19.34 SEVERE ACCIDENT PHENOMENA TREATMENT

19.34.1 Introduction

This chapter describes how severe accident phenomenology is treated in the probabilistic risk assessment (PRA). In the PRA, the Modular Accident Analysis Program, version 4.0 code (MAAP4) (Reference 19.34-1) is used mainly to estimate source terms. Severe accident phenomenological uncertainties are treated with Risk-Oriented Accident Analysis Methodology (ROAAM) (Reference 19.34-2) phenomenological evaluations, with AP600-specific decomposition event tree phenomenological evaluations, or by assuming that certain low-frequency severe accident phenomena fail the containment. The objective of these studies is to show, with a high degree of confidence, that the AP600 containment will accommodate the effects of severe accidents in a range of scenarios for at least the first 24 hours after the onset of core damage. The results of these studies show the containment does not fail even after 24 hours in the large majority of core damage events analyzed. Such evaluations are recommended in NUREG-1335, Individual Plant Examination: Submittal Guidance, (Reference 19.34-3) to demonstrate the robustness of the containment design.

19.34.2 Treatment of Physical Processes

The following eight issues are identified in Reference 19.34-4 as being representative of the phenomenological issues pertaining to severe accident conditions:

1. Loss-of-coolant accident (LOCA)
2. Fuel-coolant interaction (steam explosion)
3. Hydrogen combustion and detonation
4. Melt attack on concrete structure or containment pressure boundary
5. High-pressure melt ejection
6. Core-concrete interaction (CCI)
7. Containment pressurization from decay heat
8. Elevated temperature (equipment survivability)

The challenge to the containment integrity from a LOCA is not specifically related to severe accident phenomenology and is not discussed here. Treatment of physical processes affecting the remaining challenges is discussed in this section. For the AP600 design, issues 4, 6, and 7 above arise primarily from the same physical process, debris coolability. Therefore, they are discussed together with that subject.

ROAAM analyses and event trees are developed for key severe accident phenomena to provide a systematic and logical method to investigate the uncertainties in the phenomena. The analyses are supported by available experimental information from open literature. Phenomenological evaluation summaries (Reference 19.34-5) provide additional background information.

19.34.2.1 In-Vessel Retention of Molten Core Debris

In-vessel retention (IVR) of core debris by cooling from the outside of the vessel is a severe accident mitigation attribute of the AP600. With the reactor vessel intact and debris retained in the lower head, there is no need to examine phenomena that may occur as a result of core debris being relocated to the reactor cavity. The AP600 is provided with reactor vessel insulation that promotes in-vessel retention and surface treatment that promotes wetability of the external surface. The AP600 containment event trees include a node to ascertain that the reactor coolant system (RCS) is depressurized and a node to determine if sufficient water is available in the cavity. Success at both of these nodes is required to demonstrate that the conditions and assumptions of Reference 19.34-6 are met and vessel failure is physically unreasonable.

The engineered design features of the AP600 containment provides for flooding of the containment cavity region during accidents and, thereby, submergence of the reactor vessel lower head in water. Liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The AP600 includes a provision for draining the in-containment refueling water storage tank (IRWST) water into the reactor cavity through an operator action. Therefore, the reactor pressure vessel lower head is expected to be submerged in water.

Reference 19.34-6 contains an AP600-specific ROAAM evaluation to determine the likelihood that sufficient heat can be removed from the outside surface of the submerged reactor pressure vessel lower head to prevent reactor vessel failure and relocation of debris to containment. Keeping the core debris in the vessel eliminates the need for consideration of ex-vessel events, such as ex-vessel steam explosion and core-concrete interaction. The approach used assumes that:

- The RCS is depressurized.
- The reactor vessel is submerged above the top of the in-vessel debris bed.
- The reflective insulation does not impede water cooling of the vessel
- The external surface treatments do not impair wetability of the vessel.

