Development of Technical Bases for Severe Accident Management in New Reactors

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Introduction

The NRC has taken an active role to ensure that utilities adopt acceptable management (AM) practices. In January, 1989, the Staff issued SECY 89-012, "Staff Plans for Accident Management Regulatory and Research Programs,' discussing essential elements of a utility AM plan and offering an approach for its implementation. Subsequently, the NRC worked with the industry to define the scope and attributes of a utility AM plan and to develop guidelines for plant-specific implementation. The result was Section 5 of NEI 91-04, Revision 1, ("Severe Accident Closure Guidelines"), which lays out the elements of the industry's severe accident management (SAM) closure actions that have been accepted by the NRC staff.

The AM programs are based on a technical basis for systematically evaluating and enhancing the ability to deal with potential severe accidents. Vendor-specific AM guidelines were developed for use by individual utilities in establishing plant-specific procedures and guidance. From these, guidance and material to support utility activities related to training in severe accident prevention and mitigation was developed. From these, each utility has prepared and implemented plant-specific AM plans.

The regulatory basis for existing plants is described in NEI 91-04 (originally NUMARC 91-04), "Severe Accident Issue Closure Guidelines," which includes a summary of the important information pertaining to the agreement between the nuclear industry and the NRC on severe accident management, and contains the binding implementing guidance. This industry initiative has been endorsed by the NRC. The current industry technical basis stems from EPRI's "Severe Accident Management Technical Basis Report" (EPRI TR-101869), which was used in developing vendor-specific guidance. This guidance was provided to the NRC by the various owners groups and constitutes the technical basis of severe accident management for existing plants.

Although new reactor designs are to have enhanced capabilities for preventing and mitigating severe accidents, AM remains an important element of defense-in-depth for these designs. However, the increased attention on accident prevention and mitigation can be expected to alter the scope and focus of AM relative to that for operating reactors. For example, increased attention on accident prevention and the development of error-tolerant designs can be expected to decrease the need for operator intervention somewhat, while increasing the time available for such actions if necessary. This permits a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

For both operating and advanced reactors, the overall responsibility for AM, including development, implementation, and maintenance of the AM plan, lies with the nuclear utility. However, the vendors' guidance has continued to serve as the technical basis for SAM procedures and for training utility personnel in carrying them out. Computational aids for technical support have been developed, information needed to respond to a spectrum of severe accidents has been provided, decision-making responsibilities have been delineated, and utility self-evaluation methodologies have been developed. The NRC's Office of New Reactors (NRO) staff expects that this approach will be adopted by the applicants for new reactor licenses as well. Accordingly, the applications for design certifications are being reviewed in such a manner as to ensure that the technical basis for AM will be provided by the vendors for each design.

As stated in NEI 91-04, AM consists of those actions taken during the course of an accident by the Emergency Response Organization (ERO), including plant operations, technical support and plant management staff, in order to:

- Prevent the accident from progressing to core damage;
- Terminate core damage progression once it begins;
- Maintain the capability of the containment as long as possible; and
- Minimize on-site and off-site releases and their effects.

The latter three actions constitute a subset of accident management referred to as severe accident management (SAM). They provide guidance when Emergency Operating Procedures (EOPs) are no longer effective. For existing plants the approach is to make full use of existing plant capabilities, including standard and non-standard uses of plant systems and equipment.

Severe accident management review for new reactors

During the process of reviewing design certification applications, the NRC staff is requesting the applicants to provide the revised technical bases for the new plants, to ensure that the new features for accident prevention and mitigation are included. The NRC staff is not asking for specific severe accident management guidelines. Instead, it needs to evaluate the technical bases that would support the guidelines.

Upon completion of a design certification review, a Safety Evaluation Report (SER) will be issued and, once the design certification is granted, a utility can obtain a combined license (COL) to build and operate such a plant. Before operation can commence, the NRC must approve the AM procedures. Thus, it is important for the NRO staff to accept the vendors' description of the technical basis. Before a COL is issued, an applicant needs to demonstrate that it understands the new technical basis and has established a framework for incorporating it into the AM procedures for its new plant. After a COL is issued, it is imperative that the new plants also have their SAMG implementation in place, including procedures and training, prior to initial fuel load.

