A SYNERGISTIC USE OF CFD, EXPERIMENTS AND EFFECTIVE CONVECTIVITY MODEL TO REDUCE UNCERTAINTY IN BWR SEVERE ACCIDENT ANALYSIS

Chi-Thanh TRAN [a], Pavel KUDINOV [b]

[a] *Institute of Energy, 6 Ton That Tung, Dong Da, Hanoi, Vietnam*
[b] *Royal Institute of Technology (KTH), Roslagstullsbacken 21, SE-10691, Stockholm, Sweden*

E-mails: thanh@safety.sci.kth.se; pavel@safety.sci.kth.se

Abstract

The present paper discusses an approach to reduction of uncertainty in prediction of transient thermal loads imposed on the boiling water reactor (BWR) lower head vessel wall during a severe accident. The approach is built on synergistic use of computational fluid dynamics (CFD), experimental data and the effective convectivity model (ECM) which has been developed for simulation of melt pool heat transfer.

The paper focuses on synergistic use of detailed CFD numerical “experiments”, experimental and ECM data to identify and quantify sources of epistemic uncertainty due to modeling assumptions in the ECM. Specifically, heat transfer correlations that underlie the ECM, obtained as surface-averaged (even though implemented as spatially distributed) and derived from experiments conducted at different geometries and using fluids that are not reactor prototypical (molten corium) are examined. Previous CFD simulations have revealed so-called low fluid Prandtl number effect on local peaking of the pool’s downward heat flux for corium as working fluid. Detailed data obtained by the CFD method are implemented in the Phase-change ECM (PECM) models to examine the integral heat transfer effect on the vessel wall. The PECM is also employed to assess the influence of the experimental error uncertainty related to the downward heat transfer correlation. PECM simulation results show that the modeling uncertainty is significant factor in prediction of the integral heat transfer effects. The proposed approach offers an effective strategy for uncertainty reduction with minimal available resources.

1. INTRODUCTION

During the last years, one of the important research topics in the division of Nuclear Power Safety, KTH is to study the possibility of using the control rod guide tube (CRGT) purge flow as a severe accident management (SAM) measure for Swedish boiling water reactors (BWRs). Namely CRGTs cooled by the purge flow may help to remove effectively the decay heat from the corium melt relocated in the lower plenum and thus delay or even prevent vessel failure (Tran and Dinh, 2008) leading in the last case to in-vessel melt retention. Reliable and computationally efficient prediction of transient melt pool formation and thermal loads on the vessel is necessary for evaluation of the CRGT cooling efficiency and for determination of vessel failure modes (Rempe et al., 1993). Timing of the vessel failure and melt discharge characteristics is of paramount importance for the ex-vessel melt risk quantification in the Swedish BWR with a deep water-filled reactor cavity (Kudinov et al., 2008).

Melt pool heat transfer simulations can be performed using the CFD methods, and lumped-parameter models implemented in system codes such as RELAP/SCDAP (RELAP/SCDAP-3D, 2003), MELCOR (Gauntt et al., 2005). The high-resolution computational fluid dynamics (CFD) method is a uniquely tool capable to gain the insights into flow physics and to reveal and examine local fluid flow and heat transfer effects. However, the CFD method is computationally expensive and thus not feasible in many cases for direct application to solution of nuclear reactor safety problems, such as severe accident analysis. In contrast, simplified models which are more suitable for analysis of long transient processes often lack the capability to resolve important details, including local heat transfer effects. For example severe accident analysis codes such as RELAP/SCDAP, MELCOR, MAAP (MAAP4, 1999) are not capable to predict detailed mechanics of vessel failure modes due to limitations of lumped-parameter models for describing natural convection heat transfer of a melt pool.

Aiming at the development, validation and application of computationally efficient and sufficiently accurate tools for prediction of core melt pool heat transfer in a BWR lower head, a "five steps" approach has been proposed recently (Tran et al., 2010). The general framework is presented in Fig. 1 where the CFD method is an important tool for understanding flow physics, especially the local heat transfer effects, and generation of data for validation purpose.

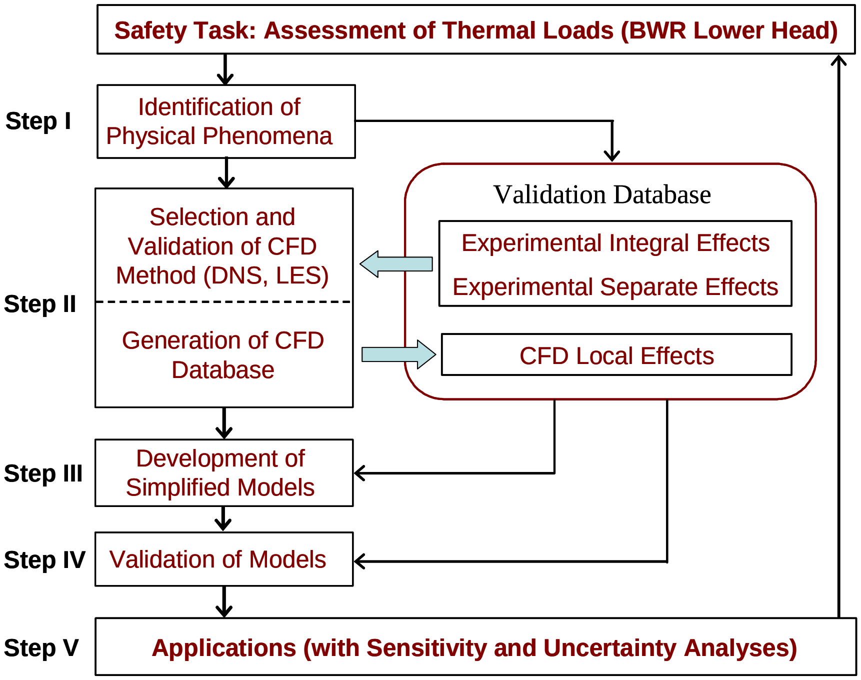


Fig. 1: A framework for development, validation and application of analysis methods for core melt pool heat transfer in a BWR lower head (Tran et al., 2010).

The effective convectivity model (ECM) and Phase-change ECM (PECM) have been developed for simulation of turbulent melt pool heat transfer in a BWR lower plenum (Tran and Dinh, 2009a, 2009b). Computational efficiency of the ECM/PECM methods is achieved by eliminating necessity of solving the full system of Navier-Stokes equations for predicting convective heat transfer. Instead the ECM is solving heat conduction equations with a source term which represents the effect of melt convection on the pool three-dimensional energy splitting. The accuracy of the ECM models is maintained by the use of the experimental heat transfer correlations which capture the effects of turbulent natural convection in a volumetrically heated liquid pool, while retaining a capability of the ECM to represent local heat transfer effects. Specifically the ECM and PECM use the directional characteristic velocities employing the Steinberner-Reineke correlations (Steinberner and Reineke, 1978) to represent turbulent heat fluxes upwards, sideward and downwards from an internally heated volume to its cooled boundaries.

Validated CFD tools are exploited in the development of the ECM for two purposes. First is to translate experimental information and correlation obtained in the tests with reduced scale, non-prototypic geometry and melt properties to the conditions of heat transfer in real geometry of BWR lower plenum. Second is to provide data for validation of the ECM, when applying to BWR specific geometry and melt properties. Insights and data obtained from the CFD simulations are used to modify ECM correlations (closures) in order to account for non-uniform spatial distribution of heat flux in a complex geometry of the lower head. Thus information about the effect of micro-scale turbulent heat transfer phenomena resolved in the CFD is implemented in the macro-scale ECM model through the modified closures. Fortunately the physics of the systems allows us to use pre-calculated closures and no two-ways coupling with feedback between micro-scale turbulent fields (resolved by CFD) and macro-scale (resolved by ECM) heat transfer characteristics is not necessary.

To take all the advantages of a commercial CFD code solver and its pre- and post processing tools, the ECM is implemented in the Fluent (Fluent, 2006). However, the effective models like ECM/PECM can be implemented and used with any other computational engines, including the advanced system codes that have 3D meshing and parabolic heat equation solver.

The ECM enables calculations of complex heat transfer phenomena during long transients thanks to its unique combination of computational efficiency and reasonable accuracy. Parametric studies based on serial simulations using higher-fidelity methods, such as CFD are not practical due to excessive computational costs. Predictive capability of the ECM and PECM has been validated against a large number of experimental and CFD generated data (Tran et al., 2010). The validation matrix covers a wide spectrum of physical phenomena involved in melt pool heat transfer: transient heat transfer including transient cooldown; turbulent natural convection; stratification; boundary layer development; flow impingement; phase change and crust formation. A wide range of Rayleigh number including reactor prototypic conditions and different fluids (Prandtl number) are considered in the validation including corium melt behavior predicted by high resolution, validated CFD method.

