MODELING AND PHENOMENOLOGICAL ASPECTS OF SEVERE ACCIDENTS IN INTEGRAL PRESSURIZED WATER REACTORS[1]

I. K. Madni
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission
Rockville, Maryland 20852

and

M. Khatib-Rahbar
Energy Research Inc.
P.O. Box 2034
Rockville, Maryland 20847

Abstract

This paper focuses on modeling and phenomenological issues relevant to analysis of severe accidents in integral Pressurized Water Reactors (iPWRs). It identifies relevant thermal-hydraulics, melt progression and fission product release and transport phenomena, and discusses the applicability of the MELCOR computer code to modeling of severe accidents in iPWRs. Areas where the current MELCOR severe accident modeling framework has limitations in the representation of phenomenological processes are identified and examples of possible modeling remedies are discussed. The paper identifies modeling and phenomenological issues that contribute to differences in the calculated reactor coolant system and containment response for iPWRs as compared to traditional PWRs under severe accident conditions.

1. INTRODUCTION

Currently, several designs that can be classified as integral Pressurized Water Reactors (iPWRs) are under pre-application review by the U.S. Nuclear Regulatory Commission (NRC). The characteristic feature of these iPWRs that differentiates them from conventional PWRs is a self-contained integral assembly consisting of the reactor core, a riser, a pressurizer, and steam generators all housed within a single pressure vessel (PV).

The primary coolant flow during steady-state operation of iPWRs is driven by internal reactor coolant pumps or by natural circulation depending on the design. The reactor core is connected to a riser that acts as a “hot leg” transporting the coolant to the steam generators, which use either once-through or helical tube configuration with the primary coolant flowing either inside or outside the tubes. Furthermore, the proposed designs have aimed at eliminating the potential for large loss of coolant accidents (LOCAs) by eliminating large pipes and other penetrations to the reactor coolant system (RCS). In addition, any pressure vessel penetrations are small and are located high up in the pressure vessel such that they would eliminate the likelihood of core uncovery by coolant blow-down and/or drainage following any LOCA events.

These designs often use passive innovative means for emergency core cooling systems (ECCS) that may include the use of depressurization systems, recirculation valves, and isolation condensers. The containment of the iPWRs is also unique to each design, where it can be similar to the large dry containment of conventional PWRs or it can be an unconventional design wherein the containment is a compact steel vessel that surrounds the PV and is immersed inside a large water pool.

[1] Work performed under the auspices of the United States Nuclear Regulatory Commission (NRC).

Due to the relatively low power output for these reactors (i.e., from 40 to 150 MWe), these designs often involve multiple identical "modules" that have minimal shared systems and are intended to be installed at a given site over a period of time, on an as-needed basis. These designs are often referred to as small modular reactors (SMRs), and they appear to offer improved safety while circumventing economic hurdles through modularization.

Even though the frequency of fuel/core damage in iPWRs is expected to be significantly lower than for conventional PWR plants, nonetheless, severe accidents cannot be totally eliminated from consideration, because these designs may remain vulnerable to natural phenomena, and other random and common cause failures. Furthermore, the unique designs of SMRs as compared to conventional PWRs may also introduce unique phenomenological challenges during severe accidents, requiring experimentation and/or development of new or revised models for implementation into the available severe accident progression and radiological source term prediction codes (e.g., MELCOR [1]).

The objectives of this paper are to identify thermal-hydraulics, melt progression, and fission product release and transport phenomena that are relevant to modeling of severe accidents in iPWRs, and to provide an assessment of the applicability of the Nuclear Regulatory Commission (NRC)-sponsored MELCOR computer code to these analyses. Examples of representative accident sequences and their simulation results are used to provide insights into phenomenological issues that can impact reactor and containment system performance under typical transient- and LOCA-induced severe accident conditions. These accidents include a leakage from the reactor coolant system (i.e., small LOCA), and a transient (e.g., station blackout).