Insights regarding severe accident mitigation features in new reactors

The new reactor designs all include features that increase the capability for mitigating severe accidents. These address the concerns expressed by the NRC staff in SECY-90-016 and SECY-93-087, and the associated Staff Requirements Memoranda (SRM) issued by the Commission that describe the new requirements that must be met. Noteworthy among the issues addressed are: hydrogen control; core debris coolability; high-pressure core melt ejection; containment performance (including the possible effects of molten core/coolant interactions); containment bypass, including from steam generator tube ruptures; and equipment survivability. The increased attention on accident prevention and mitigation for new reactors can be expected to alter the scope and focus of AM relative to that for operating reactors, while remaining an important element of defense-in-depth. For example, increased attention on accident prevention and the development of error-tolerant designs can be expected to decrease the need for operator intervention somewhat, while increasing the time available for such actions if necessary. This permits a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

Since the new designs include safety enhancements not present in existing plants, a major focus of the NRO staff review is on the roles new severe accident mitigation features would play in AM strategies. The new-plant vendors have all used insights derived from their probabilistic risk assessments (PRA), and are using the MAAP4 code to simulate accident progression for the more-likely severe accident scenarios, as bases for modifying their existing AM guidance. In some cases, different strategies from those used for existing reactors must be adopted. For example, the ESBWR design includes a device (the BiMAC) that is intended to arrest core melt progression in the lower drywell by cooling the debris both from above and below. For all currently operated LWRs, severe accident management requires that, provided a sufficient floor area available for spreading and a sufficient amount of water to cover the molten core debris, the debris will become quenched and will remain coolable thereafter. While the ESBWR satisfies the basic conditions for this approach, the core-on-the-floor approach is further improved, it is also necessary to ensure that a large ex-vessel steam explosion would not occur immediately after vessel breach. To prevent this, the lower drywell (LDW) needs to be kept dry until after the debris enters. Consequently, the vendor is recommending that the strategy for flooding containment currently in place for the existing boiling water reactors in the United States be modified for ESBWR plants. A similar argument can be made for ABWR plants.

The PWR vendors are also modifying their strategies related to depressurizing the primary system, due to the incorporation of severe accident-related depressurization valves into their designs. Such valves would reduce the risk from induced steam generator tube ruptures in high-pressure scenarios, as well as greatly mitigate the consequences of high-pressure core melt ejection.

A discussion is presented below of some of the more important mitigation features in designs for which COLA applications have been filed. Insights pertinent to severe accident management are also discussed.

AP1000 design. The AP1000 design has already been certified by the NRC and a number of COL applications are currently being reviewed. A design certification amendment is also under review.

External reactor vessel cooling (ERVC). The AP1000 design incorporates ERVC as a strategy for retaining molten core debris in-vessel in severe accidents. The objective of ERVC is to remove sufficient heat from the vessel exterior surface so that the thermal and structural loads on the vessel (from the core debris which has relocated to the lower head) do not lead to failure of the vessel. By maintaining RPV integrity, the potential for large releases due to ex-vessel severe accident phenomena, i.e., ex-vessel fuel/coolant interactions (FCIs) and core debris/concrete interactions (CCI) is eliminated.

The AP1000 design includes several features that enhance ERVC relative to operating plants, specifically:

- safety-grade systems to provide automatic or manual RCS depressurization and reactor cavity flooding;
- a "clean" lower head that has no penetrations; and
- an RPV thermal insulation system to limit thermal losses during normal operations, while providing an engineered pathway to supply water for cooling the vessel, and to vent steam from the reactor cavity, during severe accidents.

The AP1000 Level 2 PRA estimates that the ERVC would enable the majority (~97 percent) of core melt accidents (that do not involve containment bypass or containment isolation) to be arrested in the vessel. Depressurization of the RCS and reactor cavity flooding contribute to the success of ERVC.

Combustible gas control. The containment hydrogen control system serves the following functions:

- hydrogen concentration monitoring;
- hydrogen control during and following degraded core or core melt scenarios (provided by hydrogen igniters).

In addition, non-safety-related passive autocatalytic recombiners (PARs) are provided for defense-in-depth protection against the buildup of hydrogen following a LOCA. The hydrogen ignition subsystem meets the requirements of 10 CFR 50.44 for future water-cooled reactors, whereby the design must limit hydrogen concentrations in containment from a release of a 100% fuel clad-steam reaction to less than 10% by volume, and maintain containment structural integrity and appropriate accident-mitigating features. This requirement was promulgated to address the lessons learned from the TMI accident.

The hydrogen ignition subsystem is designed to promote hydrogen burning soon after the lower flammability limit is reached in the vicinity of an igniter, and prevent the concentration from reaching 10%. This would provide confidence that containment structural integrity can be maintained during hydrogen burns.

Core debris coolability. The AP1000 design relies primarily on safety grade RCS depressurization and reactor cavity flooding capabilities to prevent RPV breach and CCI, but also incorporates plant features consistent with the guidance in SECY-93-087 regarding debris coolability. In the unlikely event of RPV failure, these features would reduce the potential for containment failure from CCI. The AP1000 design features include the following items:

- a cavity floor area and sump curb to allow debris spreading without debris ingression into the reactor cavity sump;
- a manually-actuated reactor cavity flooding system to cover core debris with water and maintain long-term coolability;
- a 0.85 m (2.8 ft) thick layer of concrete to protect the embedded containment shell, with an additional 1.8 m (6 ft) thick concrete layer below the liner elevation.