In the present work we address the uncertainty in ECM/PECM prediction of heat transfer characteristics in prototypical accident conditions. General approach to the uncertainty/sensitivity analysis in nuclear power safety is based on the code scaling, applicability and uncertainty (CSAU) evaluation methodology (Boyack et al., 1990; Wilson et al., 1990; Wulff et al., 1990), and the best practice guidelines proposed in NEA/CSNI/R(2007)5 report (Mahaffy et al., 2007). Followed the methodology proposed in the CSAU, a number of methods for uncertainty quantification have been developed, see IAEA Safety Reports Series 52 (D'Auria et al., 2008), among them the GRS (Glaeser, 2008), ENUSA, IPSN (D'Auria, 2006), CIAU (D'Auria, 2004) methods are widely used for uncertainty analysis of general thermal-hydraulic system codes (e.g. RELAP/SCDAP, CATHARE).

Generally, uncertainties can be classified into two types, the aleatory uncertainty which is stochastic and epistemic uncertainty which is due to imperfect knowledge. The epistemic uncertainty is quantifiable in relation with state-of-knowledge and can be reduced through enhancement of state of relevant knowledge (Theofanous, 1996; Yamaguchi et al., 2009). Aleatory uncertainties are of stochastic nature and thus cannot be reduced by increasing knowledge about phenomena. Non-reducible aleatory uncertainties can be a component of scenario uncertainty or modeling uncertainty, e.g. experimental errors in correlations. Quantification of the aleatory uncertainty requires systematic parametric studies and thus high computational efficiency of the models. Epistemic uncertainties in ECM/PECM modeling can be reduced if higher-fidelity models can be applied to understand better the underlying physical phenomena. Thorough comparative analysis of numerous sources of uncertainty in the in-vessel accident progression is beyond the scope of the present paper. The present study is concerned with three sources of uncertainty. First, the properties of molten core materials which are dependent on the extent and timing of core melting and relocation to the reactor lower plenum define heat exchange between the pool and the vessel wall. These properties define so called low Prandtl number effect which changes the local heat transfer characteristics (Nourgaliev et al., 1997; Tran et al., 2010). Second, amount of melt mass relocated in the lower plenum also depends on scenario of core damage and defines the melt pool depth and resulting thermal loads on the vessel wall (scaling effects). The third is experimental errors uncertainty which is present in the correlations employed in ECM/PECM model.

These three sources of uncertainty are considered to demonstrate application of the developed approach. We discuss results of CFD simulations which address uncertainties associated with scaling effects and corium melt *Pr* number. Then we present results of PECM sensitivity to experimental errors accounted in the correlations and results of simulations with improved models which take into account low *Pr* number effect. The proposed approach is supplementary part to the general approach to numerical simulations and analysis of severe accidents in a BWR lower head (Fig. 1).

The structure of the paper is as follows. In Section 2, the schematic diagram of the approach to uncertainty reduction is described. Section 3 focuses on using of the CFD method for sensitivity analysis and model uncertainty quantification, the main results of the CFD study and insights into flow physics are presented. Section 4 is concerned with model improvement based on the detailed local heat transfer effects obtained by the CFD method. Integral heat transfer effects are quantified with improved PECM simulations of melt pool formation and development heat transfer in a BWR lower head. The influence of the experimental errors on the integral heat transfer effect is also examined using the PECM. Section 5 summarizes the key findings and messages of the paper.

2. SYNERGISTIC USE OF CFD, EXPERIMENTS AND PECM FOR REDUCTION OF UNCERTAINTY

This section describes the approach where the high-fidelity CFD methods and experimental data are used to reduce epistemic uncertainty in the ECM/PECM modeling of heat transfer in the BWR lower plenum during accident progression. Then it is demonstrated how high computational efficiency of the ECM/PECM enables us to address aleatory uncertainty in the accident scenario.

The CFD methods are indispensable for investigating complex flow physics, can be used to perform "numerical experiments" and obtain useful data which could not be acquired experimentally. However, high-fidelity CFD simulations are very expensive, thus not directly applicable to simulation of long transients in nuclear reactor accidents. Meanwhile the accident analysis models (e.g. correlation-based methods) which are more efficient are not capable for revealing local heat transfer effects.

In this paper, an approach to effective use of the CFD method, experiments and the ECM/PECM is developed, aiming at reduction of uncertainty in BWR safety analyses (Fig. 2). The approach is based on synergistic leveraging on the inherent advantages of both the CFD methods and the accident analysis methods.

The approach focuses on Step V of the previously proposed method (Fig. 1), namely reduction of uncertainty in thermal load prediction using the ECM/PECM. In the first stage, sources of uncertainty are identified, divided into modeling uncertainty and scenario uncertainty, and ranked according to their significance and cost of reduction (as epistemic uncertainty) or quantification (as aleatory uncertainty). Decision on whether to treat particular source of uncertainty as epistemic or aleatory is conditional upon the available resources and task at hands.

For instance, the important parameters which govern melt pool heat transfer (the task at hands) are melt properties and melt pool's scales. Particular scenario of core degradation defines both melt amount and composition in deterministic way. Thus uncertainty in core degradation scenario can be considered as epistemic, which would imply then significant investment of resources into reduction of such phenomenological uncertainties as core melt heatup, oxidation, progression and relocation to the lower plenum, boiling heat transfer in porous media, melt chemical interaction, vessel penetrations failure etc. However, the influence of melt amount and composition can be quantified at much less cost if these parameters are considered as aleatory parameters. Therefore we put detailed analysis of core degradation phenomena beyond the scope of the paper and use outcomes of these phenomenological processes (melt amount and properties) as uncertain aleatory parameters.

Another source of modeling uncertainty considered in this work is experimental correlations which are used for modeling of natural convection heat transfer in the ECM/PECM. The uncertainty in prediction of integral heat transfer effect (i.e. the predicted thermal loads) can be significant if the experimental error uncertainty in the correlations is large. It is obviously not feasible to redo all experiments trying to reduce uncertainty in the correlations. Instead we use parametric study to assess and quantify sensitivity of integral thermal loads to the uncertainties in the correlations. Thus we treat this component in ECM/PECM modeling uncertainty as aleatory (quantifiable, but not reducible).

The local heat transfer effects can be in principle affected by the scenario parameters, namely melt property, i.e. the corium melt *Pr* number, and pool scales (see Fig. 2). In the present work we consider the uncertainty in local heat transfer phenomena as reducible at affordable price and therefore we decide to treat it as epistemic uncertainty in modeling.

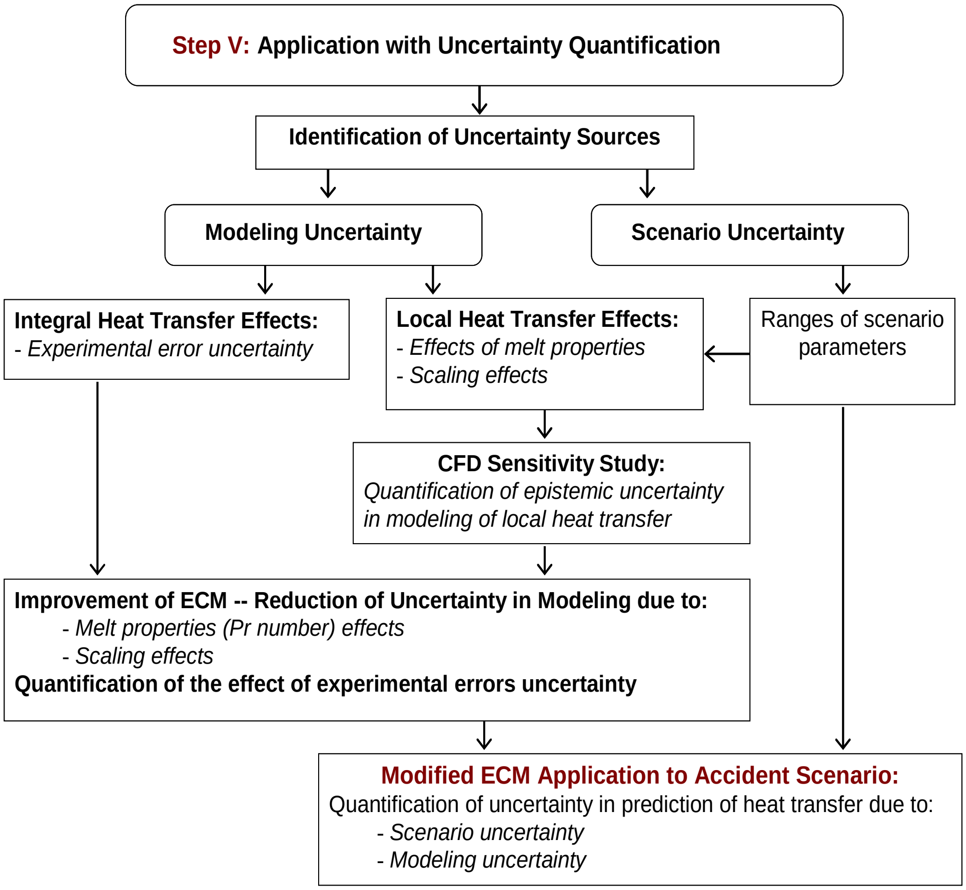


Fig. 2: Schematic diagram of the uncertainty reduction approach applied in prediction of transient thermal loads on BWR lower head vessel wall.