There has been increased interest on the part of vendors and utilities to initiate pre-application interactions with the NRC for possible future design certification consideration of their iPWR designs. This has led the NRC to increase efforts to assess various analytic tools for applications to confirmatory analyses of iPWRs.

2. PHENOMENA RELEVANT TO SEVERE ACCIDENTS IN iPWRs

To facilitate the identification of unique phenomena during severe accident progression in iPWRs, the accident progression is divided into different phases including pre-core melt (Phase 1), core melt progression (Phase 2), melt relocation into the lower plenum (Phase 3) and vessel breach (Phase 4). Both in-vessel and ex-vessel phenomena are considered.

Table 1 lists the relevant phenomena identification that can be used to identify modeling capabilities and/or shortcomings in codes that may be used for confirmatory analysis of severe accidents in iPWRs (e.g. MELCOR). The phenomena summarized below are not comprehensive in that the only phenomena considered are those that are unique to iPWRs or are likely to have higher significance in iPWRs as compared to conventional PWRs. Some phenomena of importance are common to severe accidents in both conventional PWRs and iPWRs, while others are more significant to iPWRs. Issues for which code changes and/or parametric sensitivity studies are warranted are not extensive, and for the most part, the existing code models are deemed adequate. Note that phenomena such as fission product release, aerosol transport, hydrogen generation, transport of non-condensable gases, combustion, core-concrete interactions (not included in Table 1), where they may be applicable to iPWRs, are also common to conventional PWRs, and hence MELCOR code adequacy for these phenomena is not expected to be any different for iPWRs as compared to PWRs. The MELCOR capability/limitation column in Table 1 has been prepared based on experience gained from this and earlier studies by the authors.

Figure 1 shows MELCOR-calculated steady state flow and temperature fields in the core, the region above the core, and the riser section for an example natural circulation-based iPWR design. A recirculating flow pattern caused by in-flow of water from the downcomer through a leakage path connecting the bottom of the riser and the downcomer is also shown in the figure.

Table 1 Phenomena relevant to simulation of severe accidents in iPWRs and relevant capabilities and limitations of MELCOR

Phenomena	Description		MELCOR Capability/Limitation
Natural circulation	Phase 1 - Pre-Core Melt	Single phase natural circulation of water through the riser, steam generators and into the downcomer and reactor core. Following a drop in water level below the top of the riser, recirculating and multi-dimensional flows (see Figure 1) and heat losses through structures can delay the on-set of core uncovery. Flow patterns and intermittency of natural circulation are impacted by accident type and design of ECCS (Figure 3). For instance, condensation within isolation condenser can produce chugging and oscillatory flow patterns (Figure 3) and level oscillations (Figure 4). In addition, condensation and natural circulation within the containment can impact ECCS performance.	MELCOR is considered adequate to represent these flows; however, modeling features and spatial nodalizations can influence the predictions.
	Phase 2 - Core Melt Progression	Natural circulation of steam and hydrogen continues during uncovery and melt progression. These flows can significantly impact the onset of melting and relocation, including fission product release and deposition. This is more important for transients than LOCAs and multiple SGTR events. In iPWR designs that use containment water to cool the reactor vessel from the outside, natural circulation of water inside containment impacts heat removal and melt progression. Furthermore, since iPWRs include an engineered depressurization system, the potential for creep-rupture of the reactor coolant system due to circulation of hot gases at high pressure is eliminated.	MELCOR is considered adequate to represent these flows; however, modeling features and spatial nodalizations can influence the predictions.
	Phase 3 - Relocation & Behavior in Lower Plenum	Natural circulation processes described above continue during this phase and may be impacted by nature of core relocation. Furthermore, since iPWRs include an engineered depressurization system, interaction of melt with the lower head can result in local failure of the PV by melt attack; however, potential for high pressure melt ejection and subsequent direct containment heating is eliminated.	MELCOR is considered adequate to represent these flows; however, modeling features and spatial nodalizations can influence the predictions.
	Phase 4 - Vessel Breach & Ex-Vessel Phase	Following vessel breach, natural circulation inside the RCS is only important in altering the radiological releases due to revaporization of previously deposited fission products. This is accident and plant design specific and can be important for plants in which containment integrity can be compromised over the long-term.	MELCOR models are considered adequate.

