), AREVA NP (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review proprietary and non-proprietary versions of Topical Report (TR) ANP-10268, Revision 0, "U.S. EPR Severe Accident Evaluation Topical Report." By letter dated September 21, 2007, a draft safety evaluation (SE) regarding our approval of ANP-10268(P) was provided for your review and comments (ML072681229). The staff's disposition of AREVA's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The staff has found that ANP-10268(P), Revision 0 is acceptable for referencing in licensing applications for U.S. EPR to the extent specified and under the limitations delineated in the TR and in the enclosed SE. The SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in regulatory applications, our review will ensure that the material presented applies to the specific application involved. Regulatory applications that deviate from this TR will be subject to further review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that AREVA publish the accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE after the title page. Also, the accepted version must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, AREVA will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

If you have any questions, please contact me at gxt2@nrc.gov or (301) 415-3361.

Sincerely,

/RA/

Getachew Tesfaye, Sr. Project Manager
EPR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Project No. 733

Enclosure: Safety Evaluation

cc w/encl: U.S. EPR Service List

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FINAL SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

ANP-10268(P), REVISION 0, "U.S. EVOLUTIONARY PRESSURIZED WATER REACTOR (EPR) SEVERE ACCIDENT EVALUATION TOPICAL REPORT" (TAC NO. MD3830)

PROJECT NO. 733

1.0 INTRODUCTION

By letter dated October 31, 2006 (ML063100154), as supplemented by letters dated July 13, 2007 (ML071990057), and August 29, 2007 (ML072490436), AREVA NP (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review proprietary and non-proprietary versions of Topical Report (TR) ANP-10268, Revision 0, "U.S. EPR Severe Accident Evaluation Topical Report." The proprietary version of the topical report is used as Reference 1 in this safety evaluation.

AREVA requested that the NRC issue a safety evaluation report which concludes that the technical bases for severe accident assessment, including the testing programs and the models and methods used for severe accident analyses, are adequate to meet the intent of the policies established in SECY 93-087 (Reference 2) and to support the U.S. EPR probabilistic risk assessment (PRA). AREVA plans to reference the approved version of the topical report in its design certification (DC) application and in the PRA for the U.S. EPR.

ANP-10268P includes a design description of U.S. EPR Severe Accident Management Systems, explains AREVA's safety issue resolution methodology, describes the various severe accident issues relevant to the U.S. EPR, and outlines the research and development related to these issues. The analysis methods planned for use in the DC application are presented, and several example scenarios are analyzed to illustrate how the methods will be used to address severe accident issues and support the U.S. EPR Probabilistic Risk Assessment.

2.0 REGULATORY EVALUATION

There are no specific regulatory requirements for the review of topical report submittals. The staff review was based on an evaluation of the technical merit of the material provided and compliance with any applicable regulations associated with the material presented. In this case the applicable regulation is Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(23), which requires the inclusion of "… a description and analysis of design features for the prevention and mitigation of severe accidents …" in the final safety analysis report of an application for design certification.

Specific application guidance to meet this requirement are listed in Section C.I.19, "Probabilistic Risk Assessment and Severe Accident Evaluation" of Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants," and Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors" of NUREG-0800, "Standard Review Plan."

SECY-93-087 contains the following general criteria that have been approved by the Commission. These will be used to benchmark plant safety for new light water reactor (LWR) designs,

Hydrogen Mitigation

- accommodate hydrogen generation equivalent to a 100 percent metal-water reaction of the fuel cladding.
- limit containment hydrogen concentration to no greater than 10 percent.
- provide containment-wide hydrogen control for severe accidents.

Core Debris Coolability

- provide reactor cavity floor space to enhance debris spreading.
- provide a means to flood the reactor cavity to assist in the cooling process.
- protect the containment liner and other structural members with concrete, if necessary.
- ensure that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Factored Load Category for concrete containments, for approximately 24 hours. Also ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

High Pressure Melt Ejection (HPME)

- provide a reliable depressurization system.
- provide cavity design features to decrease the amount of ejected core debris that reaches the upper containment.

Containment Performance

- The containment should maintain its role as a reliable, leak-tight barrier for approximately 24 hours following the onset of core damage under the more likely severe accident challenges. Following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

- The conditional containment failure probability (CCFP) should not exceed approximately 0.1.

Equipment Survivability

- Maintain reliability of functions during relevant severe accident scenarios.

The guidance in SECY 93-087 regarding describing and analyzing the relevant design features has been approved by the Commission. Accordingly, the staff finds AREVA's request to be reasonable. It should be pointed out, though, that there are other documents listed in Section C.I.19 of Regulatory Guide 1.206 that provide guidance on how to address severe accidents. The staff will also consider these during its review of the application for a design certification.
The topical report includes a clear and useful discussion of the U.S. EPR design features as

they relate to severe accident performance. In addition to traditional pressurized water reactor (PWR) design features and plant systems, the U.S. EPR includes a number of additional features to prevent or mitigate the effects of severe accidents. These features include:

1. In-containment refueling water storage tank (IRWST), to maintain a large reserve of borated water at a homogeneous concentration and temperature. This water is used to flood the refueling cavity for normal refueling, and is the safety-related source of water for emergency core cooling system (ECCS) during a loss-of-coolant accident (LOCA) and for core melt cooling in the event of a severe accident. It is also the water source for the severe accident heat removal system (SAHRS).

2. Severe accident depressurization valves, to avoid high-pressure core-melt ejection and eliminate the possibility of containment bypass from induced steam generator tube ruptures under high reactor coolant system pressures and temperatures during postulated severe accidents.

3. Combustible gas control system (CGCS), to avoid the risk of containment failure due to rapid hydrogen combustion. This system includes features to reduce the concentration of hydrogen in the containment, as well as features designed to enhance hydrogen mixing and distribution.

4. Core-melt stabilization system, which is a dedicated ex-vessel system to accommodate molten core debris in the event of vessel breach. The goal is to stabilize molten core debris before it can challenge containment integrity. The system consists of four features: the cavity, which utilizes a combination of sacrificial concrete and a protective layer of refractory material to provide temporary melt retention; a melt plug and gate at the bottom of the cavity, to provide a pre-defined failure location; a melt discharge channel composed of a steel duct lined with refractory material to direct the conditioned melt to a spreading compartment; and the spreading area, which consists of a dedicated cooling structure lined with sacrificial concrete to promote stabilization and coolability of the debris. The cavity would not be flooded, thus removing the threat of a large ex-vessel steam explosion.

5. Severe accident heat removal system, which is a dedicated thermal-fluid system used to control the environmental conditions inside containment following a severe accident. It has four modes of operation, including: passive cooling of molten core debris; active spray for environmental control of the containment atmosphere; active recirculation cooling of the molten core debris and containment atmosphere; and active back-flush of the IRWST. The SAHRS train is located in a dedicated, radiologically-controlled room within one of the four plant safeguard buildings. Dedicated portions of the component cooling water and emergency service water systems are used to transfer heat to the ultimate heat sink.

6. Dedicated instrumentation and controls that are part of the overall severe accident management concept. These include:

    a. provisions to support reliable reactor coolant system (RCS) depressurization (measurement of core outlet temperature, ability to manually actuate the severe accident depressurization valves, position indication for severe accident depressurization valves,

and measurement of RCS pressure);

b. monitoring of melt progression (thermocouples to indicate RPV failure, monitoring arrival of corium into spreading compartment which triggers actuation of the passive flooding valves of the SAHRS, and thermocouples to monitor basemat failure threat;

c. support of hydrogen mitigation (measurement of hydrogen concentration, actuation of hydrogen mixing dampers, and position indicators of hydrogen mixing dampers);

d. monitoring of containment heat removal (measurements of appropriate pressures, temperatures, flow rates, and water levels);

e. Monitoring of overall plant behavior during a severe accident.

7. Severe accident uninterruptible power supply system to supply power to equipment and process instrumentation needed for severe accident management.

Inclusion of these features meets the intent of the guidance in SECY 93-087, although there are additional severe accident considerations that must be addressed. These include (but are not limited to):

1. Maintaining containment integrity, preventing an uncontrolled release of fission products for approximately 24 hours following the onset of core damage. Furthermore, the conditional containment failure probability should not exceed 0.1 for the aggregate of core damage sequences.

2. Evaluating the need for a dedicated containment vent penetration.

3. Defending against common-mode failure in digital instrumentation and control systems.

4. Analyzing the potential for and effects of multiple steam generator tube ruptures.

3.0 TECHNICAL EVALUATION

The guidance in SECY 93-087 and other NRC documents is aimed at ensuring that new reactors will have significantly better capabilities of preventing and mitigating the consequences of severe accidents than current plants, as measured by core damage frequency, large release frequency, and conditional containment failure probability in a Level 2 PRA. Since the U.S. EPR is an evolutionary PWR design, generally similar to existing PWRs, reduction in core damage frequency is achieved by incorporating proven technology with innovative system configurations to enhance safety. Reducing large release frequency can be achieved by including design features to address the issues identified in SECY 93-087. Maintaining the conditional containment failure probability below 0.1, while reducing the core damage frequency, can be achieved by making the containment more robust and by essentially eliminating the probability of containment bypass during a severe accident. To reach the conclusion requested by AREVA, the staff must be convinced that the approach described in

the topical report will lead to reductions in core damage frequency and large release frequency, relative to current generation reactors, while maintaining the conditional containment failure probability below 0.1. The following technical evaluation is carried out in this context.

3.1 U.S. EPR Severe Accident Management Systems

The selection of the U.S. EPR severe accident management systems is based on the following concepts:

- Severe accident management will not be based on stabilizing the core melt while it is still in the RPV; and
- The RCS will be depressurized, essentially turning all severe accident scenarios into low pressure events.

Consequently, severe accident management strategies would rely on the ability to cool core debris ex-vessel, while maintaining containment integrity.

3.1.1 Design Features to Reduce Core Damage Frequency

Some of these features in the U.S. EPR include a large free volume in the containment building of 2.8×10^6 ft^3 and a design pressure of 62 psig; a series of independent core cooling systems [four medium head safety injection (MHSI) systems and four low head safety injection/residual heat removal (LHSI/RHR) systems that draw water from the IRWST, as shown in Figure 2-6 of Reference 1]; an emergency boration system to shut down the reactor; and a reactor coolant depressurization system that features three pressurizer safety valves.

Safety injection within the U.S. EPR is performed by an MHSI system, an LHSI system, and four accumulators (ACC). These safety-related systems consist of four independent trains that are physically separated and protected within the safeguard buildings. The MHSI system draws borated water from the IRWST and injects it into the cold leg at a pressure lower than the main steam safety valve (MSSV) setpoints to ensure that in the event of a steam generator tube rupture (SGTR), primary inventory cannot be released directly to the environment. The four accumulators are connected to the RCS cold legs (CL) and provide injection when the RCS pressure falls below the corresponding setpoint.

The RHR system is combined with the LHSI system; however, a different operating configuration is used to transfer residual heat to the plant cooling water systems. The RHR system of each plant safety train includes suction on the hot leg of each RCS loop where it draws heated water and pumps it through a heat exchanger in the safeguards building before being injected back into the cold leg of that same RCS loop. This is an active system with emergency power provided by diesel generators.

The U.S. EPR includes an Emergency Boration System (EBS) that can be used to provide borated water at high pressure to shut down the reactor following accidents. The EBS consists of 2x100 percent trains that can be used as a safety-related means of maintaining the reactor in a shutdown state at any temperature in case of unavailability of the chemical and volume control system (CVCS). The EBS is also a means to mitigate the effects of an anticipated transient without scram (ATWS) by bringing the reactor into a sub-critical state.

To prevent RCS overpressure, the U.S. EPR includes three pressurizer safety valves (PSVs) at the top of the pressurizer. These PSVs discharge to a common header connected to the pressurizer relief tank (PRT). The PRT is protected against overpressurization by use of rupture disks. If the PRT pressure exceeds a specified upper value, the rupture disks will burst, allowing fluid to exit the PRT and relieve the pressure. The outlets of the rupture disks are connected to a piping system that distributes the fluid to the reactor coolant pump (RCP) rooms.

The depressurization system also includes dedicated valves to ensure that the RCS can be depressurized in the event of a severe accident.

Containment heat removal in the U.S. EPR is ensured through the use of active systems. During normal operation or hot shutdown conditions, the containment cooling ventilation system (CCVS) removes heat released by the operation of plant components. The CCVS is not a safety-related system but is designed with sufficient redundancy to ensure reliable operation. The large, robust containment, and greater heat capacity of the containment and internal structures, allow for more time to achieve pressure and temperature control during design basis accidents. Containment heat removal would be ensured by the LHSI/RHR heat exchangers outside containment.