At the second stage (Fig. 2), the CFD method is employed to perform sensitivity analysis with the melt property parameter and melt pool depth ranges of interest. Thus the reliable CFD method is used for quantification of epistemic uncertainty in local heat transfer modeling. The model bounds are identified and quantified. The gained insights are used for improvement of the ECM.

In the next stage, the modified ECM/PECM tools are used for quantification of the influence of (i) experimental error uncertainty on the integral heat transfer effects, and (ii) local heat transfer effect uncertainty (due to the *Pr* number effect and scale effect). The ECM and PECM are the tools which allow us to study the influence of local heat transfer effect on the resulting integral thermal loading due to melt pool heat transfer.

Finally, the last block of the diagram presents application of the improved ECM/PECM, the uncertainty in thermal load prediction for BWR lower head is quantified. The proposed approach presents a synergetic use of the CFD method, in combination with the other tools to offer an effective way to reduce uncertainty in severe accident analysis.

3. SENSITIVITY ANALYSIS USING CFD METHOD

In the present work, the selected CFD method is the implicit large eddy simulation (ILES) (Margolin et al., 2006). The ILES method employs high resolution grids especially in the near wall regions to effectively provide large eddy simulations without sub-grid scale (SGS) models. The ILES method was validated against various experiments (Tran et al., 2010).

Considering a melt pool in the BWR lower plenum in the presence of CRGT cooling, it is seen that the BWR lower plenum can be divided into unit volumes. A unit volume is a rectangular cavity filled with corium melt, containing a CRGT and cooled at the top and bottom boundaries and on the CRGT surfaces (Fig. 3). It is assumed that the flow pattern and heat transfer in the unit volume are representative for the whole lower plenum. The boundary conditions applied for a unit volume are as follows. The top, bottom and CRGT walls are isothermal (corium melt liquidus temperature of 2770K is applied). On the vertical side surfaces of the unit volume symmetric adiabatic conditions are applied. Simulation start with uniform distribution of initial temperature which is liquidus temperature of the corium melt. All obtained ILES solutions are grid-dependent with about a million of computational cells.

The CFD ILES simulations in unit volumes revealed the local heat transfer effect which is attributed for corium melt (Tran et al., 2010), and is called low Prandtl number effect (Nourgaliev et al, 1997). That is, the downward heat flux from a melt pool to the vessel wall is locally enhanced, in the vicinity of CRGT which is cooled by water flow from the inside. The enhanced heat flux is significant, about 5-7 times higher than that of the peripheral areas where the heat flux is consistent with semi-empirical (analytical) model (Fig. 4). The analytical model is based on the heat transfer coefficients along the cooled boundaries (Steinberner-Reineke correlations) and the unit volume geometry (Tran and Dinh, 2009a). This effect is related to boundary layer flow development along the cooled CRGT and flow impingement on the bottom wall.

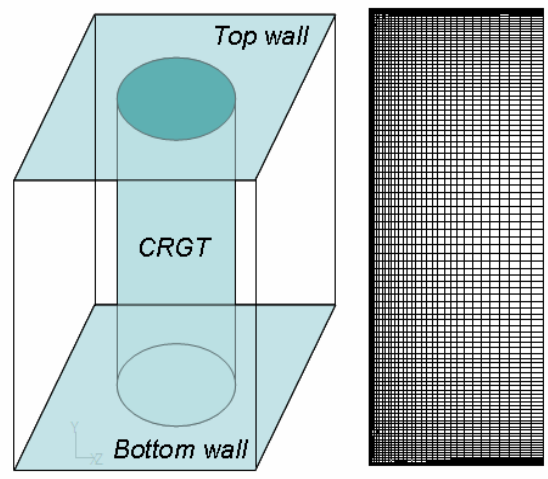


Fig. 3: Unit volume geometry and grid resolution along the cooled boundaries.

During the process of melt pool formation, the depth of melt pool can be increased as it depends on the mass of the relocated melt. Furthermore, in a debris bed, melt pools are formed gradually, further development of melt pools results in increasing of melt pool depth. It is therefore important to quantify the low *Pr* number effect on the downward heat flux for different pool's depths. CFD simulations performed for three different values of the pool depth (0.3 m; 0.4 m and 0.6 m) show that the enhancement of the downward heat flux is less sensitive to the pool depth, and consequently to internal Rayleigh number (Fig. 5). Therefore, this scaling effect will not be taken into account in the local downward heat transfer coefficient model for improvement of the ECM/PECM.

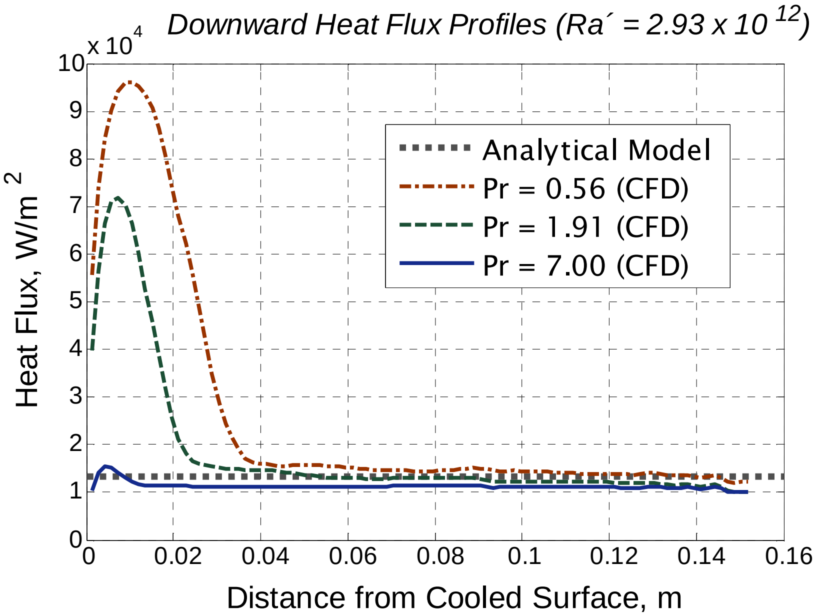


Fig. 4: Downward heat flux profiles predicted for three different fluid *Pr* numbers (Tran et al., 2010).

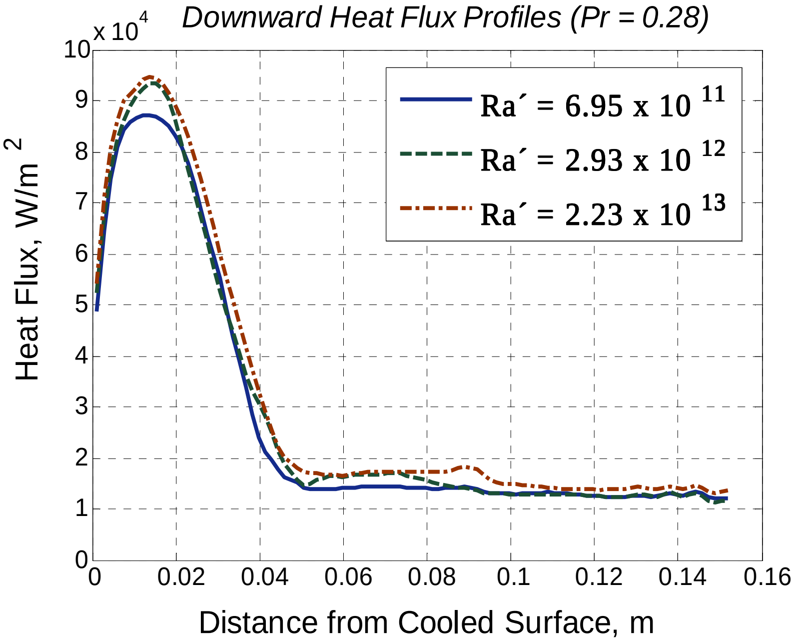


Fig. 5: CFD predicted downward heat flux profiles for different melt pool depths (different internal Rayleigh numbers).