Phenomena	Description		MELCOR Capability/Limitation
Heat transfer in steam generators	Phase 1	Heat transfer (forced convection, nucleate boiling) in once-through tubes and helical tubes is important in the early phase of accidents impacting the onset of core uncovery and fuel damage (i.e., comparatively higher heat transfer and steam superheat have been reported for helical tubes).	MELCOR heat transfer package does not include heat transfer coefficient correlations applicable to helical tubes. Furthermore, existing straight tube heat transfer coefficient correlations in MELCOR are generally expected to be applicable to straight-tube designs; however, these correlations need to be assessed against vendor data, when available. In addition, sensitivity studies can be performed using the built-in MELCOR heat transfer package to assess the significance of these heat transfer closure relations.
Venting through spargers	Phases I & 2	Steam venting through spargers into large cooling pool impacts condensation rate and containment pressurization.	MELCOR model originally developed for pressure suppression pool in boiling water reactors is considered adequate for this application.
Heat Removal from Fuel	Phase 2	Generally, in conventional PWRs, following core uncovery, the power density results in rapid heatup and melting of reactor fuel. However, using some of the existing models, predictions show that it is possible to retain partially intact fuel geometry with minimal degradation in iPWRs (see Figure 2). This is a result of steam cooling for lower power density iPWRs.	MELCOR's parametric fuel degradation models need to be examined to determine their predictive reliability under these conditions.
Metal Oxidation	Phases 2 & 3	Metal oxidation is important contributor to heat source and radiological releases during melt progression, relocation, and interactions with water in the lower plenum.	MELCOR model is considered adequate since the phenomena are not unique to SMRs. The significance of metal oxidation during interactions with water in the lower plenum needs to be evaluated for SMRs since MELCOR does not currently model metal oxidation in the lower plenum during interaction with water.

Phenomena	Description		MELCOR Capability/Limitation
Molten Pool Convection & Debris Cooling	Phase 3	Melt cooling through the reactor pressure vessel lower head is impacted by molten pool convection, amount of metals in the melt pool. Furthermore, issues important for heat removal on the outside of the lower head include the ability for steam to vent in the upper parts of the lower PV head, and natural flow of water towards the lower head.	MELCOR models all the relevant phenomena.
Failure mode of RPV lower head	Phase 3	For plants that do not include any PV lower head penetrations, under low pressure conditions, the most likely failure mode is expected to be by creep locally.	MELCOR can be used to model life-time and creep-induced failure of the lower head, including hole ablation.
Debris cooling inside containment	Phase 4	Depending on the design, different methods may be used to cool the molten debris at the bottom/floor of containment. In some designs, water cooling from below the containment vessel is used to cool the debris. While, in others, melt spreading or combination of spreading with cooling from below using an engineered "core catcher" may be utilized to arrest the damage progression.	MELCOR can only treat debris cooling from below inside the RPV lower head. Furthermore, melt spreading is not treated by MELCOR; however, it can be assessed parametrically. A model for debris cooling from below has been developed by ERI and implemented into MELCOR via the Control Function feature (see Figure 5).
Fission Product Release & Transport	Phases 2 through 4	The processes associated with release of radionuclide from fuel are similar to those for conventional PWRs; however, quantitative releases may be vastly different and are impacted by extent of cladding oxidation, and the time at temperature prior to degradation and relocation. Furthermore, deposition and transport of radionuclides are tightly coupled to thermal-hydraulic processes and material composition of reactor coolant system (RCS) structures (e.g., impacting chemisorption - presence of tin may result in the formation tin telluride which have a low propensity for vaporization).	MELCOR models all the relevant phenomena.