Accounting for the uncertainties in thermal-hydraulic parameters, the heat fluxes to the vessel wall and reactor vessel internals necessary to remove the decay heat from the debris pool are calculated. These heat fluxes are compared to the critical heat flux for downward-facing curved surfaces and vessel failure is assumed if the critical heat flux is exceeded. The results show large margin to failure for the reactor vessel if it is externally cooled by water.

19.34.2.2 Fuel-Coolant Interaction (Steam Explosions)

A steam explosion may occur as a result of molten metal or core debris mixing with water and interacting thermally. Steam explosions are postulated inside the reactor vessel when debris relocates into the lower head from the core region, and in the reactor cavity if the vessel fails and debris is ejected from it.

A ROAAM analysis of the AP600 reactor vessel lower head integrity under in-vessel steam explosion loading is presented in Reference 19.34-7. Failure of the lower head would impair the in-vessel retention capability of the reactor vessel. The ROAAM analysis concludes that lower-head vessel failure due to in-vessel steam explosions is physically unreasonable.

An evaluation specific to the AP600 to investigate the potential for containment failure induced by in-vessel steam explosions (α-mode containment failure) is presented in Reference 19.34-5. The evaluation concludes that the likelihood for vessel failure and subsequent containment failure due to in-vessel steam explosion is so small as to be negligible. The in-vessel fuel-coolant interaction has little probability of generating sufficient energy, in a short time scale, to produce a missile that could fail the AP600 containment. This is in agreement with the conclusions of the U.S. Nuclear Regulatory Commission (NRC)-sponsored Steam Explosion Review Group (Reference 19.34-8).

A significant ex-vessel steam explosion from core debris-water interaction can be postulated to occur only in the reactor cavity. This is because of the AP600 containment layout and the design features to prevent high-pressure core melt, see subsection 19.34.2.4. Ex-vessel steam explosion is mitigated by the in-vessel retention of the core debris. In the event that the reactor cavity is not flooded and the vessel fails, the PRA model does not credit containment integrity.

19.34.2.3 Hydrogen Combustion and Detonation

Section 19.41 discusses the potential for hydrogen deflagration and detonation during a severe accident sequence in the AP600 containment. The analysis examines diffusion flame burning and local detonation occurring during in-vessel hydrogen generation prior to hydrogen mixing in the containment and global deflagration and detonation, which may occur later when the hydrogen is mixed throughout the containment. Only in-vessel hydrogen generation is considered, since vessel failure and ex-vessel debris relocation is assumed to fail containment.

If the igniters are operational, the potential for diffusion-flame-induced containment failures is considered during the hydrogen generation and release from the RCS. Diffusion flames may be formed when high-concentration, nonflammable hydrogen plumes encounter oxygen and burn as a standing flame. Flames that have a large view factor or that impinge on the containment pressure boundary may fail the containment pressure boundary due to the high temperature. The pathways that in-vessel hydrogen can take to containment are reviewed for potential impact on containment integrity. Locations where diffusion flames may be postulated are examined for potential failure of the containment due to creep of the containment shell at high temperature (greater than 400°F).

The potential for directly initiated hydrogen detonations in the AP600 containment is examined in Reference 19.34-5. After examining the various potential energy sources in containment, it is concluded that the largest possible energy source in the containment, a 4 kv arc, cannot initiate a detonation. Therefore, containment failure from a directly initiated detonation wave is not considered to be a credible event for the AP600 containment.

The likelihood of a deflagration-to-detonation transition (DDT) in the AP600 containment is evaluated locally in confined compartments during in-vessel hydrogen generation and globally after in-vessel generation is concluded and hydrogen is mixed in the containment. For a DDT to occur, the combination of the gas mixture sensitivity to detonation and the geometric configuration potential for flame acceleration must be conducive to DDT. Since the hydrogen concentration necessary to form a detonable mixture depends on the size of the enclosure, concentration requirements for DDT in different regions of the AP600 containment are extrapolated from the FLAME facility data (Reference 19.34-9) using scaling arguments based on the detonation cell width. The geometric requirement is evaluated considering aspects such as the degree of confinement and the extent and type of obstacles present in the postulated flame propagation path. In all cases, DDT is assumed to result in containment failure.