The applicant calculated that adequate reactor cavity flooding is achieved in about 98 percent of the sequences identified in the AP1000 PRA. About half of the core damage events require operator actuation of the cavity flooding system to ensure successful cavity flooding, but the remaining half would adequately flood as a direct consequence of the accident progression, even without manual actions.

High-pressure core melt ejection (HPME). Two features have been incorporated into the AP1000 design to prevent and mitigate the effects of HPME (including direct containment heating (DCH)), specifically, the automatic depressurization system (ADS) and reactor cavity design features.

The ADS is one of the major features of the AP1000 design. It is an automatically-actuated, safety-grade system with four different valve stages that open sequentially to reduce RCS pressure sufficiently to ensure long-term cooling. If automatic actuation fails, then operator action from the main control room could initiate depressurization. The ADS valves are designed to remain open, thereby preventing repressurization of the RCS. It is estimated in the PRA that sufficient depressurization is achieved in about 95 percent of the core melt sequences.

The design of the reactor cavity is such that most of the ejected core debris would not reach the upper containment. The pathways for debris transport from the cavity include the following:

- annular openings between the coolant loops and the biological shield wall leading to the SG compartments;
- the area around the reactor vessel flange leading directly to the upper compartment; and
- a ventilation shaft leading to the SG compartments.

Each pathway is such that the debris particles would change direction and encounter obstacles before reaching the upper containment region. It should be noted, however, that deposition of core debris and aerosols in pump rooms and steam generator compartments is possible, and could affect accident management strategies.

ESBWR design. The ESBWR design is under review has not yet been certified by the NRC. However, a number of COL applications are currently being reviewed.

Combustible gas control. Just as for existing BWRs, the ESBWR containment would be inerted during full-power operation. Consequently, insufficient oxygen would be present to cause a hydrogen burn during a severe accident.

Results from the applicant's MAAP 4.0.6 simulations show that the time required for the oxygen concentration to increase to the de-inerting value of 5 percent is significantly greater than 24 hours for a wide range of fuel cladding-steam interaction and iodine release assumptions. Accumulation of combustible gases that may develop in the period after about 24 hours would be managed by implementing the severe accident management guidelines. Risk and safety of operations when the containment is not inerted (e.g., when the containment is open during shutdown) will need to be considered, because combustible gas generation can occur in locations with oxygen present at combustion-supporting levels.

Core debris coolability and fuel-coolant interactions. Two design features, the Gravity-Driven Cooling System (GDCS) and the Basemat Internal Melt Arrest and Coolability device (BiMAC), act to prevent significant ablation of the concrete in the lower drywell (LDW). The deluge mode of GDCS operation provides flow to flood the LDW when the temperature in the LDW increases enough to be indicative of RPV failure and core debris in the LDW. The GDCS pools supply water to the BiMAC device via squib valves that are activated on the deluge lines. Flooding the LDW after the introduction of core material minimizes the potential for energetic FCI at RPV failure. Covering core debris with water provides scrubbing of fission products released from the debris and cools the corium, thus limiting potential CCI. Failure of the squib valves to function properly would cause an order-of-magnitude increase in the large release frequency (LRF) for the ESBWR, but the total LRF would be well below the goal of 1.0×10^{-6} per reactor-year.

The BiMAC gives additional assurance of debris bed cooling by providing an engineered pathway for water flow through the debris bed. It is a passively cooled barrier to core debris on the LDW floor. The design features a series of side-by-side inclined pipes, forming a jacket, which is passively cooled by natural circulation when subjected to thermal loading. Water from the GDCS pools enters the pipes via connecting downcomers. Once the pipes fill up, the debris is also cooled from above from water that flows out of them. The timing and flows are such that cooling becomes available immediately upon actuation. Timely flooding of the LDW and a properly-functioning BiMAC device would make the issue of corium-concrete interactions inconsequential. Moreover, the design procedure of not immediately adding water greatly reduces the probability of a highly energetic steam explosion.

High-pressure core melt ejection. The probability of a HPME is significantly reduced by the highly reliable ADS. In addition, the following ESBWR containment design features mitigate the possible effects of high-pressure core melt:

- The containment is segregated into an upper drywell (UDW) and LDW that communicate directly, but the design mitigates the ability of high-pressure core melt, ejected within the lower drywell, to reach the UDW.
- The UDW atmosphere can vent into the wetwell through a large vent area, making it virtually impossible to overpressurize the drywell volume.
- The containment steel liner is structurally backed by reinforced concrete, which cannot be structurally challenged by DCH.