3.1.2 Design Features to Reduce Large Release Frequency

The most significant U.S. EPR features that would act to reduce the large release frequency, relative to current-generation reactors, are the severe accident depressurization valves, the combustible gas control system, the core melt stabilization system, the severe accident heat removal system (Reference 1, Figure 2-7), and the dedicated instrumentation and control features for severe accidents.

3.1.2.1 Severe Accident Depressurization Valves

The severe accident depressurization valves are intended to prevent RCS failure at high pressure, to avoid high-pressure core-melt ejection and direct containment heating (DCH), and eliminate the possibility of containment bypass from induced steam generator tube ruptures. The effect would be to convert high-pressure core melt sequences to low pressure sequences. As shown on Figure 2-8 of Reference 1, the severe accident depressurization valves are independent of the PSVs, a safety-grade system that provides RCS relief for an overpressurization event. Both the severe accident depressurization valves and the PSVs discharge to the PRT. The PRT is protected by rupture disks and connected to two of the four RCP rooms, as shown in Figure 2-9 of Reference 1. Failure of the PRT rupture disks encourages the mixture of non-condensable gases in the pump rooms, to prevent preferential accumulation of hydrogen in the containment rooms.

3.1.2.2 Combustible Gas Control System (CGCS)

Combustible gas control in the containment is necessary to avoid the risk of containment failure due to fast deflagration or from accidental ignition of a critical gas mixture. The purpose of the dedicated CGCS is to minimize the post-accident hydrogen combustion risk within containment. The CGCS system is divided into two subsystems corresponding to their operational functions: the Hydrogen Reduction System (HRS); and the Hydrogen Mixing and Distribution System.

The HRS consists of 41 large and 6 small passive autocatalytic recombiners (PARs) installed in various parts of the containment. Each PAR consists of a metal housing designed to promote natural convection with a gas inlet at the bottom and a lateral gas outlet at the top. The horizontal cover of the housing at the top of the recombiner protects the catalyst against direct water spray and aerosol deposition. Numerous parallel plates with a catalytically active coating (Pt/Pd substrate) are arranged vertically in the bottom of the housing. Accessibility to the catalytic plates is provided by the use of a removable inspection drawer.

Hydrogen and oxygen in containment gas mixtures are recombined upon contact with the catalyst in the lower part of the housing. The heat from this reaction in the lower part of the recombiner causes a reduction in gas density in this area promoting natural circulation through the PAR and ensuring high efficiency of recombination. In the presence of oxygen, the PARs will automatically start if the threshold hydrogen concentration is reached at the catalytic surfaces. The recombination rate depends mainly on the hydrogen density seen by the PAR. An increasing hydrogen concentration enhances the removal rate up to a type-specific upper limit.

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 PARs are installed above the floor to provide unobstructed inflow and easy access to facilitate maintenance. They are also arranged to avoid direct contact with spray water (despite their qualification to operate in the presence of water). They are designed and located to ensure that the global concentration of hydrogen in the containment atmosphere is maintained below 10 percent by volume during phases of an accident resulting in oxidation up to 100 percent of the zirconium surrounding the reactor core fuel, and ensure that the global hydrogen concentration can be maintained below the lower flammability limit of 4 percent by volume of the containment atmosphere in the long term.

The hydrogen mixing and distribution system is designed to ensure 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 equipment rooms, thereby eliminating any potential dead-end compartments where non-condensable gases could accumulate. This ability to transform the containment into a single convective volume is supported by a series of mixing dampers and blowout panels.

3.1.2.3 Core Melt Stabilization System

The U.S. EPR is equipped with a dedicated ex-vessel system to accommodate molten core debris, including the entire core inventory and reactor internals, which penetrates the RPV. The goal of this system is to stabilize molten core debris before it can challenge the integrity of the containment. It is attained through the combined effects of the following portions of the core melt stabilization system (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 concept by providing a pre-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 core debris. The general configuration of the CMSS is provided in Figure 2-13 of Reference 1.

Between the RPV and cavity is a layer of insulation limiting heat loss from the vessel and six walls aligned radially from the melt plug effectively creating six azimuthal sectors. These walls are designed to limit the downward expansion of the lower head resulting from contact with a molten pool and 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. These features ensure that the reactor cavity can withstand the loads resulting from RPV failure.

The initial conditions for melt stabilization 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 predicted 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 must be 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, ensures that even in case of an incomplete first release of melt from the RPV, practically the entire core inventory will be collected in the cavity prior to spreading and stabilization.

The sacrificial layer consists of a layer of siliceous concrete with high iron-oxide content, enabling oxidation of the remaining zirconium and uranium within the melt, ensuring a low melt temperature and viscosity for spreading. The high SiO2 composition helps the conditioning 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.

The melt plug and gate act as a predefined failure location in the reactor cavity through which melt will flow to the spreading compartment. The upper part of the melt plug is essentially a layer of sacrificial concrete with the same composition as the sacrificial layer within the cavity. This layer of concrete is backed by a metal plate (referred to as the gate). 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.

Once the molten core debris comes into contact with the gate, the intensity of the convection within the molten pool is expected to almost instantaneously destroy the gate. The rate of melt discharge after opening the melt plug is substantially greater than that necessary to ensure adequate spreading. If the gate initially failed over less than its full cross-section, the diameter of the generated hole 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.

Following the failure of the gate, the melt will progress 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 channel consists of a steel structure that is embedded within the structural concrete of the containment. The bottom, side walls and top of this structure are layered with refractory material. This protective layer consists of zirconia bricks which have a low thermal conductivity and greater mechanical strength than concrete.

The spreading compartment consists of a large horizontal concrete surface over which the molten core debris can be dispersed. Spreading increases the surface-to-volume ratio of the molten core debris to ensure fast and 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 2-18 of Reference 1.

The design of the spreading compartment prevents accumulation of any large amount of water and ensures that molten core debris will be spread 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 concrete of the spreading compartment covers a dedicated cooling structure used to cool the molten core 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 to ensure that the cooling elements will be flooded with water from the IRWST prior to the initial contact between them and the molten core debris. The structural elements are joined using flexible connections to ensure that the cooling structure is insensitive to expansion and deformation.

The siliceous sacrificial concrete of the cooling structure is different from that used in the reactor cavity. Once the concrete has been ablated by the molten core debris, the molten pool will rest on top of the cooling structure. The combined cooling elements will form a series of parallel cooling channels that provide flow paths for water from the IRWST to flow under the melt, along the side walls and onto the top of the molten core debris, cooling and stabilizing the melt.

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. 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. In parallel with the inflow of water, the spread melt interacts with the sacrificial concrete covering the horizontal and vertical cooling plates. The resulting delay ensures that the walls of the cooling structure will always be cooled on the outside prior to the first contact with the molten corium.

3.1.2.4 Severe Accident Heat Removal System

The SAHRS is a dedicated thermal-fluid system used to control the environmental conditions within the containment following a severe accident. To ensure substantial margin in containment pressure control the SAHRS has four primary modes of operation, each playing a role in controlling the environmental conditions within the containment so that its fission product retention function is maintained. These modes include:

- Passive cooling of molten core debris
- Active spray for environmental control of the containment atmosphere
- Active recirculation cooling of the molten core debris and containment atmosphere
- Active back-flush of the IRWST.

Each of the two identical SAHRS trains is located in dedicated, radiologically-controlled rooms within two of the four plant safeguard buildings, and includes:

- A dedicated suction line from the IRWST
- Containment isolation valves
- A pump to support active recirculation
- A heat exchanger for containment heat rejection
- Discharge lines 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 (CCW) and Essential Service Water (ESW) trains. During operation, the three possible flow paths downstream of the pump and the heat exchanger are:

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

The general configuration of a single SAHRS train is provided in Figure 2-23 of Reference 1.

Once molten core debris is within the spreading compartment, water from the IRWST will passively start to fill the cooling structure, which would be filled within five minutes. Water then overflows into the spreading compartment until it is hydrostatically balanced with water from the IRWST. This flooding is expected to result in submersion of the spreading area and transfer channel, as well as a portion of the reactor cavity, thereby stabilizing any residual core debris in those areas.

Operating in this passive mode, IRWST water supplied by the SAHRS will be boiled-off as steam and released into the free volume of the containment through the steam chimney directly above the spreading compartment. As this process continues, the temperature and pressure within the containment will steadily increase; however, the containment is designed with sufficient free volume and structural heat sinks that atmospheric conditions of the containment will not approach design limits for several hours following the onset of core damage. At this point the SAHRS is configured to operate in the containment spray mode. The SAHRS will then take suction from the IRWST and coolant will flow 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 as the water droplets fall through the containment atmosphere. The resulting condensate then flows along the structural elements of the containment before being routed back into the IRWST for continued recirculation.

The U.S. EPR containment spray is smaller in capacity than containment spray systems of conventional plants and other evolutionary designs, because it is only intended to be used for severe accident mitigation.

As a core melt accident progresses, it may 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 core 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 will become subcooled. Decay heat will now be removed from the spread melt by single-phase flow, instead of by evaporation into the containment atmosphere. This way an ambient pressure level can be maintained in the containment in the long-term, thereby further reducing the potential for the release of activity.

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 opened 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.

The final mode of operation of the SAHRS is to provide a back-flushing function within the IRWST. 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 SAHRS is used for back-flushing; therefore, the system can operate in this mode while continuing operation in another containment cooling mode.

3.2 Assessment of AREVA's Safety Issue Resolution Approach

The steps in the AREVA safety issue resolution approach, as shown on Figure 1, include the identification of safety goals indicating that the issues identified in SECY 93-087 have been properly addressed, documentation of severe accident engineering activities addressing these

issues, derivation of a calculation matrix, and presentation of analysis results based on the derived calculation matrix. Identifying the test programs and required analytical methods related to each issue is important in order to verify that the severe accident design features are adequate and appropriate. Identification of the necessary analyses involves engineering insights that combine regulation, industry experience, fundamental understanding of thermal-hydraulic and severe accident phenomena, and risk/consequence factors. The AREVA approach incorporates risk/consequence information from PRA to identify events that could challenge containment integrity and categorizes event classes based on dominant phenomena. Then, deterministic evaluations would demonstrate the ability to mitigate the consequences of a severe accident, and PRA would quantify the risk reduction associated with these features.