Global hydrogen deflagration and the potential for containment failure are modeled on the containment event tree. Adiabatic, isochoric, complete combustion (AICC) is assumed, and peak pressure probability distributions are developed for the accident scenarios. The probability of containment failure due to hydrogen deflagration is evaluated from the combination of the containment failure probability distribution and the peak pressure probability distribution.

19.34.2.4 High-Pressure Melt Ejection

The AP600 incorporates design features that prevent high-pressure core melt. These features include the passive residual heat removal (PRHR) system and the automatic depressurization system (ADS). These design features provide primary system heat removal and depressurization in the unlikely event they are required in a transient.

In high-pressure core damage sequences (that is, non-LOCA or very small LOCA events with the ADS and passive residual heat removal inoperable), the potential exists for creep-rupture-induced failures of the RCS piping at the hot-leg nozzles, the surge line, the steam generator tubes and, given debris relocation to the lower plenum, in the reactor vessel lower head. Failure of the hot-leg nozzle or surge line prior to failures of other components results in the rapid depressurization of the RCS. Failure of the steam generator tubes results in a containment bypass and a large release of fission products to the environment. Failure of the lower head of the reactor vessel results in the potential for high-pressure melt ejection.

The AP600 RCS loops have canned-motor pumps mounted to the steam generator outlet plenum. The coolant loops do not have water-trap loop seals as in conventional plant designs. A large natural-circulation flow heats up the reactor coolant loop components in a relatively uniform manner. Hot-leg nozzle failure is expected prior to steam generator tube failure, but because of large uncertainties, hot-leg nozzle creep rupture failure is not credited with preventing steam generator tube failure. Steam generator tube failure is assumed in high-pressure sequences unless operator action to depressurize the RCS with the ADS is successful. A deterministic assessment of the impact of reactor pressure vessel failure at high RCS pressure (HPME) on containment integrity is provided in Appendix 19B.

19.34.2.5 Core Debris Coolability

In accident sequences where the reactor pressure vessel failure is not prevented, core debris may be discharged into the reactor cavity. The AP600 cavity design enhances the potential for long-term core debris coolability by providing a reactor cavity floor area greater than 0.02 m^2/MWt for the core debris to spread. Reference 19.34-10 provides the technical basis for selection of this surface area criterion for long-term debris coolability. Condensate is returned to the reactor cavity through the in-containment refueling water storage tank and the recirculation lines, thereby providing a long-term supply of water to cool the core debris.

To accommodate the requirements for in-vessel retention of core debris, the AP600 provides highly reliable RCS depressurization and cavity flooding capability. There are significant uncertainties associated with debris spreading into a water-filled cavity. Debris spreading is mainly a function of the highly uncertain vessel failure mode. A large-scale lower-head failure releasing debris at a high rate would enhance spreading, while a localized failure mode would release debris at a slow rate, which would most likely cause the debris to pile up under the reactor vessel and minimize spreading.

Given the uncertainties in the debris spreading and in non-condensible gas generation and combustion, the PRA does not credit containment integrity in the event of failure of the lower head of the vessel and relocation of the core. The deterministic analyses of debris spreading and core-concrete interaction in the AP600 cavity are summarized in Appendix 19B.

19.34.2.6 Elevated Temperatures (Equipment Survivability)

SECY-93-087, Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs, states that equipment identified as being useful to mitigate the consequences of severe accidents must be designed to provide reasonable assurance that it will continue to operate in a severe accident environment for the duration it is needed to accomplish its function. Also, 10 CFR 50.34(f) requires safety equipment to continue performing its function after being exposed to a containment environment created as a consequence of generating a quantity of hydrogen equivalent to that from 100-percent cladding oxidation.

The functions of the equipment in containment for which credit is taken in the AP600 PRA were reviewed to determine if the equipment is required to operate in a severe accident environment and beyond design basis limits. In the calculation of the large release frequency, only the containment pressure boundary is credited to perform beyond its design basis. Other equipment is credited in the analysis, but either the containment environmental conditions do not exceed the equipment qualification conditions at the time the function is performed, or the design basis for the equipment is a severe accident environment. The radiation environment for equipment qualification for safety-related equipment in containment is based on the severe accident source term involving significant in-vessel fuel melting described in NUREG-1465. The equipment credited in the large-release frequency calculation is assumed to survive the radiation dose associated with the accidents over the time it is required to perform its function.