The upper drywell head is immersed in a water pool during normal operation. Consequently, analyses have shown that thermally-induced failure of the upper drywell head and its seals would be physically unreasonable. Moreover, bounding calculations have shown that upper drywell liner temperatures would be considerably below its melting point. However, the calculations also show short periods of potentially very high temperatures in the LDW atmosphere (up to 4000 °K) under some highly unlikely conditions. These temperatures, and the presence of potentially large quantities of melt in the LDW, indicate that the LDW liner could be subject to local failures.

Containment performance. Because of the passive cooling function in the ESBWR containment, the vacuum breakers between the wetwell and UDW are designed be essentially leak-proof, to prevent the possibility of containment bypass during a severe accident. Three vacuum breakers are installed in the diaphragm floor, to limit the magnitude of a negative pressure differential between the drywell and the wetwell. They operate passively in response to a negative drywell-to-suppression pool pressure gradient and are held closed by a combination of gravity and the normally positive pressure gradient.

Four position sensors are located around the disk periphery of the primary vacuum breakers to confirm to the plant operator that the disks are securely seated. The analysis in the ESBWR PRA assumes that the position switch that provides annunciation in the control room can sense a very small gap between the disk and the seating surface.

Each vacuum breaker is equipped with a diverse, redundant, passive, process-actuated check-type isolation valve, which provides isolation capability if the vacuum breaker sticks open or leaks in its closed position. The isolation valve is normally in the closed position and, like the vacuum breaker itself, is process-actuated by differential pressure between the structure and component (SC) and drywell. In this manner, the isolation valve is more like a redundant vacuum breaker than an isolation valve, and both valves would have to leak simultaneously to create a leakage path from the SC to the drywell. Including the isolation valves significantly decreases the large release frequency.

U.S. EPR design. The U.S. EPR design is under review has not yet been certified by the NRC. However, a number of COL applications are currently being reviewed.

Combustible gas control. The containment has a dedicated combustible gas control system (CGCS) to avoid containment failure from rapid deflagration or from accidental ignition of a critical gas mixture. The CGCS system is divided into two subsystems: the Hydrogen Reduction System (HRS) and the Hydrogen Mixing and Distribution System. The HRS consists of both large and small passive autocatalytic recombiners (PAR) installed in various parts of the containment. In the presence of oxygen, the PARs would automatically start if the threshold hydrogen concentration is reached at the catalytic surfaces.

The PAR locations and arrangement inside the equipment rooms and containment dome are such that they support global circulation within the containment, and thereby homogenize the atmosphere and reduce locally high hydrogen concentrations to below 10 percent by volume during various phases of accidents 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 would 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.

The hydrogen concentration and its distribution within various compartments of the containment are continuously monitored, and information would be available to the main control room.

Results of 59 uncertainty analysis simulations carried out by the applicant show that the global hydrogen concentration did not reach or exceed 10 percent by volume for any one of the scenarios, due to the effectiveness of the PARs.

Core debris coolability and containment performance. The Core Melt Stabilization System (CMSS) and the Severe Accident Heat Removal System (SAHRS) are features designed to ensure core debris coolability. The CMSS would stabilize core debris exiting the RPV before it could challenge containment integrity. Initial stabilization would take place in the reactor cavity, where temporary retention is achieved by a layer of sacrificial siliceous concrete that must be penetrated by the melt before it can escape from the cavity by failing a melt plug. This delay would provide a means for allowing practically the entire molten core inventory to be collected in the cavity. The sacrificial layer is backed with a protective refractory material that has a low thermal conductivity and a mechanical strength greater than concrete, to confine the melt and insulate the RPV support structure. The protective layer "guides" heat transfer from the melt toward the melt plug. Once the melt plug fails, the melt would flow down a discharge channel into a spreading compartment, which consists of a large horizontal concrete surface over which the molten core debris can be dispersed.

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 in-containment refueling water storage tank

(IRWST) into the spreading compartment. The compartment floor is covered by a sacrificial concrete layer that protects a cooling structure against thermal loads resulting from melt spreading. The cooling elements form a series of parallel cooling channels that serve as flow paths for water from the IRWST to flow under the melt, along the sidewalls and onto the top of the molten core debris, cooling and stabilizing the melt.

The SAHRS is a dedicated single-train, non-safety related, thermal-fluid system used to control the environmental conditions within the containment following a severe accident. There are four primary modes of operation, including:

- Passive cooling of molten core debris in the spreading compartment,
- 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.

During the passive cooling process, water covering the core debris would boil off as steam and be 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 would steadily increase, until the SAHRS is configured to operate in the containment spray mode. Active recirculation cooling of the CMSS would occur once the containment spray has sufficiently reduced containment pressure. In this mode of operation, the water level in the spreading compartment would rise to the top of the steam outlet chimney, overflow onto the containment floor, and drain back into the IRWST, where it could recirculate back into the spreading area cooling system.

