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Thermal Hydraulic Transient Analysis of the High Performance Light Water Reactor Using APROS and SMABRE

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1 ABSTRACT

A preliminary thermal hydraulic transient analysis for the High Performance Light Water Reactor (HPLWR) was calculated using the thermal-hydraulic system codes APROS and SMABRE. Effect of the used supercritical-pressure heat transfer correlation on the simulation results was examined with the APROS code in steady-state and during the transient. The transient analysis revealed a need for improving the proposed low-pressure residual heat removal system of the HPLWR, which was found inadequate to keep the reactor core intact in the case of a main steam line break accident with a guillotine break. The uncertainty caused by the choice of the supercritical-pressure heat transfer correlation was noted to have virtually no effect in this transient, because of the very fast drop to subcritical pressures. In a transient with a small break in main steam line, the reactor core remained adequately cooled.

2 INTRODUCTION

High Performance Light Water Reactor (HPLWR) is a supercritical water-cooled reactor (SCWR) concept that has been designed in two projects funded by the European Union: "HPLWR" in 2000-2002 [Squarer (2003)] and "HPLWR2" which is still ongoing in 2006-2010 [Starflinger (2007)]. The reactor concept is based on a thermal SCWR designed in Japan in the late 1990's.

The critical point of water is 22.064 MPa and 373.946 °C. Above these pressure and temperature values, water experiences no liquid-vapour phase-transitions, in other words, boiling or condensation. On the other hand, rapid changes in thermophysical properties, such as density, heat capacity or thermal conductivity, still occur along the extension of the saturation curve. From cooling point of view, the most interesting thermal-hydraulic property of supercritical water is its very high heat capacity along the pseudo-critical line.

Due to the use of supercritical water as coolant, HPLWR achieves very high efficiency of 44 % and simpler plant since steam generators and steam dryers are omitted and coolant is driven directly to the turbines. At supercritical pressures, boiling crisis in the core is not possible, which enhances the safety of the reactor.

Special features of the current work-in-progress design of the HPLWR are the "three pass core", which was introduced to the design for preventing hot spots in the core [Schulenberg (2006)]. The coolant flows through the core three times in the radially separated evaporator, super-heater 1 and super-heater 2 regions. Half of the feedwater coming into the reactor pressure vessel is directed upwards to the upper plenum serve as moderator and the other half flows to the downcomer. Also for the moderator, a three-stage flow scheme is applied [Koehly (2009)]. The water used as moderator flows first though the core in the moderator channels in the middle of the fuel assemblies, them upwards though the core in the gaps between the fuel assemblies and finally again downwards in holes in the reflector, after which it is mixed in the lower plenum with the coolant from the downcomer. There are two reasons for the three-stage moderator flow path. Downward moderator flow is needed in order to overcome the inadequate moderation at the upper part of the core caused by small density of hot supercritical water. On the other hand, reversed flow in the moderator channels was observed even in normal operation when a once-through moderator flow part was applied to the core. The HPLWR reactor pressure vessel with internals and coolant and moderator flow paths is shown in Figure 1.

The aims of the "HPLWR2" project are to assess the feasibility and economical competitiveness of the concept. VTT Technical Research Centre of Finland participates in the Safety Work Package of the project by calculation transient analysis with two system codes: APROS [Hänninen (2008)] and SMABRE [Miettinen (1999), Miettinen (2000)]. The codes have been modified for modelling of supercritical pressures by introducing an artificial two-phase region to the supercritical region [Kurki (2008), Seppälä (2008)]. This way the two-phase solution method from the subcritical region can also be used at supercritical pressures.

The functionality of the codes at supercritical pressures and in the transition to subcritical pressures has been tested in a simple Edwards-O'Brien blowdown scenario explained in Kurki (2008) and Seppälä (2008).