Prandtl number is the control parameter in the heat flux enhancement phenomenon. Clearly, dependent on the accident scenario, the homogeneous corium melt may have different fraction of metallic components. As a consequence, the corium *Pr* number is changed, largely is due to different thermal conductivities of the metallic components and oxidic melt. Thermal conductivity of the corium melt can be calculated as follows:

$$k^{cm} = \sum_{i=1,\ j=1}^{n,m} \left(f_i^{ox} \times k_i^{ox} + f_j^{ml} \times k_j^{ml} \right) \qquad (1)$$

Where k^{cm} is the homogeneous corium melt conductivity; f_i^{ox} and k_i^{ox} are the mass fraction and conductivity of oxidic *i*-component; f_j^{ml} and k_j^{ml} are the mass fraction and conductivity of metallic *j*-component. The value of the corium *Pr* number depends on fraction of metallic components. The calculation based on data of a reference ABB-Atom BWR design shows that the corium *Pr* number may vary from the lowest value of 0.1 to the highest value of 0.6. To quantify the low *Pr* number effect with different *Pr* values, CFD simulations are performed for three different *Pr* number in the range of interest (*Pr* = 0.12; 0.28 and 0.56). Respective values of melt viscosity and thermal conductivity are: for *Pr* = 0.56, μ=0.0046 [Pa·s] and k=3.95 [$W \cdot K^{-1} \cdot m^{-1}$]; for *Pr* = 0.12, μ=0.003 [Pa·s] and k=12 [$W \cdot K^{-1} \cdot m^{-1}$].

Three profiles of the downward heat flux obtained from CFD simulations for a unit volume of 0.4m height are presented in Fig. 6. It can be seen, the local peaked heat flux is lower for the lower bounding value of Prandtl number, however the area where heat flux enhancement is observed continues to grow when *Pr* decreases (Fig. 4, Fig. 6). In the vicinity of the CRGT, the peaked heat flux is about 7 times higher than that in the peripheral area for *Pr* = 0.56, while for *Pr* = 0.12, the peaked heat flux is about 5 times higher than that in the peripheral area. Note that the heat flux value in the peripheral area (about 13 KW/m^2) remains unchanged and consistent with that of the analytical model.

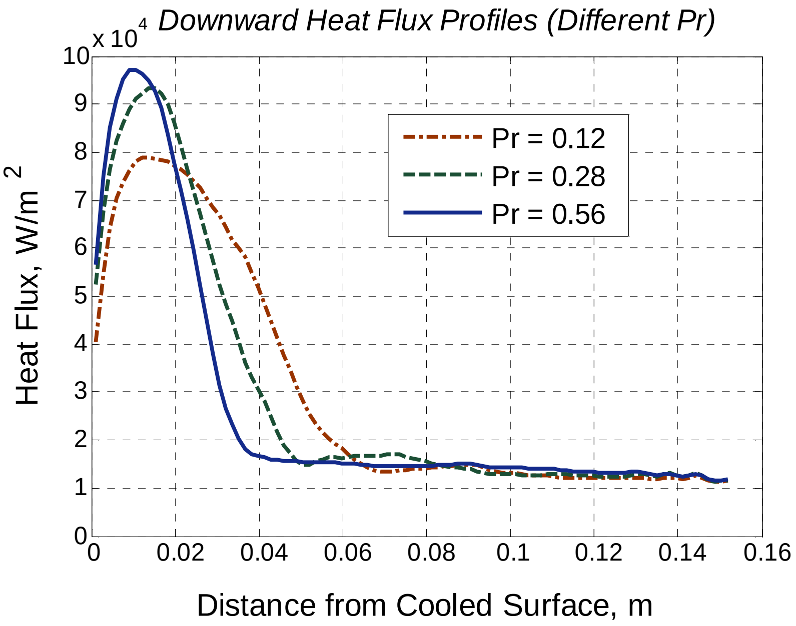
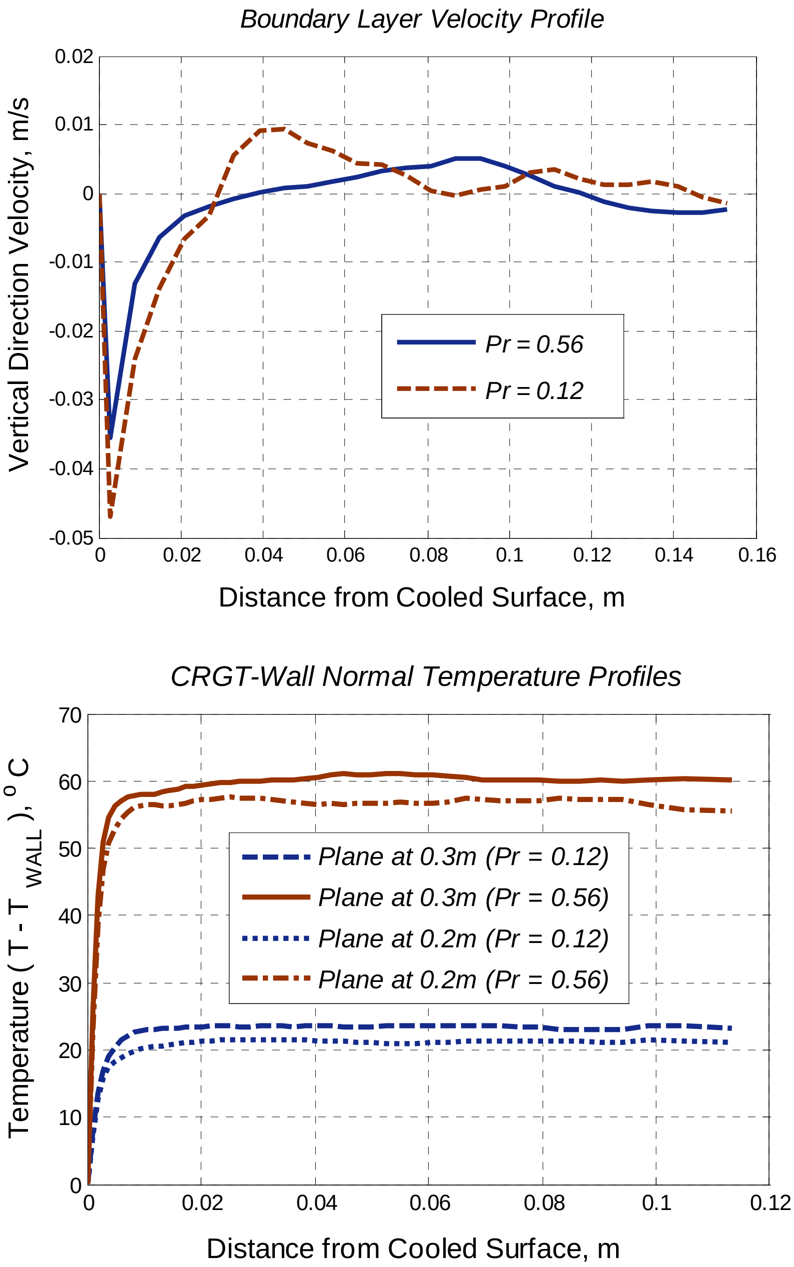


Fig. 6: Downward heat flux profiles for different melt property (different fluid Prandtl numbers).

Apparently the phenomenon of non-monotone influence of *Pr* number on the peaking value of downward heat flux (Fig. 4; Fig. 6) is a result of non-linear interplay between different factors. On the one hand, when *Pr* number is decreasing due to increasing melt thermal conductivity and decreasing viscosity a thicker and higher velocity boundary layer of gravity driven descending flow is formed along the CRGT (Fig. 7a). The thickness of thermal boundary layer grows by 50% when *Pr* decreases from 0.56 to 0.12 (Fig. 7b) suggesting that contributions of turbulence and thermal conduction to the total heat transfer are comparable. The descending flow of cold melt also entrains hot liquid from the ambient pool (Fig. 7a, b). When the descending flow impinges on the bottom wall, it leads to intensification of the convective heat transfer in the vicinity of the vertical surface of cooled CRGT (see Fig. 8). Higher inertia of the downward flow in case of Pr=0.12 leads to increase of the area where intensification of the convective heat flux is observed. On the other hand, in the stably stratified bottom layer, temperature of the entrained (bulk) liquid is reduced mostly because of the conduction, which is known as “alpha phenomenon” (Nourgaliev et al., 1997), melt superheat and thus difference between ambient and wall temperature decreases (Fig. 7a, b; Fig. 9) when Pr decreases. More detailed study of local flow and heat transfer conditions in the vicinity of flow impingement on the bottom wall is necessary to reach better understanding of these nonlinear phenomena.



a)

b)

Fig. 7: CFD simulation, profiles: *a)* Velocity magnitude at the middle horizontal plane of unit volume (vertical component of velocity); *b)* CRGT-wall normal temperature profiles at different elevations. T_{WALL} is wall temperature.