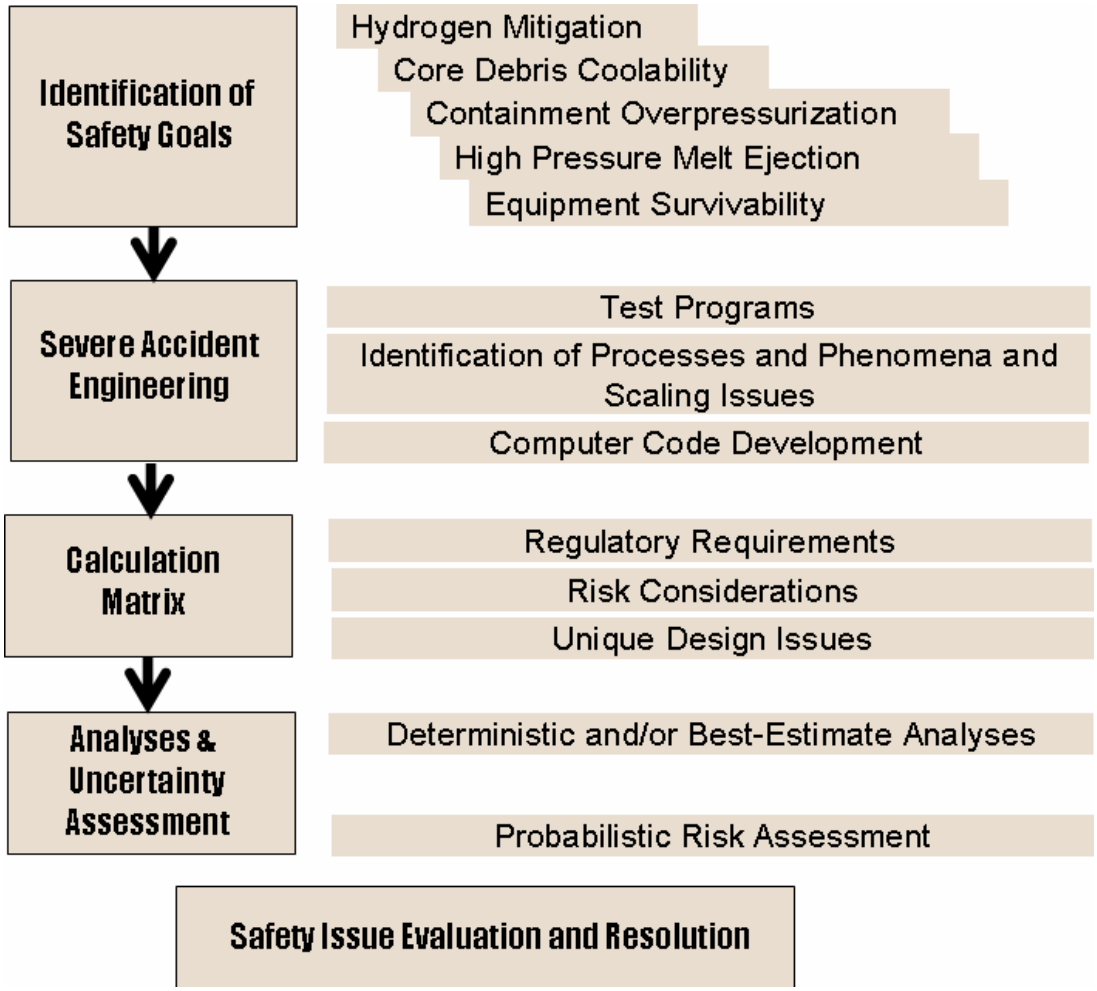

Figure 1. Process for Severe Accident Safety Issue Resolution

It is apparent from reviewing the topical report that the U.S. EPR severe accident response features have been designed to reduce or eliminate many of the uncertainties associated with severe accident progression. When AREVA prepares the PRA for the U.S. EPR, many scenarios will be exercised to evaluate the likelihood of core damage (Level 1) and containment failure (Level 2) in order to demonstrate the acceptable performance of severe accident response features of the U.S. EPR. In deriving a calculation matrix, AREVA expects that the

characterization of the relevant events and event classes of severe accidents identified by PRA will credit:

- isolated containment (no bypass issue)
- full RCS depressurization
- passive protection of the containment liner
- available, active SAHRS
- well-mixed containment atmosphere
- reliable hydrogen recombiner performance.

AREVA's approach to containment performance evaluation has evolved to one centered on best-estimate analysis plus uncertainty complemented by deterministic analyses with bounding assumptions.

The results of the AREVA Safety Issue Resolution Evaluation Methodology will appear in the U.S. EPR design control document (DCD). They will provide a list of relevant severe accident scenarios that 1) provide insight into plant-specific severe accident processes and phenomena and 2) form the basis for a calculation matrix. Processes and phenomena will be characterized in terms of parametric performance ranges (i.e., an uncertainty measure) from applicable test program results and incorporated into computational analysis tools, such as modular accident analysis program (MAAP4) for production analyses and MELCOR for self-audits. These tools are used to establish the limits of these processes and phenomena through the demonstration of the U.S. EPR severe accident response capability to the relevant severe accident events.

AREVA has highlighted the severe accident issues of greatest relevance to the U.S. EPR by describing a hypothetical bounding severe accident. This enables them to identify the dominant processes and phenomena associated with severe accident scenarios. They also consider the severe accident mitigation features included in the design to either bound or eliminate uncertainties in processes and phenomena. The scenario chosen for evaluation is a large-break LOCA resulting from a large pipe break without active safety injection. Although a highly unlikely event, plant response to this scenario is such that the various severe accident design features selected by AREVA would come into play. That is, the RCS would be depressurized from the pipe break and the severe accident would not be arrested in-vessel. (Note that, if the actual scenario is initiated by a transient or by a small-break LOCA, the depressurization would result from opening the severe accident depressurization valves.) Core debris would eventually breach the RPV and fall into the cavity. Eventually, the plug at the bottom of the cavity would be melted through and the core debris would flow into the spreading room, where it would be cooled. The SAHRS would then function to help achieve a stable state.

By analyzing this scenario, the various phenomena can be associated with the safety issues identified in SECY 93-087 and the safety systems can be properly designed. It should be noted, however, that there may be a significant uncertainty associated with RCS pressure at the moment of RPV failure. Some provision to address this uncertainty should be made with regard to the HPME issue and to the issue of thermally-induced steam generator tube rupture. Therefore, it is assessed that the AREVA approach appears to be sound and thorough.

3.3 Relevant Research and Development Activities

In the topical report, AREVA has provided a comprehensive description of the various research and development activities pertinent to the severe accident safety issues. These supporting

activities address hydrogen mitigation, core debris coolability, high pressure melt ejection, containment performance, and equipment survivability.

AREVA has a database of research and development from participation in international programs, many of which have been explicitly designed with U.S. EPR features. In addition, many testing programs providing insights into severe accident phenomena are available in various technical reports and code manuals, which are identified in this section. Specific to the objectives of this topical is an assessment of the applicability of these tests relative to system scaling and ranges of key measurable variables.

The research and development work performed in parallel with the various U.S. EPR development phases has helped to achieve a better understanding of the underlying phenomena, improve available codes, models and data bases, and provide more realistic assumptions regarding initial and boundary conditions. Specifically, these programs are being used to:

- define applicability and uncertainty ranges to be considered in characterizing analyses
- aid in resolving issues, such as DCH and fuel-coolant interactions (FCI)
- validate computational tools for production analysis (e.g., MAAP4, etc.).

This aspect of the AREVA approach is a real strength, and serves to enhance credibility of their approach.

3.3.1 Hydrogen Mitigation

The U.S. EPR hydrogen mitigation concept aims at 1) preventing flammable configurations of combustible gases capable of breaching the containment, and 2) removing hydrogen in order to achieve global hydrogen concentrations below the ignition limit.

The system consists of recombiners, rupture and convection foils, and mixing dampers. Hydrogen mitigation is supported by the fact that depressurization of the RCS occurs directly into the containment atmosphere, via a relief tank with a rupture disc at two low locations near the elevation of the steam generator supports. This guarantees a large amount of well-mixed steam in the containment for nearly all scenarios.

To maintain containment integrity, in particular in the early phase of an accident, the following must safely be avoided:

- Any deflagration with Adiabatic Isochoric Complete Combustion (AICC) pressure above containment ultimate pressure,
- Fast deflagration with the potential to initiate a detonation,
- Local temperatures that pose a threat to the containment shell.

The justification of this concept is based on the application of experimentally founded criteria to determine the potential combustion mode with the goal of excluding flame acceleration and, in particular, Deflagration-to-Detonation Transition (DDT).

Despite recent progress, some uncertainties still exist that are associated with:

- Prediction of transient hydrogen production (e.g. reflood of a molten core)

- Refined application of combustion criteria to exclude fast deflagration and DDT to real containments (degree of confinement, potential of venting)
- Conditions for the development and stability of a standing flame (e.g., resulting from ex-vessel release at high temperature).

Several test programs examining the degradation of fuel with the generation of hydrogen have been performed; these are identified and discussed in the topical report. AREVA concludes from these tests that, while the phenomena related to hydrogen production are well understood, uncertainties in the total amount of metal that can be oxidized must be considered. For severe accident safety assessments, hydrogen production must be addressed as a bounding initial condition. The staff agrees with this conclusion.

AREVA has also identified and discussed the test programs for assessing hydrogen distribution in the containment building. They conclude that, in general, the transport of vapor and gases is well understood from experimental programs and is captured in validated analytical methods. The staff generally agrees, but plans to independently calculate hydrogen distribution.

Regarding hydrogen combustion phenomenology, AREVA identifies and discusses eight testing programs, and concludes that important uncertainties still exist. They state that the conservative flammability criteria that have been experimentally verified provide the strongest basis for assessing hydrogen combustion risk in the U.S. EPR. Because of this, no situations where fast deflagration may result can be allowed to occur.

Finally, the topical report discusses experimental programs for the hydrogen recombiners. AREVA concludes that its PAR has demonstrated the capacity for hydrogen control over the full spectrum of possible accident conditions. Uncertainties remain regarding the arrangement of the recombiners within the containment and the consideration of the recombiner as an ignition source. Nevertheless, they are sufficiently developed and tested for implementation.

3.3.2 Core Debris Coolability

The U.S. EPR design involves provisions for the retention and long-term stabilization of the molten core inside the containment. The corresponding scheme presupposes a depressurization of the primary circuit prior to the formation of a molten pool within the lower plenum of the RPV. After RPV failure the molten corium is intended to first accumulate in the reactor cavity and later relocate, in one event, into a lateral compartment. Spreading of the melt will be followed by flooding, quenching and sustained cooling of the corium.

An assessment on the performance of corresponding components required for core debris coolability begins with a characterization of the main processes involved in this sequence, namely:

- reactor vessel failure and initial release of corium from the RPV
- temporary melt retention in the reactor cavity involving accumulation and conditioning of the melt, and behavior of the protective shielding during melt attack
- failure of the melt plug
- melt stabilization phenomena including spreading of the melt on concrete, and melt flooding and stabilization.

As an ex-vessel severe accident mitigation strategy, the consequences of molten corium-concrete interaction (MCCI) contribute to the transformation of the melt into a stable configuration in a two-stage stabilization process, retention and spreading.

3.3.2.1 Vessel Failure Modes

Assuming the necessary conditions for reactor vessel failure are present, two vessel failure scenarios are considered most likely: (1) a “localized failure” in which a localized opening occurs near the vessel beltline (releasing molten core debris above the breach), and over time, moves downward, releasing additional debris, or (2) a “hinged failure” in which a localized opening occurs near the vessel beltline immediately followed by tearing of the vessel around nearly its entire circumference, and the lower head hinging/swinging downward and coming to rest on the cavity floor.

The U.S. EPR ex-vessel core debris coolability strategy eliminates uncertainties related to reactor vessel failure modes by introducing a temporary core melt retention phase. Nonetheless, the phenomena associated with reactor vessel failure are relevant to the demonstration of the U.S. EPR overall core melt stabilization strategy.

The topical report identifies four test programs related to reactor vessel failure. AREVA concludes that the research to date supports the premise that reactor vessel failure is most likely to occur from a localized point and eventually expand into a global failure. Heat flux distribution and manufacturing variations were shown to have a strong influence on RV failure. While manufacturing tolerances are generally expected to be tighter for nuclear power plants relative to test programs, heat flux uncertainty is significant. Nonetheless, AREVA believes that the U.S. EPR’s temporary melt retention feature compensates for this uncertainty. The staff will do independent calculations to confirm this hypothesis.

3.3.2.2 Temporary Melt Retention in the Cavity and Failure of the Melt Plug

Core debris would attack the concrete in the cavity after leaving the RPV. The nature of the core debris-concrete interaction process has been studied in seven experimental programs identified in the topical report. Besides yielding a large amount of data on melt chemistry, gas rate and composition, temperature and erosion rate which serve as the validation basis of existing MCCI models, these experiments also provide insight in the general phenomenology of melt-concrete interaction, in particular regarding the:

- differences in the behavior between metallic and oxidic melts
- ratio between radial and axial ablation
- mixing of metallic and oxidic phases

The experiments with metallic melts showed that progression into the concrete will proceed very quickly, as long as the melt is superheated. At high melt temperatures, heat fluxes can be so high that the concrete at the interface is physically disintegrated, so that small pieces are mixed into the melt and are molten not at the interface but either in the bulk or at the upper surface. This process leads to a fast cool-down and, in later stages, to the formation of metallic crusts at the melt-concrete interface. Further, as the solidification temperature of steel is higher than the decomposition temperature of the concrete, concrete decomposition and melting has been observed to continue even after the metal was completely solidified.

The experiments that employed oxidic corium melts revealed a different behavior because the thermal conductivity of solid core oxides is very low, and solid oxide melts freeze over a wide temperature range, due to the presence of high-melting refractory (U, Zr)O2 and low-melting concrete (SiO2, CaO) components in the melt. The solidus-liquidus temperature range, within which the melt contains a certain amount of suspended refractory particles, can be more than 1000 K. During contact with a cold medium, such as concrete, the melt becomes surrounded by a crust that is enriched with refractory material. This crust maintains the molten pool at a high temperature because its low thermal conductivity limits the heat transfer from the pool. Due to its low thermal conductivity, measured concrete erosion rates for oxidic core melt were generally much lower than for metallic melts, in the period before reaching equilibrium conditions. For oxidic melts concrete melting is considered to be the dominant ablation mechanism.