It is evident that the CMSS method of assuring debris coolability is fairly complex, involving both passive and active cooling modes. A properly-functioning CMSS would keep the debris cool, and prevent sustained concrete ablation in the core spreading room. It definitely would be addressed in severe accident management procedures. The active spray and recirculation cooling modes of a properly-functioning SAHRS would effectively act to keep the pressure in the containment well below the ultimate pressure. Together, the two systems would act together to significantly reduce the LRF in the U.S. EPR and contribute to a successful accident management strategy.

High-pressure core melt ejection. The U.S. EPR design includes two dedicated severe accident depressurization valve trains, each of which consists of a DC-powered depressurization valve in series with an isolation valve connected to the pressurizer. The objective of this design is to convert high-pressure core melt sequences into low-pressure sequences, so that a high-pressure vessel breach can be excluded. The operator would actuate these valves when the core exit temperature exceeds 1200°F (829K). The anticipated loads within the reactor cavity in the event of successful RCS depressurization (i.e., pre-vessel breach RCS pressure) would be well below the reactor cavity design load. Timely operation of the depressurization valves is part of the AM strategy, and is also very important to avert possible induced ruptures of damaged steam generator tubes.

Even if the RPV would fail at high pressure, the pathways for the melt and aerosols dispersed through the reactor cavity cooling ventilation ducts are expected to be tortuous, causing entrainment and de-entrainment, and consequent significant reduction in materials entering the upper containment. It should be noted, however, that deposition of these materials in pump rooms and steam generator compartments is possible; this could affect AM strategies.

Containment bypass. The U.S. EPR design strategy for reducing potential radioactive release in an SGTR is based on having the medium head safety injection pump shut off head at a pressure below the steam generator safety relief valve set point. As a consequence, the likelihood of a SGTR progressing to containment bypass due to secondary system pressure increasing enough to open a safety valve and fail to reseat has been significantly reduced. Automatic isolation of the affected steam generator on its high level signal coincident with the end of partial cool down prevents overfilling and limits liquid release to the environment. No operator actions are required to mitigate the SGTR accident, and the secondary system remains sealed against releases to the environment after the relief valve or its block valve is closed. The subsequent plant cool down is accomplished using the remaining three intact loops.

US-APWR design. The US-APWR design is under review has not yet been certified by the NRC. One COL application is currently being reviewed.

Combustible gas control. The US-APWR containment has a hydrogen gas control system to avoid the risk of containment failure due to fast deflagration or accidental ignition of a critical gas mixture. For controlling hydrogen generated during a severe accident, a number of hydrogen igniters are provided.

Containment response would be monitored to ensure that the pressure loads resulting from the accumulation and combustion of hydrogen could not exceed the containment ultimate capacity pressure limit. To provide reasonable assurance that structural integrity is not compromised, the containment is qualified to withstand global hydrogen deflagration, and flame acceleration.

Although the igniters control the combustible gas concentration, certain scenarios involve the failure of the containment spray system (CSS). In this case, the containment atmosphere may become steam-inerted for some period of time. Condensing the steam upon CSS recovery may lead to combustible conditions. If the hydrogen monitoring system detects high concentrations of combustible gases, the sprays would be turned off.

External reactor vessel cooling (ERVC). In-vessel retention of core debris by external RV cooling is considered as effective potential mechanism for severe accident mitigation. Various physical phenomena related to severe accidents such as steam explosions and CCI, which are the consequences of a result of core debris relocation to the reactor cavity, would be prevented by attaining in-vessel retention. Since the US-APWR is designed to fill the reactor cavity with coolant water when a severe accident occurs, ERVC may be possible. However, in-vessel retention is not credited for the US-APWR severe accident treatment or in the Level 2 PRA study due to its inherent uncertainty.

Flooding of the reactor cavity would occur be manually initiated when core damage is detected, provided the water level is below a certain level. The objective would be to cool down molten debris in the cavity after vessel failure.

Core debris coolability. The US-APWR design includes a large area in the reactor cavity to provide floor space for debris spreading and quenching capability to cool the debris. The design would provide retention and long-term stabilization of the molten core debris inside the containment. It has been calculated that the core debris would spread to a depth of between 7 and 10 inches. The melt would be cooled by the water from two independent sources: the in-containment reactor water storage pit (RWSP) by manually activating containment spray; and fire protection water supply. There would be no cooling from below. Water would flow into the cavity through a drain line. In order to utilize the fire water service system for reactor cavity flooding, it is necessary to establish lineup before activating the fire water service pump.

High-pressure core melt ejection. The US-APWR severe accident dedicated depressurization valves (DVs) design consists of a flow path with two redundant motor-operated remote manual valves connected in series. Non-condensed gas or steam is directly discharged to the containment vessel. The valve arrangement with two normally-closed valves in series minimizes the possibility of inadvertent actuation. The motor-operated valves are controlled from main control room.