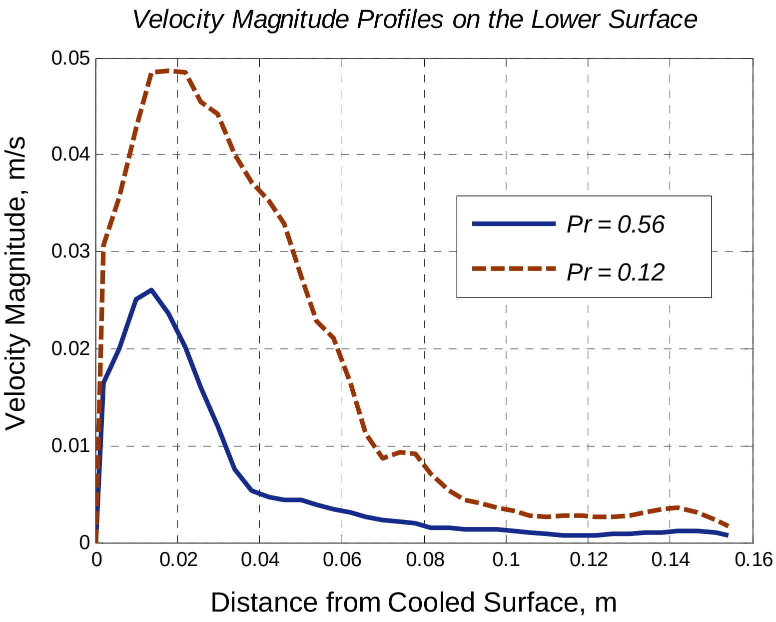


Fig. 8: CFD simulation: Fluid velocity magnitude radial distribution above the bottom surface (average values at elevations of 1mm and 2 mm above the bottom wall).

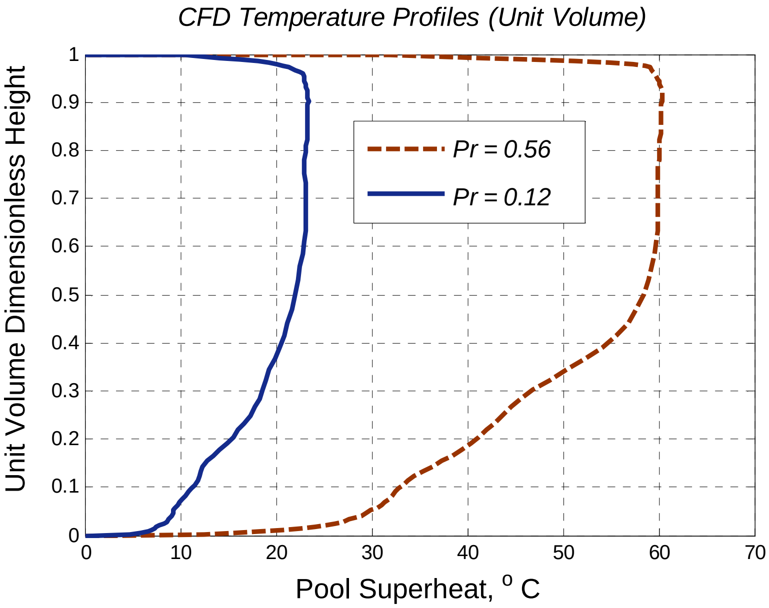


Fig. 9: CFD simulation: Bulk fluid temperature distribution along vertical direction (*Pr* = 0.56 and *Pr* = 0.12).

Fig. 8 shows the fluid velocity magnitude above the cooled bottom surface. It can be seen that in the case of *Pr* = 0.12, the fluid is moving faster, over a broader area on the bottom surface, due to higher flow rate and inertia of the downward impinging flow. Therefore, the lower peaked value and broader area of the enhanced downward heat flux in the case of *Pr* = 0.12 is a result of a less amount, and a larger spreading area of the entrained hot liquid accumulated in the unit volume lower region.

To demonstrate the synergy of methods presented in the approach, in the next section, the PECM tool is employed to examine the influence of the local heat transfer effects revealed by CFD method, on the integral heat transfer effect, i.e. the transient thermal loads on the vessel wall.

Two bounding profiles of the downward heat fluxes obtained by the CFD (Fig. 6) are implemented in the ECM/PECM for the downward Nusselt number model (Fig. 10). The lower bounding profile is for Pr = 0.1, the upper bounding profile is for Pr = 0.6. For the lower bounding profile, the peaking value of the downward Nusselt in the vicinity of the CRGT (Nu^{max}) is 5 times higher than peripheral Nusselt number which is calculated by the downward Steinberner-Reineke correlation (Nu^{min}). For the upper bounding profile, Nu^{max} is 7 times higher than Nu^{min}. The area where Nu is enhanced for the upper bounding case is narrower than that of the lower bounding case, according to the obtained with CFD data. The linear profile is applied for the Nusselt in the area where it changes from Nu^{max} to Nu^{min}.

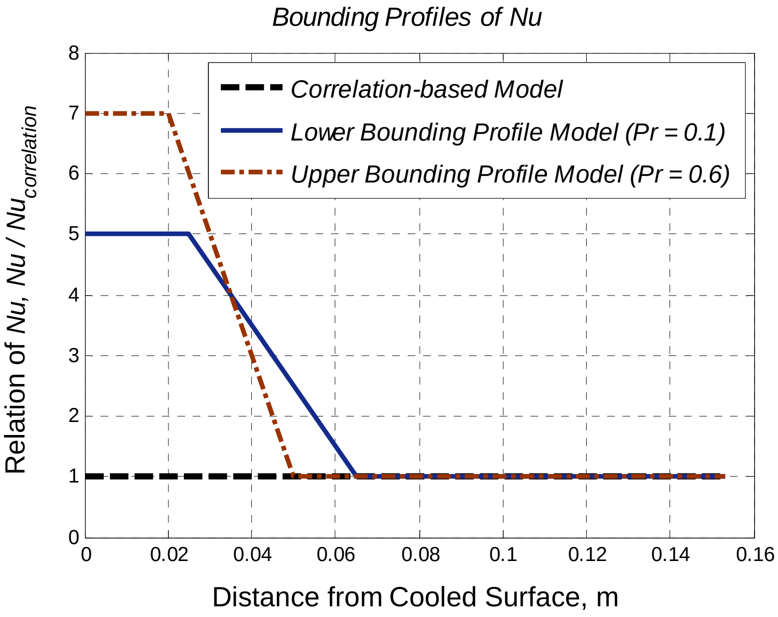


Fig. 10: Different models of the downward Nusselt number implemented in the ECM.

4. PECM SIMULATION TRANSIENT THERMAL LOADS ON THE BWR LOWER HEAD VESSEL WALL

In this section, the PECM is applied for heat transfer simulations of melt pool formation in the BWR lower plenum. Thermal loads on the lower head vessel wall and subsequent transient vessel wall temperature are calculated. 3D slice geometry of the ABB-Atom BWR lower plenum is used for PECM simulations (Fig. 11). The slice is a segment of lower plenum, full of corium, containing 6 cooled CRGTs. Simulations with 3D geometry ensure that the local heat transfer effects are preserved. The corium inside the slice is connected below with the vessel wall which is 182 mm thick.

We assume that a debris bed of 0.7m thick (about 25 tons of melt) is formed in the lower plenum. Due to inadequate cooling, the debris bed is heated up and remelted. Melt pools are forming in the midst of cooled CRGTs inside the debris bed, developing and merging together to form a common melt pool which is assumed to be homogeneous. It is assumed that water is spilled out of CRGTs and is available on top of the debris bed. Therefore, the boundary conditions applied to the slice are as follows. The debris bed top, CRGT surfaces are isothermal with boiling temperature at 3 bars. The vessel external surface is insulated, therefore, a small heat flux is allowed (about 20 W/m^2). Other boundaries are either adiabatic or symmetric. Simulations are performed with the assumption that instrumentation guide tubes are intact during the accident progression, do not influence melt pool heat transfer.

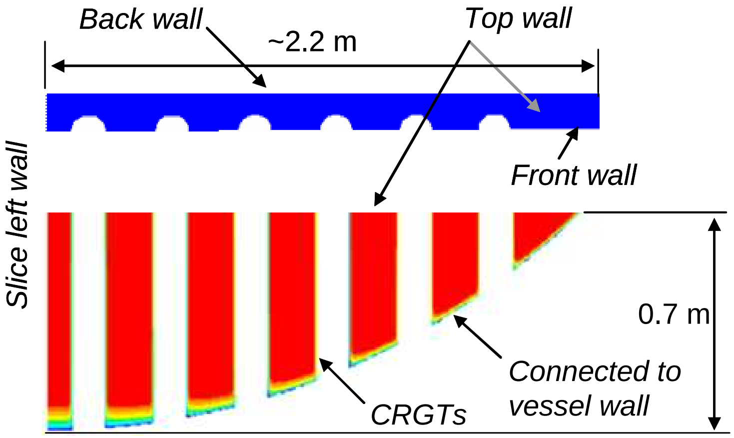


Fig. 11: 3D slice geometry of the ABB-Atom BWR lower plenum ($H = 0.7$ m).