At the beginning of the phase of temporary retention in the reactor cavity, the lighter metallic corium is located atop the heavier oxide. In this period, the speed of concrete erosion in the region of the metal phase (primarily lateral) will be faster than that in the oxidic phase. This is due to its significant level of superheat, and to the so-called “focusing effect,” in which the metal redirects part of the thermal energy it receives from the oxide into the surrounding wall, thus leading to a fast penetration of the sacrificial concrete layer.

To protect the structural concrete from the metallic melt during this phase, zirconia shielding is installed behind the sacrificial concrete. This refractory protective shield surrounds the entire lower reactor cavity region and thus establishes a defined final limit for melt progression. It essentially predefines the maximum amount of sacrificial concrete that can be added to the melt during its retention in the reactor cavity and makes the retention function independent of the uncertainties regarding two dimensional melt progression, i.e., lateral vs. axial erosion.

Under most conditions, this zirconia layer will only be reached by the oxidic melt at the end of the retention phase. Therefore, its main objective is to survive the contact with the metallic melt during the early period, when the metal melt is still atop and heated by the oxide melt. Tests have shown that, even after many hours, penetration of the refractory material would be minimal. AREVA concludes that zirconia is an appropriate protective material that can be used in the cavity.

The melt plug, which isolates the reactor cavity from the spreading compartment, consists of a metallic plate with a low melting temperature – the “gate” - below a sacrificial concrete cover. There is no zirconia layer below this cover. The concrete cover is as thick as the sacrificial concrete layer on the cavity floor (see Figure 2-13 of Reference 1). Experiments have shown a predominantly uniform axial ablation of concrete, suggesting that retention time in the cavity is primarily controlled by the thickness of the sacrificial concrete cover and not by the time to fail the gate after melt contact.

The experimental programs on core debris-concrete interactions and the effectiveness of the protective ceramic layer have allowed AREVA to conclude that, by including a sacrificial concrete layer, sufficient time would be available to allow most, if not all, of the core debris to exit the RPV before the core debris reaches the melt plug. Then, upon failure of the melt plug, the molten material would be flushed into the spreading compartment. The staff will carry out MELCOR calculations to confirm that most of the core debris in the cavity would flushed into the spreading compartment.

3.3.2.3 Melt Stabilization in the Spreading Compartment

Once the melt plug and gate are thermally destroyed, the melt is guided through a horizontal transfer channel into the spreading compartment. Due to its large cross section and its inert, temperature-resistant walls, the transfer channel itself is expected to have no retarding effect on the melt flow. Once reaching the dry spreading compartment the melt will quickly spread out over the entire floor and begin to interact with the concrete. The key advantage of the spreading compartment is the increase of the surface area-to-volume ratio, thus, enhancing the overall coolability of the melt. SECY-93-087 recommends a “floor sizing criterion” surface area of 0.02 m^2/MWth. Considering the U.S. EPR’s full power of 4590 MWth, the design value of 170 m^2 greatly exceeds the recommended area of 92 m^2.

The topical report references six experimental programs pertaining to melt spreading. These confirm that the melt would spread uniformly and cover the entire floor area, provided it has sufficient flow rate and melt superheat.

Stabilization of the melt would occur once sufficient water from the IRWST enters the cooling channels and, at the same time, floods the melt from above. AREVA expects that molten metal would solidify within a few hours and molten oxide material within a few days.

Flooding and quenching mechanisms have been observed in several test programs. These experiments showed that flooding will lead to bulk cooling and will create an oxidic crust on top of the melt. The top flooding would tend to crack the crust and allow water ingression to cool the molten material.

AREVA concludes that, due to the high surface area-to-volume ratio created by the spreading process, and the fact that the melt is completely surrounded by cooled surfaces, a safe enclosure of the corium within crusts will be achieved soon after the end of the MCCI in the spreading area. The denser metallic melt fraction at the bottom is predicted to freeze within the first few hours. Solidification of the decay-heated oxidic melt will take longer but is estimated to be complete also after a few days. Thus, the melt would be stabilized.

The staff will carry out independent calculations to confirm that most of the core debris in the cavity would be flushed into the spreading compartment.

3.3.3 High Pressure Melt Ejection

AREVA states that HPME and associated direct containment heating (DCH) are not considered relevant severe accident phenomena for the U.S. EPR, primarily because of the severe accident depressurization system. In addition, a low core power density core, a RPV lower head without penetrations, and a torturous pathway from the reactor cavity to the upper containment all contribute to preventing or mitigating the potential consequences of high-pressure melt ejection. In coming to the conclusion that containment challenge from a HPME event is remote, the existing literature on this subject was evaluated by AREVA and summarized in the topical report.

Several experiments and analyses have been performed to study HPME. In addition, the evaluation of the various reactor cavity configurations in current-generation nuclear power plants under a variety of scenarios was presented in a DCH closure report for Westinghouse large dry containment plants (Reference 3). In terms of size and environmental conditions, the

U.S. EPR is very similar to the Westinghouse containment. This work on DCH closure demonstrated a conditional containment failure probability of less than 0.1, which was considered a sufficient threshold.

Based on the available experiments and their analyses, the issue of HPME-driven DCH is considered by the staff to be resolved for current-generation PWRs. The U.S. EPR design, incorporating several design features that are capable of either similar or enhanced preventive response to an HPME, precludes the potential mechanisms for HPME initiation and subsequent DCH. Consequently, AREVA states that this issue should also be considered resolved for the U.S. EPR design. The staff would concur once a confirmatory integrated analysis is provided to show that, under HPME/DCH conditions, the calculated peak pressure in the containment is well below the ultimate failure pressure. In parallel, the staff plans to independently confirm the adequacy of the severe accident depressurization system, and to determine the pressure increase from HPME/DCH.

3.3.4 Containment Performance

The containment provides the ultimate protective barrier with regard to release of radioactivity to the environment in the event of a severe accident. Related to containment overpressurization, the relevant severe accident design objectives for the U.S. EPR containment are:

- to maintain a leak-tight barrier for 24 hours following core damage
- beyond the initial 24 hours, the containment will continue to act as a barrier against uncontrolled fission product release
- to assure that elevated pressures and temperatures that may result from a severe accident do not cause its uncontrollable failure.

During a severe accident, the primary sources of mass and energy that could cause containment overpressurization can occur as the result of RCS depressurization (either by LOCA or actuation of the primary depressurization valves) coupled with the generation of noncondensable gases from MCCI and steam addition resulting from quenching and stabilization of molten core debris in the spreading compartment.

Following the initial pressure rise from RCS depressurization, the containment pressure will be moderated passively by the heat capacity of the containment walls and internal structures. Further pressure reduction is made possible first by the CGCS, which through the use of PARs, in the presence of oxygen, recombines hydrogen and oxygen into water vapor. The recombination of hydrogen alone does not impact containment pressure; however, the conversion of hydrogen into a condensable form enhances the performance of the SAHRS containment spray by maximizing the water vapor concentration.

The SAHRS containment sprays are started manually following the initial 12 hours of a severe accident. This feature both decreases containment pressure by condensing the steam generated in the containment and reduces the potential for further pressure increase by removing decay heat from within the containment airspace and from the molten core material in the spreading compartment.

The containment spray system is functionally equivalent to those present in current-generation PWRs for the mitigation of design basis LOCAs. The basis for these installed units draws on

test program experience including Carolinas-Virginia Tube Reactor Containment (Reference 4) and Battelle Northwest Laboratory's Containment Spray Experiments (Reference 5). The 12-hour delay allows the CGCS to perform with a high degree of reliability to fulfill its design goal of reducing hydrogen concentration to 4 percent in within that time.

If core debris and water come into contact after vessel breach, fuel-coolant interactions can cause the containment pressure to increase. In certain circumstances, steam explosions could possibly occur, leading to a highly-energetic pressure rise. The U.S. EPR includes several design measures that preclude ex-vessel steam explosions, such as an initially dry reactor cavity and dry spreading area, the addition of silica-rich sacrificial material to the melt before ex-vessel flooding, and the controlled addition of water to the top of the melt after spreading.

Following a LOCA or RCS depressurization, condensation may form in the spreading compartment. However, several tests, including KATS, have shown that for water layers of height 1 - 10 mm, the presence of water did not have any adverse effects on the global spreading. Moreover, while local steam explosions occurred for both the metallic and oxidic melt, they were never very energetic and they blew away the water in front of the melt thus avoiding further reactions along the flow path. Discussion of the principal tests related to flooding of the spreading compartment appears in the topical report.

AREVA concludes that, based on the unique design of the U.S. EPR, the large body of research conducted by several institutions, and the opinions of experts worldwide, the likelihood of fuel-coolant interactions (FCI) resulting in containment failure from either in-vessel or ex-vessel mechanisms is considered remote. Related to the U.S. EPR-specific design, two design characteristics inherently impede potential containment failure:

- the additional structure surrounding the U.S. EPR reactor vessel further mitigates adverse consequences of an in-vessel FCI-induced steam explosion by eliminating a clear path to the containment shell.
- the design of the spreading compartment preserves a dry environment eliminating the "fuel" required to create an ex-vessel steam explosion.

AREVA concludes that explicit treatment of U.S. EPR fuel/coolant interactions is unwarranted. The staff plans to confirm the adequacy of AREVA's approach by performing independent calculations.

3.4 Analysis Methods

AREVA provides an overview of the analytical methodology in the topical report, that includes a description of how the various computer codes are used together to provide comprehensive coverage of relevant severe accident phenomena associated with the U.S. EPR. A description is provided of each code's applicability to severe accident phenomena and a summary of code capabilities is provided. In addition, this section provides a summary of the validation and verification basis for each code and the scope of deterministic severe accident analyses for the U.S. EPR.

The U.S. EPR analytical methodology is based on the use of MAAP4.07 (Reference 6) as the primary integral analysis tool and the use of MELTSPREAD-1.4 (Reference 7) and WALTER (Reference 8) to supplement the integrated results. A significant code development effort was undertaken to expand the capabilities of MAAP4 to allow the analytical treatment of severe

accident phenomena in the U.S. EPR. The end result of these efforts was a revision to MAAP4 (i.e., version 4.07), which now allows for the vast majority of U.S. EPR severe accident phenomena to be analyzed in an integrated method. The use of other separate-effects codes and the need for manual coupling of inputs and outputs has been effectively minimized resulting in a robust analytical methodology.

The separate-effects codes MELTSPREAD-1.4 and WALTER were used to confirm some of the MAAP4.07 calculations and, in some cases, to provide a more detailed treatment of one or more phases of a severe accident. The flow of data to and from the various codes is depicted in Figure 2. The relationship between the three codes is not one of mutual exclusion. All phases of the severe accident are treated by MAAP4.07 and there is considerable overlap in the scope of phenomena addressed by each code.

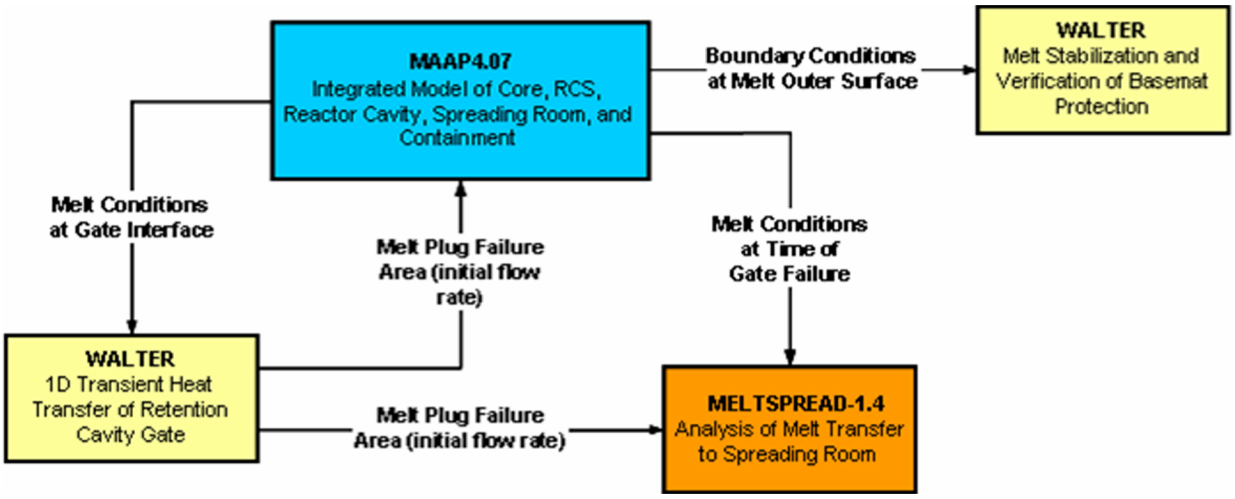

Figure 2. Overview of U. S. EPR Severe Accident Analytical Methodology

MAAP4.07 is used to develop the boundary and initial conditions for defining the problems analyzed with MELTSPREAD-1.4 and WALTER. The separate-effects codes are then used to verify selected parameters used to characterize the various system performance parameters in MAAP4.07.