The objective of this approach is to convert high-pressure core melt sequences into low-pressure sequences, so that a HPME can be excluded. Timely operation of the DVs is part of the AM strategy, and is a key action to avert possible induced ruptures of damaged steam generator tubes.

Containment bypass. The RCS depressurization feature would act to reduce the probability of temperature-induced SGTR. The capacity of the depressurization valve is considered to be sufficient to reduce RCS pressure for preventing temperature-induced SGTR.

ABWR design. The ABWR design has already been certified by the NRC and one COL application is currently being reviewed.

Combustible gas control. Just as for existing BWRs, the ABWR containment would be inerted during full-power operation. Consequently, insufficient oxygen would be present to cause a hydrogen burn during a severe accident.

Core debris coolability. Numerous features are incorporated into the ABWR design to help mitigate the effects of CCI. The most important are: a large lower drywell floor area with minimal obstructions to the spreading of core debris; a lower drywell flooder (LDF) system; an ac-independent water addition (ACIWA) system; use of sacrificial basaltic concrete for the lower drywell floor; a thick reactor pedestal wall; and a Containment Overpressure Protection System (COPS). The LDF consists of ten piping lines from the suppression pool to the lower drywell, with thermally activated flooder valves attached to them. The thermally activated flooder valves open when the LDW air temperature reaches 260 °C (500 °F), which would be soon after the core debris enters the LDW. The time delay would effectively eliminate energetic steam explosions.

Injection to the reactor vessel using the ACIWA system is intended to prevent core damage. In the event that it is not initiated in time, and reactor vessel melt-through occurs, the ACIWA would provide water to the lower drywell through the breach in the reactor vessel to assist in cooling ex-vessel core debris. This flooding of the lower drywell could be in addition to or in-place of the flooding provided by the LDF. The actual circumstances are accident-sequence specific.

Operation of the ACIWA in the containment spray mode controls atmospheric temperatures in the upper drywell and provides fission product scrubbing. This system is very beneficial in delaying the time to or preventing the opening of the COPS. The COPS passively relieves containment pressurization before containment pressure reaches ASME Service Level C limits. This system provides for a controlled release through a containment vent pathway with fission product scrubbing provided by the suppression pool. The COPS would also prevent catastrophic overpressurization failure of the containment for severe accident sequences involving prolonged periods of CCI.

High-pressure core melt ejection. The probability of HPME is significantly reduced by the highly reliable ADS. In addition, the following ABWR containment design features mitigate the possible effects of high-pressure core melt:

- The containment is segregated into upper drywell (UDW) and LDW regions that communicate directly. However, the design mitigates the ability of high-pressure core melt, ejected within the lower drywell, to reach the UDW.

- Once the horizontal vents have been cleared, the gas and debris leaving the lower drywell would split into two paths: one to the upper drywell and the other to the suppression pool.
- The inerted containment prevents pressurization from combustion of hydrogen generated from oxidation of the metallic constituents of the core debris.

The NRC staff approved this approach because the ADS system is to be provided with a reliable nitrogen supply and dc power to ensure its operability, and the containment design would sufficiently reduce the amount of core debris that would reach the upper drywell. The staff concluded that the criteria of SECY-93-087 would be met.

Severe accident management insights from NRC confirmatory assessments

A complementary activity to NRO's severe accident evaluation is confirmatory assessment by the NRC's Office of Research (RES). Severe accident scenario simulations are done using the MELCOR code, and the results are compared against the MAAP simulations. These results are shared with NRO, and insights obtained from these calculations are factored into the Safety Evaluation Reports (SER) prepared by for each design. Some insights from the confirmatory assessments pertaining to the prevention and mitigation features are provided below in the context of the AM technical basis for each design. Although a number of other examples are described, the focus is on core debris coolability because it is a concern for all reactor designs.

AP1000 insights from confirmatory assessment

Ex-reactor vessel retention. Confirmatory analyses were performed by the staff to assess lower head thermal behavior under severe accident conditions. The following configurations were evaluated:

- Configuration I: a molten ceramic (oxide) pool above a molten metallic layer; and
- Configuration II: a molten ceramic pool sandwiched between a bottom heavy metallic layer and an overlying metallic layer

These configurations are considered bounding in terms of their impact on the lower head integrity.

For Configuration I, a thin top metallic layer could form that is that could cause significant focusing of heat onto the RPV wall. For a low ceramic pool mass, the lower core support plate would not be submerged, and the amount of steel in the metallic layer would be limited, resulting in increased heat fluxes to the RPV wall. For higher ceramic pool masses, the core support plate would be submerged, resulting in a thick metallic layer and reduced heat fluxes to the RPV wall. Results show that the critical heat flux (CHF) would not be exceeded within the molten oxide region. However, the probability of exceeding CHF is about 0.15 within the metallic layer region.