The equipment that is credited in the evaluation of the large-release frequency in the AP600 containment event tree analysis in the PRA includes:

- Containment pressure boundary
- ADS valves and controllers required for post-core damage recovery of RCS depressurization
- Containment isolation
- Passive containment cooling
- Hydrogen igniters
- Reactor cavity flooding
- Post-accident monitoring equipment

19.34.2.7 Summary

The potential for and the consequences of severe accident phenomena are evaluated. This information is applied to the containment event trees and used in the quantification of the large release frequency.

These severe accident phenomena are studied in the PRA models to understand the AP600 containment response and determine the source terms.

19.34.3 Analysis Method

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19.34.4 Severe Accident Analyses

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19.34.5 Summary

The consequences of severe accident phenomena are evaluated to support the containment performance requirements of SECY-93-087. This information is applied to the release categories. The fission-product source terms are determined from analyses that utilize sequences bounding the potential source terms from a severe accident.

19.34.6 Insights and Conclusions

The analyses of the severe accident phenomena for the AP600 PRA highlight the following insights and conclusions:

- The design of the AP600 reactor vessel, vessel insulation, and reactor cavity, and the ability to flood the cavity after a severe accident, reduce the potential challenges to the containment integrity by maintaining the vessel integrity.

- Should a failure of the reactor vessel occur, the design of the reactor cavity enhances the ability to cool any core debris that should exit the vessel.
- Lower-head vessel failure due to in-vessel steam explosions is physically unreasonable.
- The ADS and PRHR system are design features that can be used to prevent high-pressure core melt in a severe accident.
- A directly initiated hydrogen detonation in the AP600 containment is not a credible event.
- The equipment needed to mitigate the consequences of a severe accident is designed to provide reasonable assurance that it will continue to operate during an accident.

19.34.7 References

19.34-1 "EPRI MAAP 4.0 Users Manual."

19.34-2 Theofanous, T.G., "On the Proper Formulation of Safety Goals and Assessment of Safety Margins for Rare and High Consequence Hazards," Reliability Engineering and System Safety, 1996.

19.34-3 NUREG-1335, "Individual Plant Examination: Submittal Guidance," August 1989.

19.34-4 Letter from D. A. Ward, Advisory Committee on Reactor Safeguards, to K. A. Carr, Chairman, Nuclear Regulatory Commission, "Proposed Criteria to Accommodate Severe Accidents in Containment Design," dated May 17, 1991.

19.34-5 "AP600 Phenomenological Evaluation Summaries," WCAP-13388 (Proprietary) and WCAP-13389 (Nonproprietary) Rev. 1, June 1994.

19.34-6 Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

19.34-7 Theofanous, T.G., et al., "Lower Head Integrity Under In-Vessel Steam Explosion Loads," DOE/ID-10541, July 1996.

19.34-8 NUREG-1116, "A Review of the Current Understanding of the Potential for Containment Failure From In-Vessel Steam Explosions," 1985.

19.34-9 Sherman, M. P., Tieszen, S. R., and Benedick, W. B., "FLAME Facility - The Effects of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation for Hydrogen-Air Mixtures at Large Scale," NUREG/CR-5275, April 1989.

19.34-10 "Technical Support for The Debris Coolability Requirements for Advanced Light Water Reactors in the Utility/EPRI Light Water Reactor Requirements Document," Advanced Reactor Safety Program, DOE/ID-10278, June 1990.

19.34-11 Deleted.

19.34-12 Deleted.

TABLES 34-1 THROUGH 34-38 NOT INCLUDED IN THE DCD.
FIGURES 34-1 THROUGH 34-420 NOT INCLUDED IN THE DCD.

19.35 Containment Event Tree Analysis

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