3.4.1 MAAP 4.0.7 Description

MAAP4 is an integral systems analysis computer code for assessing severe accidents. It includes models required for analysis of a wide spectrum of severe accident phenomena that might occur within the primary system and containment. It has the capability to simulate the progression of a severe accident sequence, including the release of fission products, from a set of initiating events to either a safe, stable state or to an impaired containment condition.

The basic MAAP4 code architecture consists of a main program that directs the execution through several high level subroutines. These subroutines call a sequence of system and region subroutines at each time step, which in turn call phenomenological subprograms as required. The MAAP4 time steps vary through program execution based on user specified maximum and minimum limits. Separate time step ranges can be specified for the core and containment regions.

The MAAP 4.0.7 severe accident modeling capability is summarized in Table 1. The topical report discusses each of the phenomenological models as applicable to the U.S. EPR. A few new and/or modified models need to be discussed here, however.

Table 1. MAAP4.07 Severe Accident Phenomena Modeling Capability

Phenomena	MAAP 4.0.7
1. Fuel rod degradation	Yes
2. Melt progression/ablation through surrounding core structure	Yes*
3. Core melt relocation to lower head	Yes
4. In-vessel fuel-coolant interaction	No
5. In-Vessel Oxide/Metal separation	Yes
6. Crust formation and failure	Yes
7. In-vessel debris formation	Yes
8. RCS/RPV failure modes	Yes*
9. Melt conditioning in reactor cavity/MCCI	Yes*
10. Melt spreading in spreading compartment	No
11. MCCI in spreading compartment (w/oxide-metal stratification)	Yes* (No)
12. Spreading compartment flooding and basemat cooling	Yes*
13. Steam/hydrogen transport	Yes
14. Hydrogen recombination	Yes*
15. In-containment hydrogen combustion	Yes
16. Long-term heat removal from the containment	Yes*
17. Fission product transport in containment	Yes

* New and/or modified capability with MAAP4.07

Melt Progression/Ablation Through Surrounding Core Structure

The U.S. EPR core design differs from the current-generation PWR core designs modeled by the MAAP4 code. Specifically, the current designs have a core region surrounded by a water baffle region, which is typically considered the core bypass, with the core barrel region surrounding this bypass. With the use of a heavy reflector/core barrel in the U.S. EPR, and the resulting elimination of the bypass region, the accumulation of molten material within the core under severe accident conditions would result in direct attack by the molten core material on the core barrel. The relatively large thickness of this steel component will subsequently delay the relocation of the melt into the lower plenum. In the MAAP4.07 model, severe core degradation would result in melting of the core material, accumulation of this material within the core region, and eventual melting through the core former plates that form the inner region of the bypass zone (water baffle).

The MAAP4 model was modified to represent this more substantial heat structure and the radial attack associated with lateral relocation of the molten core material. As the thermal attack of the core barrel progresses, the core barrel is assumed to remain intact until the molten steel layer penetrates through the entire thickness at a given elevation. When penetration occurs, the molten core debris at that elevation and all elevations above the failure location is influenced by drainage of the molten core material through the failure site in the core barrel and, thus, into the downcomer region. This enables molten core material to flow into the lower plenum.

Following the initial failure, core material drains over the outer surface of the core barrel and energy is transported from the molten core debris to the core barrel outer surface. If this energy transport is sufficient to cause the wall to approach the melting temperature, the failure location can propagate downward and enable an increased amount of molten material to drain into the lower plenum. This downward propagation of the failure location is consistent with the observations from the Three Mile Island (TMI-2) post-accident inspection.

RCS/RPV Failure Modes

The RCS and RPV failure occur primarily by creep rupture according to MAAP4.07. The RPV lower head, hot leg, and steam generator tube material properties, including Larson-Miller creep rupture parameters, are entered by the user. Using the high temperature and stress within these primary system components, a failure by creep rupture is calculated.

A thermal radiation model for the reactor cavity was developed to approximate the energy transfer from the accumulated core debris to the reactor vessel outer wall, as well as to the vertical cylindrical segment of the reactor cavity. Prior to RPV failure, the vessel is covered with reflective insulation. When vessel failure occurs, it is assumed that the dynamics of the failure destroy the insulation. Therefore, when the core materials accumulate in the reactor cavity, the debris is assumed to radiate upward to the outer surfaces of the RPV lower plenum nodes and the cylindrical section of the reactor cavity.

The reactor cavity thermal radiation model becomes active when the RPV ruptures and a corium pool begins to form in the reactor cavity. Thermal radiation is transferred from the top surface of the corium pool to the RPV lower head based on the calculated temperatures of each surface and the view factor between them. After RPV lower head failure, the reactor cavity radiation model continues to calculate the exchange of energy between the material in the cavity and the remaining components in the RPV.

MCCI/Melt Conditioning in the Reactor Cavity

In MAAP4.07, several subroutines are combined to model the MCCI process in the reactor cavity and the spreading room. Molten core debris is considered to be in direct contact with the containment heat structures, allowing the heat transfer within these heat structures to be calculated by the original heat structure subroutine. MAAP4.07 has been enhanced with the new capability to calculate the heat transfer coefficients within the melt based on a Rayleigh number correlation, rather than by user input. Therefore, as the convection characteristics change during the course of MCCI, the heat transfer characteristics also change based on phenomenological evaluation. The debris within the reactor cavity is homogeneously mixed and debris crusts have the same composition as the molten debris. Steady-state relationships are used for heat transfer and level swell, and chemical reactions are treated by an equilibrium model.

MCCI in the Spreading Compartment

The MCCI in the spreading compartment is treated in the same way by MAAP4.07 as MCCI in the reactor cavity. MAAP4.07 treats the corium pools outside of the RPV as homogeneously mixed. The distribution of mass and energy between chemical species is tracked; however, there is no segregation of layers based on density.

Spreading Compartment Flooding and Basemat Cooling

The MAAP4.07 Generalized Containment Model (GCM) has all the capabilities necessary to model melt flooding and quenching. Compartments containing water pools can be modeled (i.e., IRWST) and connected to the rest of the containment through various methods. Each flow path in MAAP4.07 can represent a simple opening which is always open throughout a transient, or the flow path can be initially closed and fail open based on several different criteria: overpressure, heat sink ablation thickness, or at a user-specified time. The containment thermal-hydraulic routines assure the conservation of energy and momentum during transient flooding and quenching processes.

The MAAP4.07 has several new capabilities for modeling basemat cooling. Containment heat sinks may be described with heat slabs containing up to 40 variable-sized nodes for detailed modeling of the thermal profile. This new feature also allows modeling of a liner on each side of the containment heat slabs. These capabilities allow for modeling of composite walls that contain heat slabs of multiple material types and that are exposed to a high thermal gradient, as in the case of basemat cooling. The MAAP4.07 GCM is capable of modeling nodes with water throughput and heat transfer. Thus, cooling of molten corium from below can be appropriately modeled along with heat transfer and evaluation of potential for CHF. The MAAP4.07 heat sinks have also been made more flexible for modeling complicated heat transfer problems by making available a user-specified Kutateladze number for each side of containment heat sinks. This allows the user to have some control over the heat transfer correlation used to determine CHF during basemat cooling.

Hydrogen Recombination

The MAAP4.07 allows for the use of two different hydrogen recombination models. In addition to the original NIS correlation (as presented in Reference 6), MAAP4.07 includes the AREVA model based on experiments with AREVA PARs. The expanded MAAP4 PAR model also includes the new capability of defining multiple PARs of different sizes within each compartment.

Long-Term Heat Removal From The Containment

The MAAP4.07 Generalized Engineered Safeguards Features (GESF) model has all the capabilities to model long-term heat removal from the containment. The GESF model contains pumps, heat exchangers, a water source, a containment discharge location, and various user specified controls. There can be up to three separate containment spray trains modeled in MAAP4.07. The first two spray trains are dedicated to the upper and lower compartment sprays, respectively. However, the third train is more general and can be used to model a variety of different containment cooling systems.

Some additional code modifications made to model severe accident behavior in the U.S. EPR are as follows:

- Modification of the Generalized Containment Model (GCM) to permit various concrete properties to be specified for various containment nodes. Of particular interest in the U.S. EPR design is the need to differentiate between the concrete used for structures in

most of the nodes and the sacrificial concretes that are used in the reactor cavity region and the spreading room.

- A new heat sink subroutine was developed for calculating the heat transfer associated with containment structures. This subroutine includes variable-size nodes and the ability to represent a liner of any material type.
- A new model was implemented to include the potential for accelerated failure of the lower head as a result of corium discharge through an initial local failure site. As molten debris is discharged through the initial failure site, the surrounding wall will be ablated by the high temperature flow and the extent of this ablation will determine the number of azimuthal regions destroyed by the molten debris flow.
- The numbers of allowed containment nodes and junctions have been increased to 120 and 200, respectively.
- The core heatup model was modified to include the heavy reflector.
- A model was added to treat hybrid control rods (Ag-In-Cd and B_4C).
- The modeling of a break in the pressurizer has been generalized to allow the break anywhere in the pressurizer.
- Two simultaneous breaks from the pressurizer relief tank to the containment are now allowed.

3.4.2 MELTSPREAD-1.4 Description

MELTSPREAD is a transient, one-dimensional, implicit finite difference computer code used to predict the spreading behavior of high temperature melts flowing under the influence of gravity across horizontal surfaces submerged in a depth of water, or without the effects of water if the surface is initially dry. MELTSPREAD-1.4 contains several evolutionary code changes from MELTSPREAD-1.0 described in Reference 7. For current applications in nuclear reactor safety, the substrates specifically treated include steel and concrete, and the high temperature melts encompass a wide range of compositions of reactor core materials (corium) including a distinct oxide phase (predominantly UO2, ZrO2, steel oxides) plus a metallic phase (predominantly Zr, steel). The code requires inputs pertaining to the containment geometry and the melt “pour” conditions onto the containment floor and calculates the spreading and freezing behavior of the melt within the confines of the containment, including the following physical processes:

- heat transfer to overlying water and to the substrate
- heat-up/ablation of the substrate
- concrete decomposition
- heat transfer enhancement from gas sparging
- internal heat generation from both decay heat and chemical reactions between concrete decomposition products and melt constituents
- leading edge immobilization resulting from crust growth or bulk (slurry) freezing
- spreading of melt over previously spread/solidified material
- impingement heat flux where containment structures are contacted by the flowing melt

The code is used to predict conditions at the end of the transient spreading stage, including melt relocation distance, depth profile, substrate ablation profile, and transient wall heat-up. The code output provides input to other codes, such as MAAP4 for MCCI or structure response codes, to evaluate long-term behavior following the transient spreading stage. Alternatively, MELTSPREAD-1.4 contains simplified modules for quasi-steady MCCI and melt coolability calculations that may be used for long-term, quasi-steady solutions.

3.4.3 WALTER

WALTER is a computer program for the assessment of one-dimensional heat conduction problems. WALTER is used to calculate transient temperature-profiles in the sacrificial and protective layers within the reactor cavity as well as simulate the effect of basemat cooling in the spreading area.

Besides pure conduction, WALTER also considers melting/freezing processes. While for single-component systems such phase changes are assumed to occur at a certain melting temperature, multi-component systems are treated in a different way. Here a freezing range is considered spanning between input defined solidus and liquidus temperatures. To characterize the completeness of melting for each node, the liquid fraction is introduced as the ratio between liquid and total mass. The determination of the actual liquid fraction of a node is not part of the solution of the heat conduction equation but results from a post processing of the resulting temperature distribution.

Through the numerical scheme of the code, WALTER is adapted to the solution of heat conduction problems in solids. However, it can also account for natural convection in stratified, horizontal and volumetrically heated liquid layers. This ability allows its application to the EPR melt retention concept.