For Configuration II, parametric calculations were performed using point estimate mean values of the masses from Configuration I. The mass fraction of uranium in the bottom layer was fixed at 0.4, and had a density greater than that of the oxide layer, consistent with this configuration. The results of these calculations indicate that the heat fluxes from the vessel remain below CHF at all locations. Thus, the vessel would not be expected to fail if partitioning of the heavy metals from the ceramic pool were to occur.

The applicant did not consider Configuration I to be applicable to the AP1000 because its analyses indicated that the lower plenum debris pool would contact the lower support plate and create a thick metal layer, and in the transient stages before the debris contacts the lower support plate, the debris would be either water cooled or quenched rather than a fully developed naturally circulating pool. For Configuration II, the applicant provided an analysis that produced results similar to those from the staff analysis, and concluded that RPV failure would be physically unreasonable.

The staff concluded that the applicant's position was not justified in light of the uncertainties in the late-phase melt progression and the melt configuration in the lower head. Nevertheless, the probability of vessel failure was judged to be small, and this assumption is inconsequential from the overall risk perspective. From an accident management perspective, the consequences of a breach of the RPV must be taken into account.

<u>*Debris coolability*</u>. The applicant performed deterministic calculations of CCI for a postulated vessel breach event using MAAP4 for two different reactor cavity/basemat concrete compositions, i.e., limestone/common sand and basaltic concrete. With limestone concrete (for which noncondensable gas generation is maximized and concrete ablation minimized), basemat penetration would occur after about 3 days following the onset of core damage. With basaltic concrete (which maximizes concrete ablation and minimizes noncondensable gas generation), the predicted time of basemat melt-through is reduced to about 2 days, with containment overpressure failure expected some time later.

The staff performed calculations using the MELCOR code to confirm the degree of basemat ablation. The calculations indicate a maximum ablation depth of about 1.3 m (4.3 ft) for both limestone and basaltic concrete 2.5 days or more after accident initiation, assuming a dry reactor cavity and uniform distribution of debris within it. The ablation rates predicted by MELCOR are considerably lower than those predicted by MAAP4, partially as a result of a later time of RPV failure in the MELCOR calculation (8 hours in MELCOR versus 2 hours in MAAP). While not directly comparable to the applicant's calculations, the MELCOR calculations support the applicant's finding that basemat penetration would not occur for several days.

The staff concluded that if core debris were not retained in the vessel, the AP1000 design would still provide adequate protection against early containment failure and large releases resulting from CCI. In short, the AP1000 incorporates features that adequately address the guidance called out in SECY-93-087 related to core debris coolability.

<u>**ESBWR insights from confirmatory assessment**</u>

An independent assessment of the ESBWR design response to selected severe accident scenarios was performed using the MELCOR 1.8.6 computer code. The assessment examined 13 accident scenarios from the ESBWR PRA, most of which were simulated by the applicant using MAAP 4.0.6. The results generally support and confirm the PRA accident progression analysis methodology and the applicant's interpretations of its severe accident analyses.

<u>*Core debris coolability*</u>. The applicant provided the results of sensitivity studies using MAAP 4.0.6, given a depressurized RPV, performed to estimate concrete ablation for both limestone and basaltic concrete to assess the potential for RPV pedestal failure. The calculated times from RPV failure to pedestal failure ranged from 26 hours (dry LDW with basaltic concrete) to 55 hours (dry LDW with limestone concrete), to beyond 72 hours (either limestone or basaltic concrete in a flooded LDW). An independent assessment of CCI using MELCOR 1.8.6 confirmed that concrete ablation depths in the axial direction would be of similar or somewhat smaller magnitudes than those predicted by MAAP 4.0.6. The staff believes that, while it is possible that a horizontal "blowout" may occur into the lower reactor building somewhat before 20 hours because of local thinning of the pressure boundary in the region of the BiMAC trough, further analysis of this event is of questionable value given the very low

probability of a dry CCI event. It is reasonable to assume that the containment would fail from over-pressurization before basemat melt-through or pedestal failure.

U.S. EPR insights from confirmatory assessment

Extensive MELCOR confirmatory calculations were performed using the MELCOR 1.8.6 computer code to analyze the representative accident scenarios identified by the applicant and simulated using MAAP 4.0.7. Insights from some of these are discussed below.

Combustible gas control.

Generally, the MAAP-predicted in-vessel hydrogen generation was higher than the MELCOR predictions. This was attributed, for the large part, to the conservative enhancement of the oxidation rate as modeled in MAAP. However, ex-vessel hydrogen production, in the absence of passive flooding of the containment, was reported to be higher in MELCOR, even though the concrete erosion rate was lower than the MAAP prediction. For either mode of hydrogen production, both MAAP and MELCOR results showed that hydrogen concentration in the containment to remain low due to the effective recombination of hydrogen and oxygen by PARs.