Table 1 shows the PECM simulation matrix which includes sensitivity analysis to the experimental errors (15%) and uncertainty analysis with improved models based on the insights gained from the CFD sensitivity analysis data. Different downward heat transfer coefficient models defined for PECM calculations are as follows. In the base case, namely the correlation-based model, the downward heat transfer coefficient is calculated using the original downward Steinberner-Reineke correlation. In the other cases, the downward heat transfer coefficient is modified according to the models described in the table.

Table 1: PECM simulation matrix with different models of downward Nusselt number

Type of analysis	Downward heat transfer coefficient model
Sensitivity analysis with experimental errors (±15%)	85% of downward Steinberner-Reineke correlation
	100% of downward Steinberner-Reineke correlation
	115% of downward Steinberner-Reineke correlation
Uncertainty analysis with improved models (based on the insights gained from CFD study)	Lower bounding *Nu* profile ($Pr = 0.1$)
	Correlation-based model (no modification of *Nu*)
	Upper bounding *Nu* profile ($Pr = 0.6$)

Results of the PECM simulations according to the simulation matrix show that integral energy splitting in the formed melt pool is insensitive to different models of the downward heat transfer coefficient. The transient upward heat flux from the debris bed (and melt pool later on) to the top cooled surface where water is available is nearly identical to the sensitivity analysis (with experimental errors of 15%). Even with different *Nu* profile models the transient heat flux is merely changed, about 1.6% (Fig. 12). PECM simulations for different *Nu* profile models show that the transient surface-averaged heat fluxes to the cooled CRGTs for 12.5 hours are not significantly changed (Fig. 13). The difference between the base case (correlation-based *Nu*) and upper bounding profile case is not larger than 10%.

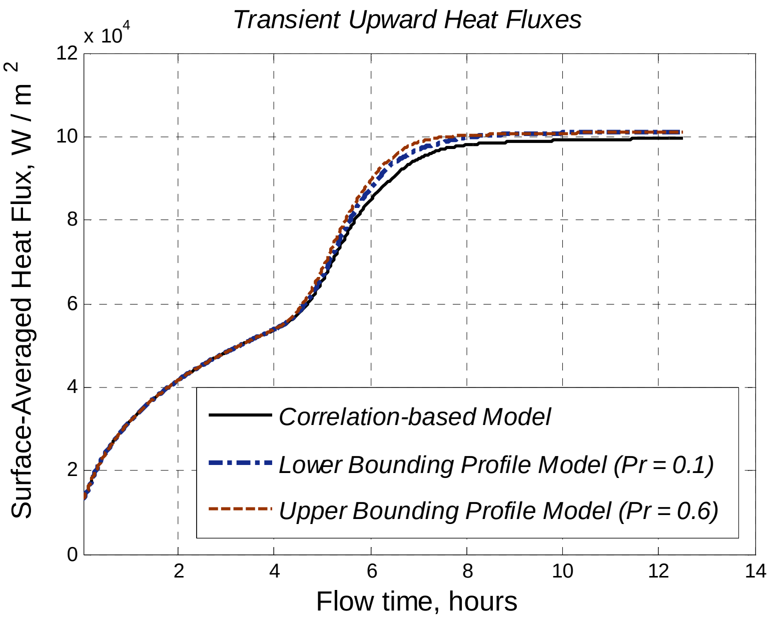


Fig. 12: Transient surface-averaged heat flux to the top cooled surface for different models of downward Nusselt number.

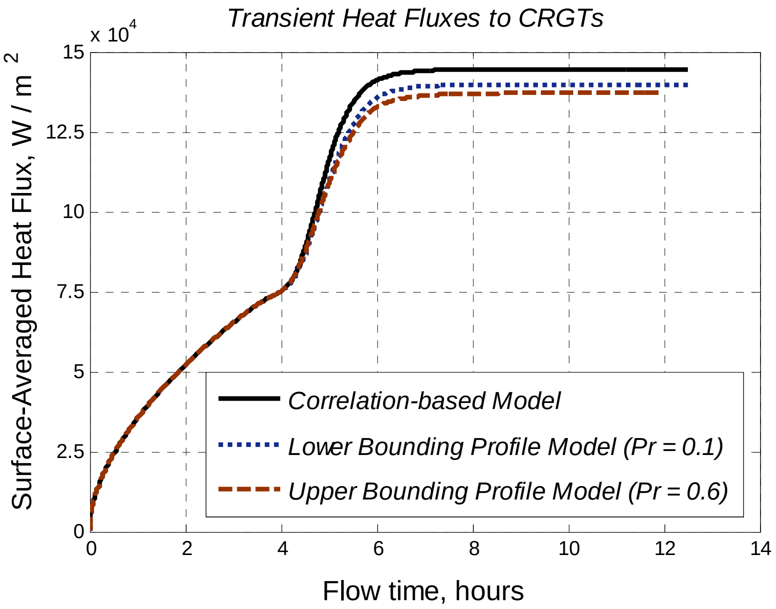


Fig. 13: Transient surface-averaged heat flux to CRGTs for different models of downward Nusselt number.

There is significant influence of different *Nu* profile models on the transient downward heat flux to the vessel wall (Fig. 14). It can be seen that there is very small influence of possible 15% experimental error implemented in the PECM correlations on the transient downward heat flux, i.e. the thermal loads to the vessel wall. However, different models of the downward heat transfer coefficient profile implemented in the PECM result in a significant change of the transient downward heat flux. The upper bounding profile model (*Pr* = 0.6) increases the heat flux to the vessel wall about 30% compared with the base case (correlation-based model). The lower bounding profile model (Pr = 0.1) also results in 20% increase of the downward heat flux in the later time period.

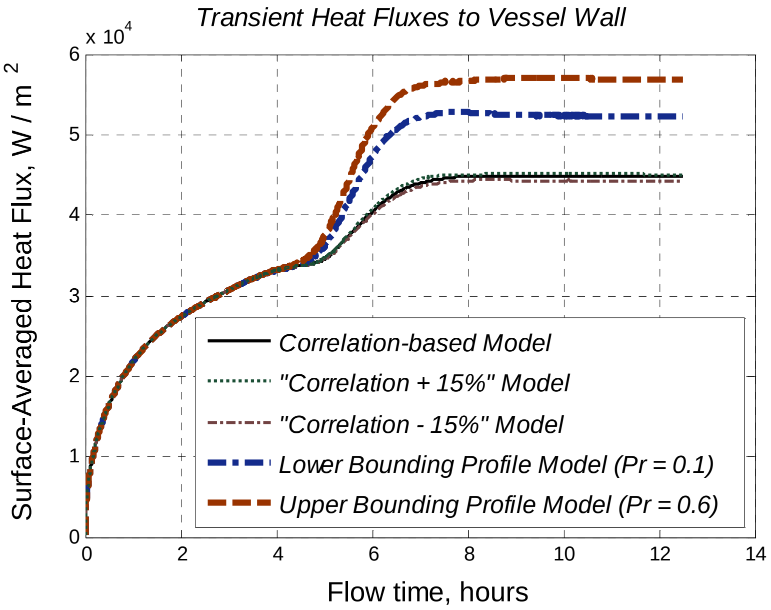


Fig. 14: Transient surface-averaged heat fluxes to the vessel wall for different models of downward Nusselt number (thermal loads to the vessel wall).

Apparently, the integral heat flux imposed on the vessel wall from the melt pool through the crust surrounding the melt pool is sensitive to the local distribution of downward heat flux and sideward heat flux (due to inclination of the vessel wall). The crust thickness distribution (Fig. 15) depends on the distribution of heat fluxes imposed on the crust boundary. Thus different models of the downward heat transfer coefficients implemented in the ECM/PECM result in different dynamics of the local crust thickness.

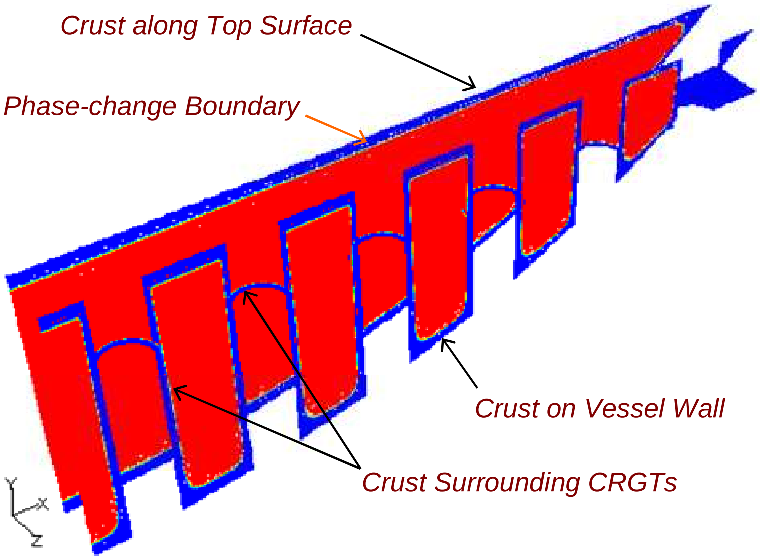


Fig. 15: Melt pool and its surrounding crust in the BWR lower head.