3.5 Validation of Analytical Tools

AREVA sponsored considerable model validation against relevant experiments to provide credibility in the results of the accident simulations. Considerable work was performed in validating the main integral code MAAP4.07 since it is used as the primary analysis tool. The main focus was in the area of MCCI. The bulk of code modifications was to these models, which is appropriate since debris retention in the cavity and coolability in the spreading compartment are key severe accident management strategies in the U.S. EPR.

The MAAP4.07 GCM was compared to the ACE, BETA, CCI-2 and CCI-3 molten debris coolability experiments as part of the MAAP4 dynamic benchmark feature. While this benchmark focuses on MCCI, its scope is not limited to the primary issue of ablation. Rather, this is an integral benchmark that can contribute to the validation of multiple models, including ablation, debris temperature, heat transfer to the overlying gas space and heat transfer to surrounding structural heat sinks in both the gas space and the debris space.

The benchmark calculations generally showed good agreement between models and experiments. One exception is for the CCI-3 test comparisons, which disagreed in ablation predictions by factors of 2-3. Specifically, sideward ablation was under-predicted and downward ablation was over-predicted. Moreover, the tests themselves exhibited large differences between axial and radial ablation rates. AREVA explained, in a response to an RAI, that the discrepancies are not entirely understood. They argued that the axial ablation predictions were within a factor of 2 of the data, and that the uncertainty has no negative consequences because of the presence of the protective zirconia layer in the cavity of the U.S. EPR would focus the ablation to primarily occur downward.

The results of the validation studies indicate a proper distribution of the input power to the upward (thermal radiation plus convection to gas) and lateral/axial (convection to concrete)

power losses. Thus, the selection of model parameters for corium convection is appropriate for these tests. The good agreement of the MAAP4.07 predictions for the ablation rate demonstrates that the MAAP4.07 MCCI model contains an adequate treatment of concrete thermo-physical data and chemistry. The good agreement of the MAAP4.07 predictions for the corium temperature shows that the MAAP4.07 MCCI subroutines correctly treat the heat transfer performance at the crust/concrete interface as well as the natural circulation conditions within the corium pool.

To validate MELTSPREAD-1.4 for analysis of the U.S. EPR, AREVA relies upon the studies described in Section 7.0 of Reference 7. An important part of the MELTSPREAD-1.4 development has been the comparison of model predictions with relevant analytical solutions and experiment data. These studies have been performed by both ANL and AREVA. Some of the validation studies completed to date that support the use of MELTSPREAD-1.4 as an analytical tool include comparisons of:

- Fluid mechanics algorithm with the analytical solution for a one-dimensional dam break problem. These results indicate that the correct fully-developed flow behavior is calculated to within a reasonable tolerance.
- Fluid mechanics algorithm with water simulant spreading test data in a 1/10 linear scale model of the Mark I containment. The results of the comparison of the predictions of the hydrodynamics model in MELTSPREAD-1.4 with the water simulant data demonstrate reasonable consistency between predicted and actual arrival times and transient depth profiles within the pedestal and drywell regions of a Mark I containment. Nonetheless, it is important to note that in these scenarios the heat transfer and freezing effects of the melt have been neglected.
- Integrated code predictions with the molten steel spreading test data in an approximate mockup of a Mark I containment. These results indicate that MELTSPREAD-1.4 can not only rely upon the validation database discussed in Reference 7 but that it also models melt spreading phenomena with reasonable accuracy. Nonetheless, it is important to note that while MELTSPREAD-1.4 reasonably predicts spreading phenomena, detailed comparison of analytical predictions to experimental spreading data for materials for which the code was principally developed (UO2/ZrO2/concrete oxides) has not been performed.

To validate WALTER for analysis of the U.S. EPR, AREVA has analyzed a number of physical scenarios and compared the code predictions with known analytical solutions. The conclusions drawn relative to WALTER for each of these validation case categories are provided below.

- *Stationary temperature profiles in solid layers.* Cases investigating steady-state heat transport through a composite wall and heat generation and transport within a volumetrically heated layer were performed. WALTER was able to accurately predict the steady-state temperature profiles within solid layers.
- *Transient temperature profiles in infinite solid layers.* A case investigating the transient temperature profile within an infinite solid layer was performed. Temperature profiles predicted by WALTER exactly matched the theoretical results.
- *Modeling of liquid layers and phase-change.* Cases investigating the freezing front into a liquid layer, ice formation on a top-cooled water pool heated from the bottom, convective heat transfer through a horizontal liquid layer, and volumetrically heat pools with both cooled and adiabatic upper and lower surfaces were performed. The phase change models were found to be fully symmetric in their calculation of freezing or

melting front. Calculation of the progression of the freezing front position and timing matches the analytical solution well. Calculation of convective heat transport through a horizontal liquid layer is well in range of the uncertainties given by the correlations used for a wide range of Rayleigh and Prandtl numbers. The predicted heat flux out of a volumetrically heated pool with cool upper and/or lower surface and the calculated up/down split match well with the predicted dependence upon Rayleigh number. The calculated pool temperatures, for both the heated and cooled cases, match the steady-state temperatures well. These results support the conclusion that WALTER is a stable numerical solver that can be used to solve transient one-dimensional heat transfer problems.

3.6 Calculation Matrix for Safety Resolution

An important step to take for establishing that the technical bases are adequate to meet the intent of the policies established in SECY 93-087 is to develop and execute a set of analyses that address all of the major issues. Such an attempt is made in the topical report, by defining a set of representative accident scenarios and simulating their behavior using MAAP 4.0.7, MELTSPREAD-1.4, and WALTER. The results of best-estimate analyses of three scenarios are presented in the report. These results, along with uncertainty analyses on CGCS performance and supplemental deterministic sensitivity studies that have yet to be provided to the staff, are asserted by AREVA to comprise a sufficient calculation matrix for safety issue resolution. In practice, however, the relevant severe accident scenarios will be identified based on reviewing Level 1 PRA results, supplemented with qualitative assessments of Level 2 PRA scenarios, as they relate to possible containment failure modes and accompanying releases of radioactive materials to the environment. The staff intends to review this further after the Level 1 PRA results are submitted.

In anticipation of what the PRA for the U.S. EPR will reveal, AREVA has identified a set of sixteen postulated initiating events, which includes the three scenarios analyzed in the topical report. These events are listed in Table 2.

Table 2. Postulated Initiating Events

Initiating Event (IE) Description
ATWS
Fire
Flood
Interfacing System LOCA
Loss of BOP
LOCA (Small, Medium, or Large)
Loss of Condenser Heat Sink
Loss of Component Cooling Water
Loss of Feedwater Flow
Loss of Instrument or Control Air
LOOP
Inadvertent Opening of a MSSV
Loss of One Emergency Bus
Steam Generator Tube Rupture
Steam Line Break (Inside and Outside Containment)
Turbine Trip

The strategy adopted by AREVA is to distinguish between relevant and bounding scenarios, based on the results of a Level 1 PRA, according to the following definitions:

> Relevant scenarios are defined as those having core damage frequency (CDF) greater than 10^{-8}/yr. These cases form the basis for the design and verification of the severe accident design measures. They are equally applied to all severe accident measures. The verification goal for this scenario class is to show that the severe accident measures function as designed and the leak-tightness and operability of the containment system is maintained.
>
> Bounding scenarios are exclusively employed to demonstrate the robustness of both individual and integral measures. These cases may have a CDF less than 10^{-8}/yr; however, they are selected based on processes and phenomena that also characterize one or more relevant scenarios.

For the justification of the mitigation features, these two kinds of scenarios are further defined as:

> The relevant scenarios are core melt scenarios with successful RCS depressurization before ultimate severe accident criteria (e.g., core outlet temperature 650 °C) which are bounding with respect to the severity, for the challenge under consideration, and which have a non-negligible frequency. They are used as a basis for the design measures

> The bounding scenarios are chosen from among the different core melt scenarios with passive reflooding via the accumulators, due to delayed RCS depressurization or active in-vessel reflooding: they are considered "bounding" when the worst time of the reflooding is considered to determine the most severe boundary conditions. These scenarios are used to demonstrate the robustness of the mitigation features, and to justify that the containment leak-tightness and integrity are maintained.

The results of the U.S. EPR Level 1 PRA are expected to reflect robustness of the design to severe accidents with the number of relevant scenarios limited to a small subset, such as the large-break LOCA (LBLOCA), small-break LOCA (SBLOCA), and station blackout (SBO) scenarios. Because of this expectation, these three scenarios were chosen for analysis in the topical report.

The staff believes that the overall approach of selecting a calculation matrix to facilitate safety issue resolution is sound, but that more than three scenarios need to be selected. Although the prescribed analyses are outside the scope of this topical report, plant behavior for scenarios where the RCS fails at high pressure must be considered in developing the calculation matrix, as should an assessment of containment bypass scenarios (such as from induced steam generator tube ruptures).

3.7 Sample Problem Analyses

In order to demonstrate expected severe accident performance of the U.S. EPR, a diverse spectrum of accident scenarios was selected. These scenarios include a LBLOCA, LOOP, and SBLOCA. Each of these scenarios represents a specific family of severe accident classes for which the bounding plant responses are investigated.

- **LBLOCA** – Due to maximized decay heat, the LBLOCA represents the limiting case with respect to melt stabilization and early containment pressurization.
- **LOOP** – A loss of all AC power results in a peak pressure transient within the primary system. Peak primary system pressure represents the limiting conditions for depressurization and well as the likely initiator of a SGTR. LOOP also represents the category of scenarios in which a minimum set of equipment is available and establishes the time window within which AC power must be restored to control containment pressure. The station blackout (SBO) scenario represents a bounding LOOP in which all backup power supplies are also unavailable.
- **SBLOCA** – Due to the high steam levels within the primary system combined with the slow nature of the accident evolution, the SBLOCA is the limiting case in terms of hydrogen production because the conditions for oxidation are maximized.

3.7.1 LBLOCA

The initiator for the LBLOCA is a complete severance of the pressurizer surge line followed by trip of the reactor and RCPs, isolation of main feedwater (MFW), start of the emergency feedwater (EFW) and a coincident failure of all four active trains of SI. Following injection of the accumulators, the water level in the core will reach a level below the top of the active fuel. Because the active SI is unavailable, the core will start to heat up and ultimately reach the severe accident pressurization valve setpoint, signaling the start of the severe accident.

The progression of this accident scenario and its associated severe accident phenomena are quantitatively described in the topical report. As would be expected, the RCS is calculated to depressurize very quickly, followed by core uncovery, cladding oxidation, and fuel melting in less than an hour. The MAAP 4.0.7 then predicts vessel failure some hours later, followed by core debris relocation to the spreading compartment after a holdup period in the cavity sufficient to allow a very high fraction of the core inventory to reach the cavity. Then, the debris was indeed cooled in the spreading compartment, at first by quenching the molten debris with water from the IRWST, and later by the SAHRS containment sprays. The pressure rises quickly during the initial quench, quickly decreases, and rises again until the SAHRS sprays are activated. The pressure then decreases to an acceptable level.

The staff checked the calculation by running the sequence using MAAP 4.0.7 and a parameter file and input deck provided by AREVA, albeit on a different computer platform and operating system than AREVA. The results were similar (but not identical) to those provided in the topical report. However, a number of code performance issues were raised, based on certain containment modeling details included in the U.S. EPR MAAP parameter file.

The staff noted that the plant model for the U.S. EPR, as implemented in the MAAP 4.0.7 parameter file, is very elaborate, with 27 containment regions and 188 junctions. Most other plant models have been far less elaborate. Because of the complexity, each sequence takes about two orders of magnitude longer to run than for similar sequences in other plants. This will compromise the ability to run many sequence variations when doing the PRA during the design certification phase. Investigation of the plot files in the large-break LOCA sequence reveals that the containment gases are well-mixed, and the hydrogen concentrations are close to each other and behave very similarly vs. time. The only differences are the spreading room and chimney, the reactor pit, and the cooling channel. Given this, the staff wrote an RAI asking for an explanation of why the containment is modeled with so many regions and junctions, and what information is being sought by doing this. The AREVA response agreed that a less-detailed model could suffice for many analyses, but a more detailed model is needed to track fission products to a sufficient level of detail to support level 3 PRA analyses.

The staff also noted that the time step was sometimes reduced to the minimum allowable value and remained there for significant numbers of time steps. AREVA was asked to describe the steps being taken to assure that the minimum allowable time step chosen allows for sufficient accuracy and stability. They responded that, for the most part, the minimum time step chosen for the topical report analyses was satisfactory. However, in calculations being prepared for the DCD, the time step had to be reduced to obtain the necessary code stability.