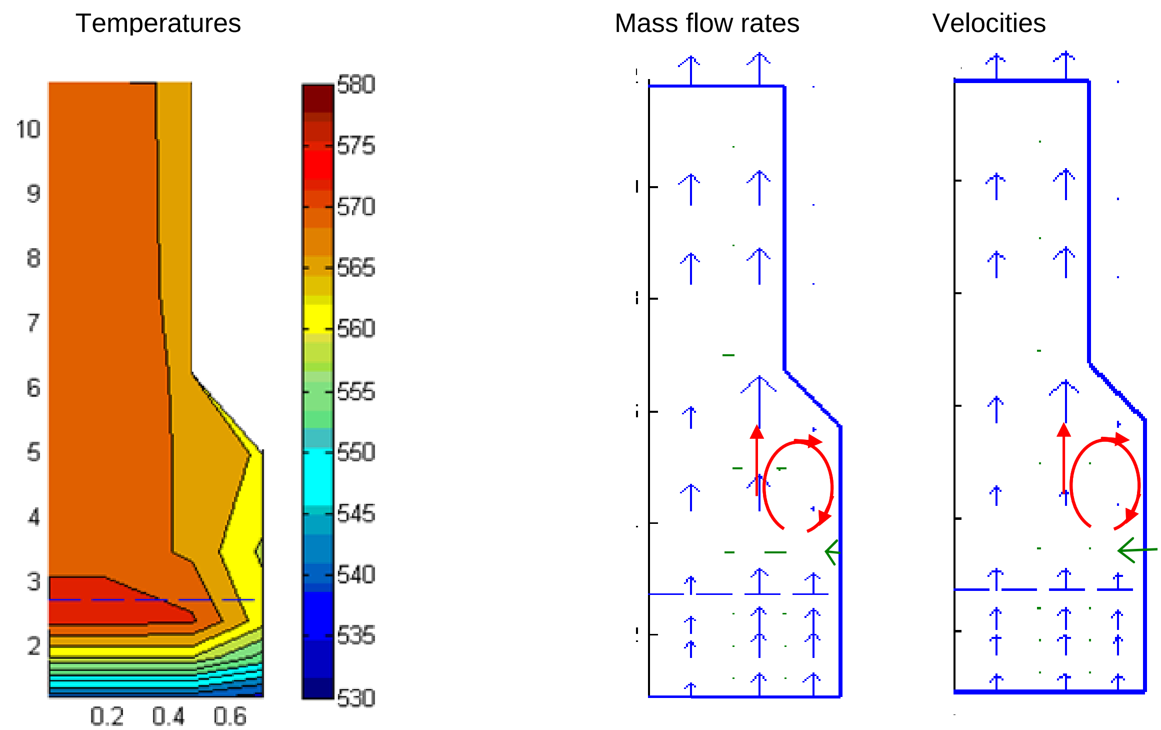

Figure 1 Temperature, flow rate and velocity fields in the core and the riser section

Figure 2 shows MELCOR-calculated fuel temperature results for an example small LOCA-initiated severe accident scenario. Calculations using MELCOR have shown that even though fuel relocation was not predicted to occur for some accident scenarios (see Figure 2), the fuel remained at elevated temperatures for a very long time, resulting in substantial release of radionuclides to the RCS (e.g., ~80% of the initial inventory of volatile groups such as iodine and cesium).