Explanation of the difference in predicted thermal loads in the lower and upper bounding cases compared with the base case (Fig. 14) is as follows. First of all, it can be seen in Fig. 10 that localized enhancement of the heat transfer in the vicinity of the CRGT surface leads to increase of the average Nusselt for the upper and lower bounding profiles in comparison with that of the base case. Second, due to the peaked value of the Nusselt, the thinning of the crust in the vicinity of cooled CRGT wall is faster in the upper and lower bounding cases than in the base case. Crust thinning causes slight redistribution in the energy splitting between vertical and horizontal directions. Larger amount of heat is transferred to the bottom vessel wall. Consequently the vessel wall is heated to higher temperature as shown in Fig. 14.

Note that two different models of *Nu* (i.e. upper and lower models) also result in different transient heat flux, although the difference between them is about 10%. The difference between the cases calculated with experimental errors (±15%) and with base case is only about 1.5%.

For the preliminary evaluation of vessel wall creep failure (failure mode and timing), the PECM calculated vessel wall temperature profiles along polar angle are plotted in Fig. 16.

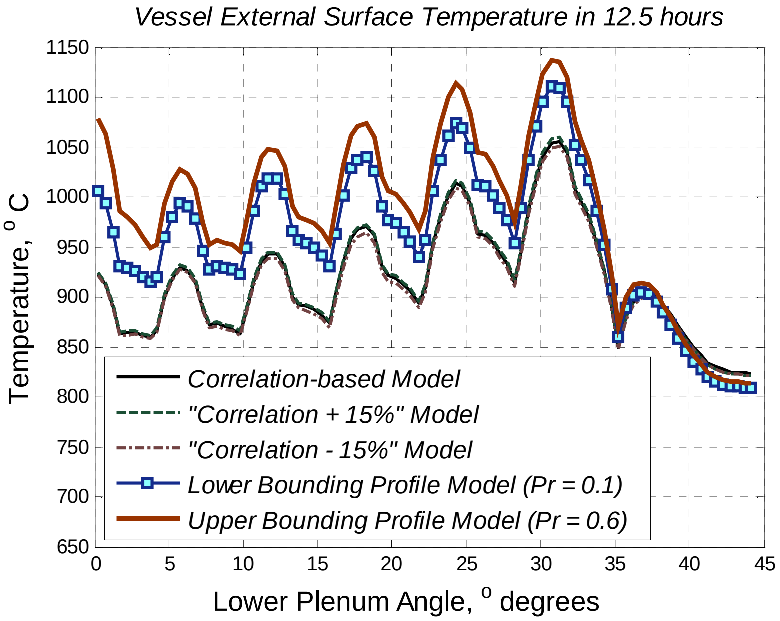


Fig. 16: PECM predicted vessel wall temperature profiles along the lower plenum polar angle for different models of the downward *Nu* at 12.5 hours.

As it is shown in Fig. 16, the influence of experimental error uncertainty is negligible for the vessel wall temperature. However, different models of the downward *Nu* profiles significantly affect the vessel wall temperature. The temperature obtained with the upper and lower bounding *Nu* profile models are 80-150^{o}C higher than that of the base case. According to Rempe et al. (1993) thermal creep in the vessel steel may take place at temperature of about 1100^{o}C. In the cases predicted with the models for enhanced downward heat flux, vessel temperature exceeds the creep limit, suggesting vessel failure in about 8 hours since the start of the debris bed heating up in the lower plenum (Fig. 17). Further study on the vessel wall creep and failure using the PECM predicted transient thermal loads is necessary to determine the vessel failure mode and timing.

Plots of the maximum vessel wall temperature predicted with different models show that the vessel wall temperature exceeds the creep limit at much earlier time, at about 8 hours after the start of debris bed heatup if the upper and lower bounding *Nu* profiles models are used, while for the base case vessel wall temperature is lower than creep limit after 12 hours.

Comparison of transient temperature profiles in two cases at different time moments (Fig. 18) shows that the maximum temperature of the vessel wall predicted with the upper bounding *Nu* profile model reaches at 7.8h the same level as predicted in the base case model at 12.5h (see Fig. 17). The difference in time is about 5 hours. This implies that additional severe accident management (SAM) measures (e.g. external cooling for the BWR lower plenum) have to be activated at much earlier time to be most effective.

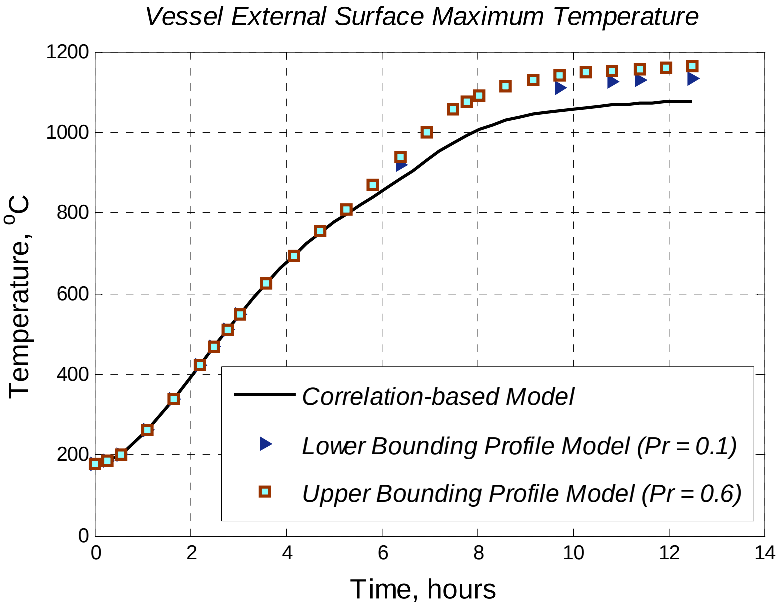


Fig. 17: Transient maximum vessel wall temperature for different models of downward Nusselt number.

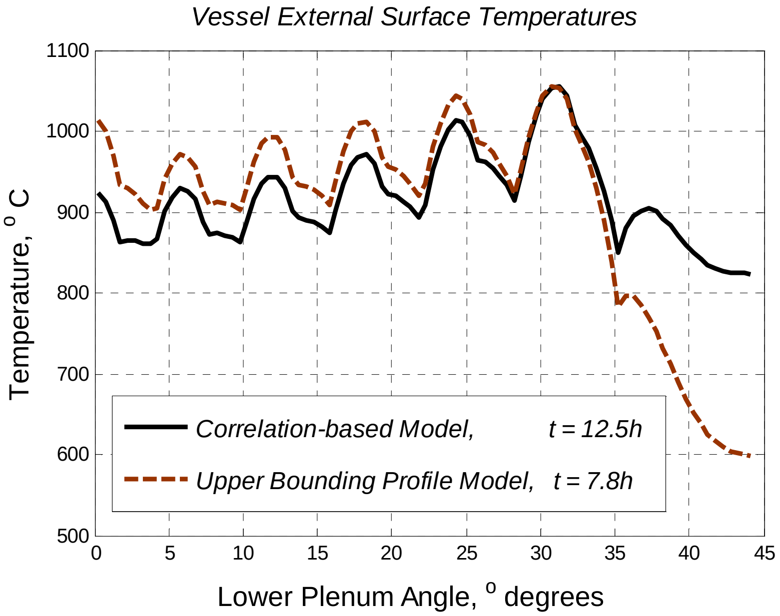


Fig. 18: Transient vessel wall temperature profiles for different models of downward Nusselt number.

5. CONCLUDING REMARKS

The present work proposes an approach to uncertainty reduction in boiling water reactor (BWR) severe accident analysis by synergistic use of computational fluid dynamics (CFD) methods, experiments, and the effective convectivity model (ECM) (Tran and Dinh, 2009a, 2009b). The present study is a next step in development of the previously proposed approach to numerical simulations and analysis of molten corium coolability during a severe accident in a BWR (Tran et al., 2010). In the framework of this approach the effective convectivity model (ECM) makes use of experimental heat transfer correlations to capture the effect of turbulent natural convection in a volumetrically heated liquid pool, while retaining the pool three-dimensional energy splitting and employing insights from CFD modeling to represent 3D local heat transfer effects.