MELCOR calculations for the representative accident scenarios have confirmed that, due to efficient recombination by PARs, there is little potential for formation of pockets of high hydrogen concentration inside the EPR containment and hence deflagration or detonation is unlikely.

Molten fuel/coolant interactions.

The time duration from vessel breach to reactor pit melt plug failure was found to be much shorter in MELCOR as compared to MAAP predictions, even though the MAAP-predicted debris temperature in the reactor pit before the melt plug failure was found to be 500 °K to 600 °K higher than the MELCOR predictions. The MELCOR results showed that the entire core debris content may not be in the reactor pit by the time the reactor melt plug melt-through occurs. Nonetheless, the mass of any remaining core debris arriving into the reactor pit after melt plug failure was calculated to be small (~5%). This delayed relocation can have implications in terms of ex-vessel energetic fuel coolant interactions, which the EPR design of the reactor pit was intended to circumvent.

Core debris coolability. Differences in MELCOR and MAAP prediction of concrete erosion were found with the MELCOR-calculated erosion rate being lower than that of the MAAP prediction. For most of the scenarios examined, the MELCOR-predicted debris temperature in the spreading compartment was shown to be lower due to lower initial debris temperature as compared with MAAP.

Provided that full uniform spreading of the melt over the floor of the spreading compartment occurs, and provided that IRWST passive injection is initiated as designed, melt cooling and stabilization were predicted to take place by both MAAP and MELCOR. Nonetheless, for most of the scenarios examined, the predicted core debris cool-down rate on the spreading compartment floor was shown to be faster in MELCOR as compared with MAAP.

US-APWR Insights from Confirmatory Assessment

A number of severe accident scenarios were simulated for the US-APWR using the MELCOR 1.8.6 computer code. A total of six representative scenarios were selected, based on the accident sequences described in the US-APWR PRA. Two additional calculations were also performed to examine the effect of vessel depressurization as a result of creep rupture of the hot leg. The results of the MELCOR calculations were compared to MAAP simulations documented in the US-APWR PRA. Differences in the modeling approach and assumptions in the initial and boundary conditions, resulted in variations in the details of the accident progression. Overall, the timing of accident progression and failure of the lower head were in relatively good agreement given the uncertainties in severe accidents. A specific comparison on debris coolability is briefly described below.

Core debris coolability. Results of accident progression analyses using MAAP 4.0.6 for selected representative accident sequences, in which both features of the diverse reactor cavity flooding system are available, indicate that molten debris is appropriately cooled down in a reactor cavity water pool and no concrete erosion occurs for accident sequences in which molten debris drops into water pool. Results of accident progression analyses for characteristic accident sequences, in which no continuous reactor cavity flooding means is available, indicate that the earliest possibility of complete erosion of the reactor cavity floor concrete (i.e. more than 40 in. erosion of concrete) is approximately 28 hours after onset of core damage.

MELCOR predicts that cavity ablation can be averted as long as water remains in the lower cavity region. The core concrete interaction shows that after vessel breach and relocation of the core to the cavity, the core debris is cooled by the cavity water and reaches a near steady state, where the decay heat generation is balanced by the heat loss to the cavity water. As soon as the cavity water is boiled off, ablation of the 40 inch basemat results in the failure of the containment pressure boundary. Basemat melt through does not occur before 24 hours following the initiation of the accident. Thus, the two simulations are in good agreement with respect to core debris coolability.

ABWR Insights from Confirmatory Assessment

Core debris coolability. General Electric carried out analyses for each accident sequence using a very early version of the MAAP code, adapted to be able to simulate the ABWR. These analyses generally indicate core debris coolability and little, if any, CCI if LDW flooding would occur. The time-to-release of fission products ranged from 8.6 to 50 hours from the start of the transient, with the most likely fission product release location through the containment overpressure protection system (COPS). In most cases the time to fission product release exceeded 24 hours. For cases where the LDW would not be flooded and containment heat removal is lost, the time to COPS initiation would be less – sometimes less than 24 hours after accident initiation.

The staff analyzed in-house the response of the ABWR using the MELCOR code. In addition, the staff performed additional analyses using the MELCOR code. The MELCOR results generally reproduced the event sequences predicted by MAAP, albeit usually with timing shifts. These timing shifts did not affect the safety insights for the containment analyses.

Conclusions

The design application reviews, both complete and ongoing, are confirming that the new reactors will be safer by including the new severe accident mitigation systems that address the concerns expressed in SECY-90-016 and SECY-93-087. The PRAs all calculate significantly lower core damage frequencies and large release frequencies than for existing reactors. All of the applicants claim that the new regulatory requirements emanating from SECY-90-016 and SECY-93-087 will be met. Both the preparation of the design certification applications by the applicants and the technical reviews by the NRC staff are revealing insights on how the use of these design features will enhance the technical basis now in place for the existing reactors, so that appropriate accident management procedures will be put in place.