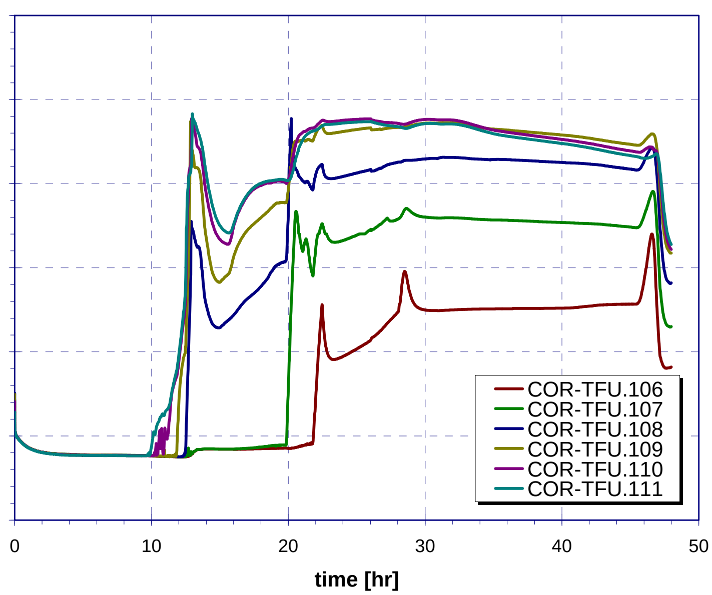

Figure 2 Typical fuel temperature responses under a postulated LOCA condition

The MELCOR predictions (Figure 3) show the natural circulation path that is responsible for heat removal through the PV. The hot steam exiting the core flows into the riser section and into the steam generator, flowing down into the downcomer, where it condenses on the inside surface of the PV, and the condensate returns to the lower region of the downcomer where it flows back to the core. This natural circulation pattern results in adequate removal of core-generated decay heat through the PV cylindrical wall and lower head. As a result, the fuel temperature gradually decreases after ~20 hours into the accident.

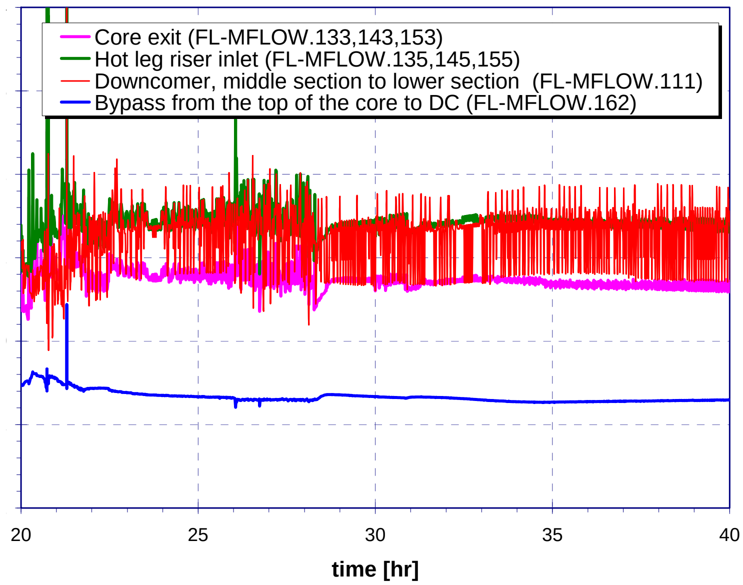

Figure 3 Mass flow in the reactor vessel

Figure 4 shows the impact of condensation-induced chugging on the reactor water level for a small LOCA event with failure of valves that are intended to enable recirculation of water into the reactor vessel in an iPWR.