Three sources of uncertainty in prediction of melt pool heat transfer and thermal loads on the vessel wall are considered in present work: (i) uncertainty in melt properties (*Pr* number) and melt pool scales; (ii) modeling uncertainty due to the experimental error in correlations used in the ECM/PECM; (iii) uncertainty in local heat transfer effects due to melt properties (*Pr* number) and pool scales. First two sources are treated as aleatory (non-reducible) uncertainties. Parametric studies are used to quantify the influence of these two sources of uncertainties. The uncertainty in modeling local heat transfer is considered as epistemic (reducible). Detailed CFD modeling is used to perform "numerical experiments" and reduce the uncertainty by developing more accurate models which take into account melt material properties. CFD study revealed that the modeling uncertainty due to scale of a formed melt pool is negligible while the influence of corium melt Prandtl number is significant.

The improved models are implemented in the phase-change ECM (PECM) to examine the effects of the modeling uncertainty on the transient thermal loads imposed on the BWR lower head vessel wall. The sensitivity analysis on the experimental errors is performed with the PECM tool. Results of the sensitivity study for the thermal loads on the vessel wall show that the uncertainty in the ECM correlations due to experimental errors plays a minor role in comparison with the uncertainty due to modeling of the local heat transfer effects. Calculations with the improved models of the downward *Nu* show that vessel wall temperature increases faster which can cause earlier failure of the vessel. Additional SAM measure such as external vessel wall cooling is required to increase the probability of in-vessel retention for Swedish BWRs.

The reliable CFD methods are indispensable tools for examining and better understanding of flow physics. Data produced by CFD is used for improvement of "effective" models to reduce epistemic uncertainty in the accident analysis of BWR lower head heat transfer. The proposed approach provides guidance towards an effective strategy for uncertainty reduction and quantification with optimal use of available resources.

ACHNOWLEDGEMENT

The present work is supported by the Swedish Nuclear Radiation Protection Authority (SSM), Swedish Power Companies, European Commission (SARNET-2), Nordic Nuclear Safety Program (NKS), and Swiss Federal Nuclear Safety Inspectorate (ENSI) under the "Accident Phenomena of Risk Importance" (APRI) – "Melt-Structure-Water Interactions" (MSWI) program at the Division of Nuclear Power Safety, Royal Institute of Technology (KTH), Stockholm, Sweden. The authors are grateful to Professor Nam Dinh for his insights, his guides, valuable discussions and comments.

NOMENCLATURE

Arabic

f	Fraction
g	Gravitational acceleration, m/s^2
H	Height of a volume or fluid layer, depth of a melt pool, m
k	Thermal conductivity, W/(m.K)
Nu	Nusselt number, $Nu = \frac{qH}{k\Delta T}$
Pr	Prandtl number, $\Pr = \nu / \alpha$
q	Heat flux, W/m^2
Q_v	Volumetric heat source, W/m^3
Ra'	Internal Rayleigh number, $Ra' = \frac{g\beta Q_v H_{pool}^5}{k\nu\alpha}$
T	Temperature, K

Greek

α	Thermal diffusivity, m^2/s, $\alpha = \frac{k}{\rho . C_p}$
β	Thermal expansion coefficient, 1/K
ΔT	Temperature difference, K
ν	Kinematics viscosity, m^2/s

Subscripts and superscripts

pool	Pool
max, min	Maximum, Minimum
cm	Corium melt
ox	Oxide
ml	Metal

REFERENCES

B.E. Boyack et al., “Quantifying Reactor Safety Margins. Part 1: An Overview of the Code Scaling, Applicability, and Uncertainty Evaluation Methodology”, *J. Nuclear Engineering and Design*, **119**, 1-15 (1990).

F. D’Auria, “Approach and Methods to Evaluate the Uncertainty in System Thermalhydraulic Calculations”, *Mecanica Computacional*, Bariloche, Argentina, XXIII (2004).

F. D’Auria, A. Bousbia-Salah, A. Petruzzi and A. Del Nevo, “State of the Art in Using Best Estimate Calculation Tools in Nuclear Technology”, *Nuclear Engineering and Technology*, **38**, 11-32 (2006).

F. D’Auria et al., “Best Estimate safety Analysis for Nuclear Power Plants: Uncertainty Evaluation”, *IAEA Safety Reports Series*, **52** (2008).

Fluent Inc., Fluent User’s Guide, 2006.

Gauntt, R.O., Cash, J.E., Cole, R.K., Erickson, C.M., Humphries, L.L., Rodriguez, S.B. and Young M.F., “MELCOR Computer Code Manual, Core (COR) Package Reference Manuals”, NUREG/CR-6119, 2 (2), (2005).

H. Glaeser, “Results from the Application of Uncertainty Methods in the CSNI Uncertainty Methods Study (UMS)”, *Proc. of THICKET-2008*, Pisa, Italy, May 5-9 (2008).

P. Kudinov, A. Karbojian, W.M. Ma and T.N. Dinh, “An Experimental Study on Debris Formation with Corium Simulant Materials”, *Proc. of 2008 International Congress on Advances in Nuclear Power Plants (ICAPP’08)*, Anaheim, California, USA, June 8-12 (2008).

“MAAP4 Users Manual”, Fauske Associated Inc., 2 (1999).

J. Mahaffy et al., “Best Practice Guidelines for the Use of CFD in Nuclear Reactor Safety Applications”, *Nuclear Energy Agency Report*, NEA/CSNI/R(2007)5 (2007).

L.G. Margolin, W.J. Rider, F.F. Grinstein, “Modeling Turbulent Flow with Implicit LES”, *Journal of Turbulence*, **7 (15)**, 1-27 (2006).

Nourgaliev, R.R., Dinh, T.N. and Sehgal, B.R., “Effect of Fluid Prandtl Number on Heat Transfer Characteristics in Internally Heated Liquid Pools with Rayleigh Numbers up to 10^{12}. *J. Nuclear Engineering and Design*, **169**, 165-184 (1997).

J.L. Rempe, S.A. Chavez, G.L. Thinnes, C.M. Allison, G.E. Korth, R.J. Witt, J.J. Sienicki, S.K. Wang, L.A. Stickler, C.H. Heath, S.D. Snow, “Light Water Reactor Lower Head Failure Analysis”, NUREG/CR-5642, EGG-2618, Idaho National Engineering Laboratory, USA (1993).

SCDAP/RELAP5-3D Code Development Team, “SCDAP/RELAP5-3D© Code Manual”, Report INEEL/EXT-02-00589, Revision 2.2, Idaho National Engineering and Environmental Laboratory, USA (2003).

U. Steinberner and H.H. Reineke, “Turbulent Buoyancy Convection Heat Transfer with Internal Heat Sources”, *Proc. of the 6th Int. Heat Transfer Conference*, Toronto, Canada, Vol.2, 305-310 (1978).

T.G. Theofanous, “On the Proper Formulation of Safety Goals and Assessment of Safety Margins for Rare and High-consequence Hazards”, *Reliability Engineering & Systems Safety*, **54**, 243-257 (1996).

C.T. Tran and T.N. Dinh, “Application of the Phase-change Effective Convectivity Model to Analysis of Core Melt Pool Formation and Heat Transfer in a BWR Lower Head”, *Transactions of ANS 2008 Annual Meeting*, Anaheim, California, USA, June 8-12 (2008).

C.T. Tran and T.N. Dinh, “The Effective Convectivity Model for Simulation of Melt Pool Heat Transfer in a Light Water Reactor Lower Head. Part I: Physical Processes, Modeling and Model Implementation”, *J. Progress in Nuclear Energy*, **51 (8)**, 849-859 (2009a).

C.T. Tran and T.N. Dinh, “The Effective Convectivity Model for Simulation of Melt Pool Heat Transfer in a Light Water Reactor Lower Head. Part II: Model Assessment and Application”, *J. Progress in Nuclear Energy*, **51 (8)**, 860-871 (2009b).

C.T. Tran, P. Kudinov and T.N. Dinh, “An Approach to Numerical Simulation and Analysis of Molten Corium Coolability in a Boiling Water Reactor Lower Head”, *J. Nuclear Engineering and Design*, **240 (9)**, 2148-2159, 2010.

G.E. Wilson et al., “Quantifying Reactor Safety Margins. Part 2: Characterization of Important Contributors to Uncertainty”, *J. Nuclear Engineering and Design*, **119**, 17-31 (1990).

W. Wulff et al., “Quantifying Reactor Safety Margins. Part 3: Assessment and Ranging of Parameters”, *J. Nuclear Engineering and Design*, **119**, 33-65 (1990).

A. Yamaguchi, S. Mizokami, Y. Kudo, A. Hotta, “Uncertainty and Conservatism in Safety Evaluations Based on BEPU Approach”, *Proc. of the 13th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-13)*, Kanazawa, Japan, September 27 – October 02 (2009).