3.7.2 LOOP/SBO

The initiator for the SBO is a complete loss of all off-site sources of AC power (i.e., a LOOP), all four emergency diesel generators (EDG), and both SBO diesels. While this is an extremely remote event due to the number of AC power sources available for the U.S. EPR, it provides a bounding representation of the plant response. Given that all AC power sources are unavailable, only those features that can be powered off of DC power would be available (i.e., the dedicated severe accident valves would be operable but the SI systems and active SAHRS modes would not be). Following the initiator the reactor will trip and MFW and EFW would not be available. Following boil-off of the secondary inventory of the steam generators, the RCS temperature and pressure will start to increase until the point where the SRVs on the

pressurizer will start to cycle. RCS fluid will drain into the PRT until the point that the rupture disks fail on over pressure and release inventory into the containment. The continued loss of inventory into the containment will result in core uncovery and heat up until the severe accident depressurization valve set-point is reached, signaling the start of a severe accident.

The event progression is very similar to that seen in existing PWRs until the severe accident depressurization valves open. The RCS then depressurizes rapidly, causing a rapid increase in containment pressure to a level still below the containment design pressure. From then on, behavior is qualitatively similar to that of the LBLOCA accident described above.

The staff checked the calculation by running the sequence using MAAP 4.0.7 and a parameter file and input deck provided by AREVA, albeit on a different computer platform and operating system than AREVA. The results were similar (but not identical to) those provided in the topical report. In addition, the staff ran a variation where the severe accident depressurization valves were assumed to not function. It was also assumed that all steam generators depressurized after the first calculated opening of the steam generator safety valves (High/Dry/Low scenario). Steam generator tube creep rupture of a degraded tube was predicted for this case, well after the time when the severe accident depressurization valves would have been actuated. Nevertheless, the staff requires assurance that steam generator tubes would not fail during a severe accident and give rise to a containment bypass event.

3.7.3 SBLOCA

The initiator for the SBLOCA is a small break in the cold leg mid-loop followed by trip of the reactor and RCPs, isolation of MFW, start of EFW and a coincident failure of all four active trains of SI. Following the break, the RCS will slowly start to depressurize and empty the pressurizer. On the low pressurizer level signal, a partial cooldown via the MSRT will occur. The accumulator will inject providing inventory to cover the core and the heat from the primary system will be removed via the EFW system. Once the Condensate Storage Tanks (CST) inventory is depleted, core cooling is lost causing an increase in core temperature. Ultimately core uncovery is expected to occur resulting in a heat up until the severe accident depressurization valve setpoint is reaches signaling the start of a severe accident.

The event progression is very similar to that seen in existing PWRs until the severe accident depressurization valves open. The RCS then depressurizes rapidly, causing a rapid increase in containment pressure to a level still below the containment design pressure. From then on, behavior is qualitatively similar to that of the LBLOCA accident described above.

The staff did not check the AREVA calculations of this scenario.

3.7.4 Loss of BOP

This accident initiator is identified in Table 2, but was not analyzed in the topical report. Instead, AREVA sent an input file to the staff, as an example of an extremely unlikely, bounding case, where the severe accident depressurization valves may not function in a timely fashion to reduce the RCS pressure. For this scenario, loss of injection and loss of all feedwater are assumed. Pressurizer heaters and pressurizer sprays are also not available. Additionally, a significant delay in actuating the severe accident depressurization valves is assumed.

The results from running the input file supplied by AREVA were reviewed. During this review, certain thermal hydraulic phenomena not generally observed in high RCS pressure, dry steam generator cases (high/dry/high cases) played a dominant role, leading to prediction of induced creep rupture of steam generator tubes before hot leg creep rupture was predicted (the severe accident depressurization valve was not activated for this case). Specifically, unidirectional flow of steam and hydrogen through the loops is calculated to occur (following loop seal clearing), while no steam enters the bottom of the core. In addition, no upper plenum-to-steam generator, or steam generator inlet plenum to outlet plenum countercurrent flows are calculated to occur. A request for additional information (RAI) was written, asking AREVA to explain whether or not these phenomena are to be expected for such scenarios given the U.S. EPR design. A variation of this scenario was also suggested, where the secondary sides of all four steam generators would be depressurized when the steam generator safety valves are first opened shortly after S/G dryout (assume the valves stick open after the first demand). This is called a high/dry/low situation, which is a risk-dominant scenario in existing LWRs. Since it was not clear whether the calculation is realistic, a request was also made to identify any potential numerical problems (such as prolonged periods of time when the minimum allowable time step is taken).

In response, AREVA stated that, in the simulation, the reactor coolant pumps remain on until the void fraction at the pump exceeds 1.0. This, coupled with cycling of the pressurizer relief valves, leads to nearly complete depletion of the RCS inventory, and to onset of natural circulation of the steam. In reality, the RCPs would be tripped when the differential pressure drops to 80% of nominal. In this case, the loop seals would remain filled with water, leading to the countercurrent natural circulation flows through the steam generators and hot legs usually calculated to occur in high-pressure scenarios in existing reactors.

3.8 Separate Effects Analyses

The separate effect codes MELTSPREAD and WALTER were used to model certain severe accident phenomena important to the U.S. EPR design. These phenomena were modeled using separate effect codes to allow for a more detailed analysis of the phenomena than was offered by MAAP4. These phenomena include spreading of molten core debris, failure of the reactor cavity gate, and melt stabilization of the spreading compartment cooling structure.

Failure of the gate in the reactor cavity was modeled using the separate effects code WALTER. WALTER was used to predict the failure of the gate by determining if the 4 cm-thick aluminum gate is ablated rapidly when contacted by molten core debris. This analysis was performed in a limiting manner as opposed to modeling scenario specific timing. Therefore, the conclusions of WALTER hold for any scenario not just for the LBLOCA. Two limiting cases were run; one where the oxidic layer was resting on the aluminum plate and the other was performed with the metallic layer resting on the aluminum plate. In the case where the oxidic layer was atop the plate, the aluminum melt plug completely melts within 25 to 30 seconds after first contact with the melt. In the case where the metallic layer was atop the plate, the plug completely melts within 20 to 25 seconds after first contact with the melt.

MELTSPREAD was used to determine the degree of molten debris spreading. Based on the mass and physical conditions of the molten core debris within the reactor cavity at the time of gate failure, MELTSPREAD analysis show that completes spreading is expected within a few minutes, with only a minimal amount of concrete ablation.

The conclusions of MAAP4 relative to melt stabilization are corroborated by those predicted by WALTER. Like the WALTER analysis performed for gate performance, this analysis was also done in a bounding manner as opposed to on a scenario specific basis. Therefore, the conclusions of WALTER hold for any scenario not just for the LBLOCA. In this scenario, WALTER was used to predict the response of the surface of the cooling structure when contacted by molten core debris with passive flooding active. This limiting analysis predicted none of the cast iron cooling structure was ablated. After 12 hours of contact between the corium and the cast iron cooling structure, the interface temperature begins to decrease. Therefore, in this scenario, with the maximum temperature of the cast iron structure being 1256 K at 12 hours, there is nearly 200 K difference between the temperature and the melting point of the structure.

4.0 CONCLUSIONS

The staff finds that, with a few qualifications, the technical bases for severe accident assessment, including the testing programs and the models and methods used for severe accident analyses are adequate to meet the intent of the policies established in SECY 93-087 and to support the U.S. EPR PRA. These qualifications are:

- An analysis must be provided to show the plant response to an accident where the reactor coolant system remains at high pressure until vessel breach, in order to assess the impact on containment performance of a high pressure melt ejection/direct containment heating event.
- An analysis must be provided to assess the impact of an event that leads to induced steam generator tube ruptures, and consequent containment bypass.
- Reasonable assurance must be provided that the integrated analyses using MAAP 4.0.7 are carried out in a numerically stable manner.

The staff plans to carry out a number of confirmatory analyses during the DC phase. Some of these are:

- Hydrogen distribution in the containment.
- Vessel failure modes.
- Temporary core debris retention in the cavity.
- Long-term coolability of the debris in the spreading compartment.
- High pressure melt ejection/direct containment heating.

The staff plans to confirm AREVA's approach to assessing the consequences of in-vessel or ex-vessel steam explosions.

5.0 REFERENCES

1. ANP-10268P, Revision 0, "U.S. EPR Severe Accident Evaluation Topical Report," October 2006.
2. US NRC, SECY-93-087, "Evolutionary Light Water (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," issued April 2,1993, and the corresponding SRM, issued July 21, 1993.
3. Pilch, M. M., et al., "Resolution of the Direct Containment Heating Issue for All Westinghouse Plants With Large Dry Containments or Subatmospheric Containments," NUREG/CR-6338, SAND95-2381, 1996.

4. Schmitt, R. C. et al., “Simulated Design Basis Accident Tests of the Carolinas Virginia Tube Reactor Containment – Final Report. Idaho Nuclear Corporation Report IN-1403, 1970.
5. Hilliard, R. K., et al., “Removal of Iodine and Particles by Sprays in the Containment Systems Experiment,” Nuc. Tech. 10, pp. 499-519, April 1971.
6. Fauske and Associates, Inc., 1994a. MAAP4—Modular Accident Analysis Program for LWR Power Plants, vol. 2, Part 1: Code Structure and Theory, prepared for Electric Power Research Institute, May 1994.
7. EPRI TR-103413, “The MELTSPREAD-1 Computer Code for the Analysis of Transient Spreading and Cooling of High-Temperature Melts – Code Manual,” December 1993.
8. AREVA NP Inc. Technical Document, 38-9008026-000, “Description, Verification and Application of the Computer Code WALTER,” December 2005.

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The Staff's Disposition of AREVA's Comments on the Draft SE

1. Page 2, Section 2.0: A list summarizing SECY-93-087 general criteria is provided. The wording in the list is identical to that of the SECY with the exception of the "Equipment Survivability" statement, which copies AREVA's paraphrase of that section, and the addition of a bullet under "Containment Performance" that states, "The conditional containment failure probability (CCFP) should not exceed approximately 0.1." This is not identified in the SECY-93-087 as a "general criterion;" rather the SECY-93-087 specifically states:

   > The staff concluded that the following general criterion would be an appropriate substitute for a CCFP in evaluating evolutionary ALWR containment performance during a severe-accident challenge:
   >
   > > The containment should maintain its role as a reliable, leak-tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage, and that following this 24-hour period the containment should continue to provide a barrier against the uncontrolled release of fission products."

   In fact, SECY-93-087 specifically states:

   > In its SRM of June 26, 1990, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for the evolutionary ALWRs. The Commission directed, however, that this objective should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention.

   As such, an accurate representation of the SECY should exclude the general criterion statement of CCFP < 0.1.

   AREVA believes that the issues of core damage frequency (CDF) and large release frequency (LRF), along with the related issue of CCFP, are outside the scope of the topical, since they are related primarily to plant design and the PRA, not the severe accident analysis methodology. Accordingly, we believe that statements in the SER that refer to these parameters for the U.S. EPR design should be removed from the staff's evaluation. If the staff declines to do so, the following additional comment is relevant to the previous discussion.

   The statement "should not exceed 0.1" that appears on page 4 in Section 2.0 should be removed, and the statement, "the staff must be convinced that the approach described in the topical report will lead to reductions in core damage frequency and large release frequency, relative to current generation reactors, while maintaining the conditional containment failure probability below 0.1," should exclude the last clause related to CCFP. We also believe that this statement, if retained, should refer to "reasonable assurance" with regard to the "approach described in the topical report," rather than the need to "convince" the staff.

   <u>DISPOSITION</u>
   Based on the following excerpt from the Commission's SRM for SECY 93-087 (dated

July 21, 1993), the Staff has continued to require adherence to a CCFP of 0.1 or less for future ALWRs:

> The recommendations on containment performance, as outlined in SECY 93-087, could be read to imply that the staff is no longer proposing to use the concept of conditional containment failure probabilities (CCFP). However, based on discussions held during the Commission meeting on this subject, the staff informed the Commission that it intends to continue to apply the 0.1 CCFP in implementing the Commission's defense in depth regulatory philosophy and the Commission's policy on Safety Goals.
>
> Therefore, the Commission approves the staff's position to use the following deterministic containment performance goal in the evaluation of the passive ALWRs as a complement to the CCFP approach approved by the Commission in its SRM of June 26, 1990:
>
>> The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containments stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for 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.
>
> The Commission approves the staff's interim approach subject to the staff's review and recommendations resulting from public comments on the "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future ALWRs.