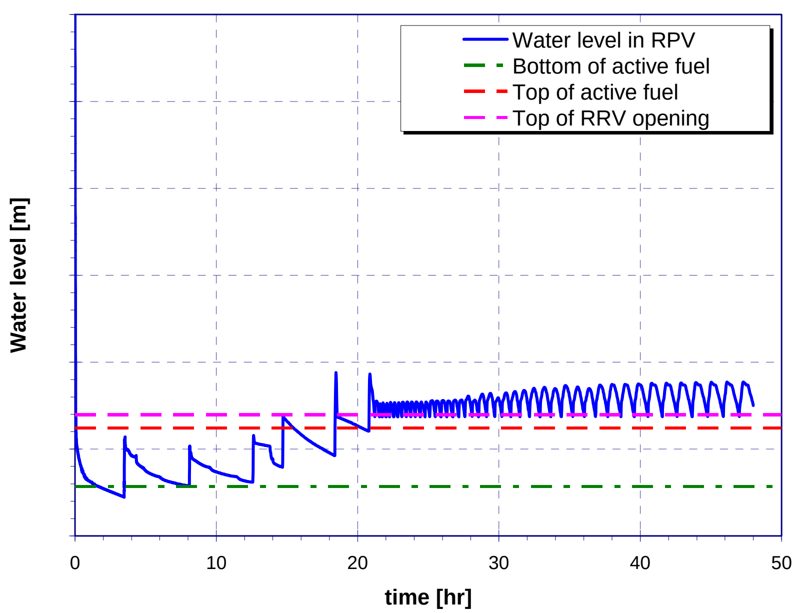

Figure 4 RPV swollen water level

As discussed in Table 1, MELCOR has a model to represent molten debris cooling from below inside the RPV lower head only; however, in designs that enable debris cooling from outside of a steel containment vessel, a model has been developed and implemented into MELCOR via the Control Function feature of the code. This model enables simulation of a molten debris pool configuration, with provisions for heat removal from the surface by radiation, and from below by pool boiling [2], provided the calculated local heat flux remains below the calculated critical heat flux limit. Figure 5 shows the typical debris temperature for another example scenario that leads to fuel relocation, vessel breach and relocation into the containment, showing rapid cooling and freezing on the debris pool due to external cooling.

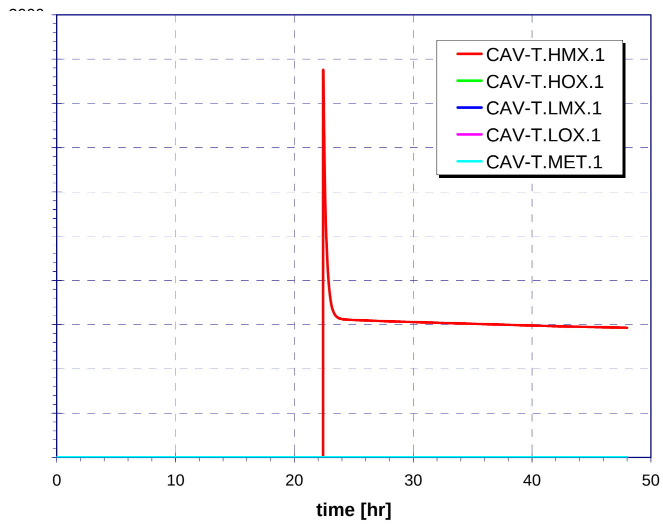

Figure 5 Core debris temperature inside the containment vessel as it is cooled from below

3. SUMMARY AND CONCLUSIONS

NRC is currently developing pre-application MELCOR models for NuScale and mPower, both SMR (iPWR) designs that are still evolving and hence not finalized. These models have focused on assessing the applicability or capability of MELCOR to simulate severe accidents in SMRs, and on gaining insights into the predicted phenomenological behavior under typical severe accident conditions.

Comparing the dominant phenomenological issues during severe accidents in typical iPWRs to MELCOR modeling capabilities shows that MELCOR is able to simulate the majority of the key phenomena. In areas where either limitations exist or MELCOR models do not account for specific processes (e.g., heat transfer in helical tubes), parametric sensitivity calculations can help to assess the potential impact of the identified limitation. Furthermore, unique phenomenological processes are predicted to occur such as condensation-induced chugging, or sustained heatup of fuel to high temperatures but below relocation conditions. Although these may be different from conditions observed in simulations for conventional PWRs, they result in predicted radiological release signatures from the fuel (i.e., not to the environment) in case of core damage, that are similar to those for conventional PWRs.

4. REFERENCES

1. R. O. Gauntt, et al., " MELCOR Computer Code Manuals," NUREG/CR-6119.

2. H. Esmaili and M. Khatib-Rahbar, "Analysis of Likelihood of Lower Head Failure and Ex-Vessel Fuel Coolant Interaction Energetics for AP1000," Nuclear Engineering & Design, Volume 235, 1583-1605 (2005).