Accordingly, Section 19.2.4 of NUREG-0800 states that there is a “probabilistic goal that the containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA.”

Thus, the SER wording will stand as written.

2. Page 8, Section 3.1.2.3: The SE states, “Between the RPV and cavity is a layer of insulation limiting heat loss from the vessel and four walls aligned radially from the melt plug effectively creating four 90° azimuthal sectors.” AREVA’s design now includes 6 walls instead of 4. It is recommended that the text be revised without explicitly specifying the number of walls.

   DISPOSITION
   The text is revised to read: “Between the RPV and cavity is a layer of insulation limiting heat loss from the vessel and six walls aligned radially from the melt plug effectively creating six azimuthal sectors.”

3. Page 10, Section 3.1.2.4: With the August 29, 2007 letter submitting AREVA’s responses to the NRC’s second RAI, we identified a change from a two-train SAHRS to a one-train SAHRS. This section of the SER does not acknowledge this update; however, a reference

to SAHRS appearing in Section 2.0 does correctly state, "[t]he SAHRS train is located in a dedicated, radiologically-controlled room within one of the four plant safeguard buildings."

Section 3.1.2.4 should be rewritten to correctly reflect the single train design.

DISPOSITION
The text is revised to read: To ensure substantial margin in containment pressure control the SAHRS has four primary modes of operation, each playing a role in controlling the environmental conditions within the containment so that its fission product retention function is maintained."

4. Page 13, Section 3.2: Our approach to containment performance evaluation has evolved to one centered on best-estimate analysis plus uncertainty complemented by deterministic analyses with bounding assumptions. This is in contrast to paragraph 3 of this section where it states, "Wherever possible, AREVA plans to apply deterministic analyses ... As needed, best-estimate analysis…." It would be more accurate to state that "AREVA plans to apply both best-estimate and deterministic methods to demonstrate acceptable containment response. For best-estimate analyses, phenomenological insights gained from experiments and/or analytical models would be convolved into the task of quantifying uncertainty."

   DISPOSITION
   The text is revised to read: AREVA's approach to containment performance evaluation has evolved to one centered on best-estimate analysis plus uncertainty complemented by deterministic analyses with bounding assumptions.

5. Page 13, Section 3.2: In assessing the assumptions established to identify important severe accident phenomena, the SER states, "It should be noted, however, that there would still be some likelihood of severe accidents where the RCS pressure would remain high. Thus, some provision must be made to manage these accident scenarios as well." It is stated in Section 4.2.1 of the Topical Report, "The rapid depressurization of the RCS removes a degree of uncertainty associated with postulated scenarios since many such events become very similar to a large-break LOCA (LBLOCA). Therefore, for purposes of identifying important processes and phenomena challenging the U.S. EPR severe accident response strategy, a hypothetical phenomenologically-bounding severe accident for challenging the U.S. EPR melt retention measures is an initiating large primary system pipe rupture followed by scram and subsequent accumulator injection."

   A primary depressurization system (PDS), comprising two redundant valves, is included in the U.S. EPR design to specifically preclude high pressure melt scenarios. It also specifically addresses one of the two general criteria appearing in SECY-93-087 that are associated with high pressure melt ejection. The SER statement appears to ignore the role of the PDS in resolving this concern.

   As an acknowledgement of the importance of High Pressure Melt Ejection (HPME), HPME was retained as a Safety Issue in Table 4-2 of the Topical Report, appearing in items 4 and 8.

   It is recommended that the statement be revised to read, "It should be noted, however, that there may be a significant uncertainty associated with RCS pressure at the moment of RPV

failure. Some provision to address this uncertainty should be made with regard to the HPME issue."

DISPOSITION
The text is revised to read: "It should be noted, however, that there may be a significant uncertainty associated with RCS pressure at the moment of RPV failure. Some provision to address this uncertainty should be made with regard to the HPME issue and to the issue of thermally-induced steam generator tube rupture."

6. Page 14, Section 3.3.1: The topical makes the statement that "[t]o maintain containment integrity… the following must safely be avoided: Any deflagration with AICC pressure above design pressure…" In fact, the industry benchmark with this criterion is relative to the containment's ultimate capacity pressure. We request that this section of the SER be revised to reflect this benchmark.

DISPOSITION
The statement quoted above is made by AREVA in the topical report and repeated in the SER. However, AREVA also states above that the criterion is actually relative to the ultimate containment capacity. On this basis, the SER wording is changed to: "To maintain containment integrity, in particular in the early phase of an accident, the following must safely be avoided:

- Any deflagration with Adiabatic Isochoric Complete Combustion (AICC) pressure above containment ultimate pressure
- Fast deflagration with the potential to initiate a detonation
- Local temperatures that pose a threat to the containment shell."

7. Page 15, Section 3.3.1; page 16, Section 3.3.2.1; page 18, Section 3.3.2.2; page 18, Section 3.3.2.3; page 19, Section 3.3.3; page 20, Section 3.3.4; and page 34, Section 4.0: Specific statements about the staff's future plans (MELCOR and TEXAS analysis) do not seem appropriate, particularly insofar as the current capability of those codes to model the U.S. EPR's severe accident mitigation design features is open to question. Rather, it is recommended that statements be more general, such as, "Analyses referencing this Topical Report may be subject to independent confirmation."

DISPOSITION
The text is revised to read:
on page 15, "The staff generally agrees, but plans to independently calculate hydrogen distribution."
on page 16, "The staff will do independent calculations to confirm this hypothesis."
on page 18, "The staff will carry out independent calculations to confirm that most of the core debris in the cavity would flushed into the spreading compartment."
on page 19, "In parallel, the staff plans to use MELCOR to confirm the adequacy of the severe accident depressurization system, and to determine the pressure increase from HPME/DCH."
on page 20, "The staff plans to confirm the adequacy of AREVA's approach by performing independent calculations."
on page 34, "The staff plans to carry out a number of confirmatory analyses during the DC phase. Some of these are:

- Hydrogen distribution in the containment.
- Vessel failure modes.
- Temporary core debris retention in the cavity.
- Long-term coolability of the debris in the spreading compartment.
- High pressure melt ejection/direct containment heating.

The staff also plans to confirm AREVA's approach to assessing the consequences of in-vessel or ex-vessel steam explosions."

8. Page 19, Section 3.3.3: The statement with regard to HPME analysis that "[t]he staff would concur once a confirmatory integrated analysis is provided to show that, under HPME/DCH conditions, the calculated peak pressure in the containment is well below the ultimate failure pressure," is too prescriptive. The low frequency, potential high consequence, and large phenomenological uncertainties of HPME make analysis of HPME and subsequent DCH a probabilistic exercise, and not an issue related explicitly to the severe accident analysis methodology. Moreover, we do not believe that the staff has any regulatory basis to require analysis of a specific severe accident sequence absent a PRA analysis showing that the scenario represents a risk-significant sequence.

   It is recommended that the statement be rephrased as, "Containment performance analyses referencing this Topical Report are expected to include an analysis demonstrating that probability of containment failure from DCH is extremely low…"

   DISPOSITION
   The SER wording will stand as written.

9. Page 20, Section 3.3.4: The Containment Spray Experiments indicated as having been carried out at "Hanford National Laboratory" were actually performed at Brookhaven National Laboratory.

   DISPOSITION
   The Containment Systems Experiment (CSE) was a facility at Battelle Northwest Laboratory (BNWL) in Richland Washington. The experiments on iodine and aerosol removal by sprays were performed there by Bob Hilliard and his colleagues, and the results were published in BNWL-1244 in 1970 and in Nuclear Technology in 1971. The tests <u>were not</u> performed at Brookhaven National Laboratory.

   The SER is modified to read: "The basis for these installed units draws on test program experience including Carolinas-Virginia Tube Reactor Containment (Reference 4) and Battelle Northwest Laboratory's Containment Spray Experiments (Reference 5)." It is suggested that the topical report be changed as well.

10. Page 21, Figure 2, "Overview of U.S. EPR Severe Accident Analytical Methodology," was updated with the replacement document pages submitted August 29, 2007 along with the 2nd set of RAI responses. Please use the updated figure.

    DISPOSITION
    The revised figure was identical to the original figure. The Figure remains unchanged.

11. Page 26, Section 3.5: The first sentence of this section should be revised by changing the words "carried out" to "sponsored" to be more accurate and acknowledge that AREVA did not perform all of this work in-house.

    DISPOSITION
    The text is revised to read: "AREVA sponsored considerable model validation against relevant experiments to provide credibility in the results of the accident simulations."

12. Page 28, Section 3.6: Near the end of the first paragraph the SE states, "[i]t remains to be seen whether the three accident scenarios chosen indeed address all of the issues." AREVA objective in including the limited set of analyses in the Topical Report was not to imply that they would resolve all severe accident issues. Rather, they were provided as sample problems, as indicated by the title of Section 7.0 ("Sample Problem Analyses"). The statement can be interpreted as meaning that these cases were represented as something more than simply sample problems. Given that this was clearly not the case, it is recommended that this statement be omitted.

    DISPOSITION
    The text is revised to remove the sentence in question.

13. Page 30, Section 3.6: The statement, "plant behavior for scenarios where the RCS fails at high pressure must be considered, as should an assessment of containment bypass scenarios" ignores the focus on relevant or more likely scenarios. While these analyses are important for deriving the U.S. EPR PRA, the prescribed analyses are outside the scope of this topical report.

    It is recommended that the statement be revised to read, "The staff believes that the overall approach of selecting a calculation matrix to facilitate safety issue resolution is sound for examination of the relevant or more likely scenarios, but that more than three scenarios need to be selected. Exclusion of certain important low frequency, high consequence scenarios, such as RCS ruptures at high pressure and containment bypass, from the class of relevant or more likely scenarios should be justified."

    DISPOSITION
    The SER wording is changed to read: "Although the prescribed analyses are outside the scope of this topical report, plant behavior for scenarios where the RCS fails at high pressure must be considered in developing the calculation matrix, as should an assessment of containment bypass scenarios (such as from induced steam generator tube ruptures)."

14. Page 31, Section 3.7.1 (and p. 32, Section 3.7.2): The SE states, "The staff checked the calculation by running the sequence using MAAP 4.0.7 and a parameter file and input deck provided by AREVA." It is recommended that the text be revised to indicate that the staff ran MAAP 4.0.7 on a different computer platform and operating system than AREVA, which may account for the "similar (but not identical)" results obtained by the staff.

    DISPOSITION
    The text is revised in both sections to read: "The staff checked the calculation by running the sequence using MAAP 4.0.7 and a parameter file and input deck provided by AREVA, albeit on a different computer platform and operating system than AREVA."

15. Page 34, Section 4.0: The qualification stating that "[r]easonable assurance must be provided that the integrated analyses using MAAP 4.0.7 are carried out in a numerically stable manner" is open-ended with regard to the U.S. EPR Design Control Document submittal. Inclusion of such a numerical study is unprecedented for a DCD; if, as a result of the DCD review, the staff has concerns about the code's stability, AREVA would expect to receive an RAI in that regard. Therefore it is recommended that the clause "upon request" be included to reflect this approach.

    DISPOSITION
    The SER wording will stand as written.

16. Page 34, Section 4.0: Per the previous comments, the following revision is recommended,

    The staff finds that, with a few qualifications, the technical bases for severe accident assessment, including the testing programs and the models and methods used for severe accident analyses, are adequate to meet the intent of the policies established in SECY-93-087 and to support the U.S. EPR PRA. These qualifications are:

    - Containment performance analyses referencing this Topical Report are expected to include an analysis demonstrating that the probability of containment failure from high-pressure melt ejection/direct containment heating is extremely low.
    - Exclusion of certain important low frequency, high consequence scenarios, such as RCS ruptures at high pressure and containment bypass, from the class of relevant or more likely scenarios should be justified.
    - Reasonable assurance must be provided, upon request, to demonstrate that the integrated analyses using MAAP 4.0.7 are carried out in a numerically stable manner.

    Analyses referencing this Topical Report, expected for the U.S. EPR Design Certification, may be subject to independent confirmation. Potential areas for further study are:

    - Hydrogen distribution in the containment.
    - Vessel failure modes.
    - Temporary core debris retention in the cavity.
    - Long-term coolability of the debris in the spreading compartment.
    - High-pressure melt ejection/direct containment heating.
    - In-vessel and ex-vessel fuel-coolant interactions

    DISPOSITION
    The SER wording will stand as written.