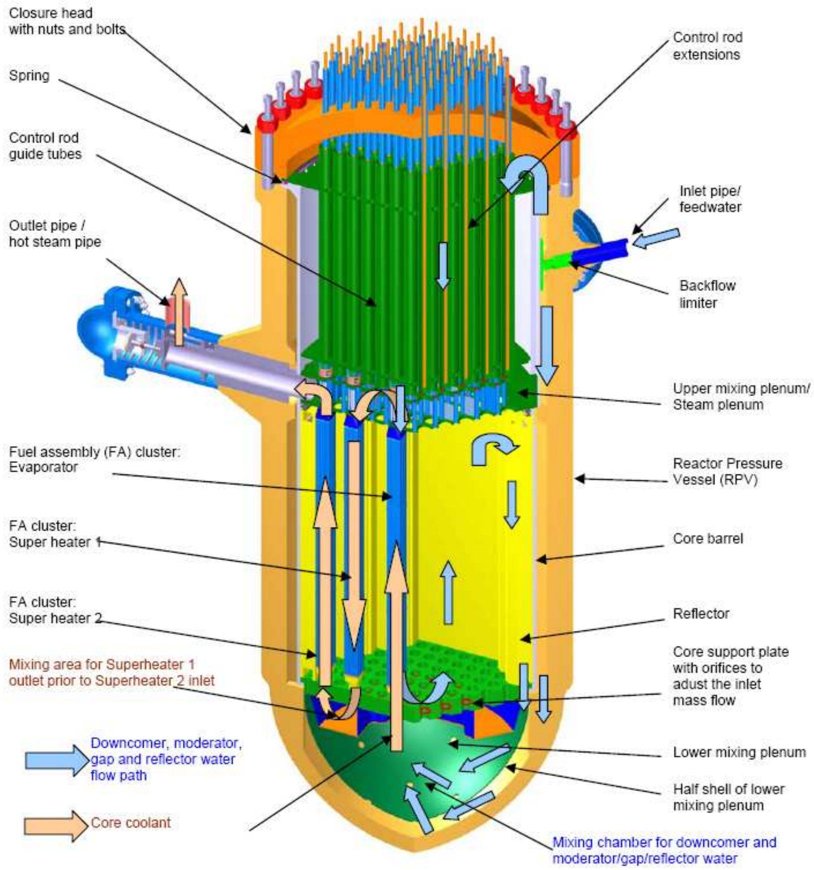

Figure 1. HPLWR reactor pressure vessel with internals [Koehly (2009)].

3 ANALYSIS CODES AND SIMULATION MODELS

Two thermal hydraulic system codes developed originally at VTT, APROS and SMABRE, were upgraded to cope with the supercritical-pressure conditions.

APROS (Advanced PROcess Simulator) is a general-purpose simulation software intended for analysis of industrial processes, developed at VTT in cooperation with Fortum co. since the late 1980's. APROS incorporates various different thermal-hydraulic, neutronic and automation system solvers, and is well suited for whole-plant simulation. SMABRE (SMAll BREak analysis program) is an in-house nuclear reactor safety-analysis code also developed at VTT since the beginning 1980's. As the name suggests, SMABRE is mainly intended for analysis of small break loss-of-coolant accidents, and its thermal hydraulic solver is based on the drift-flux model.

3.1 Thermal hydraulics at the supercritical pressure region

The process of upgrading these two codes to support simulation at supercritical pressures consisted of extending and refining the steam tables used by these system codes to describe the thermophysical properties of water at near- and supercritical pressures, implementing heat transfer and wall friction correlations suitable for the supercritical pressure region, and introducing a pseudo-phase-transition at supercritical pressures.

Numerous heat transfer correlations for the supercritical pressure region have been proposed [Pioro (2004), Pioro (2005)]. Of these correlations two were found most suitable for the simulation of the HPLWR, and thus implemented in APROS. These are the correlation of Bishop et al. [Bishop (1965)]:

$$\mathrm{Nu}_{\mathrm{b}} = 0.0069\,\mathrm{Re}_{\mathrm{b}}^{0.9}\,\overline{\mathrm{Pr}}_{\mathrm{b}}^{0.66}\left(\frac{\rho_{\mathrm{w}}}{\rho_{\mathrm{b}}}\right) \tag{1}$$

and the correlation of Jackson and Hall [Jackson (1979)]:

$$\mathrm{Nu}_{\mathrm{b}} = 0.0183\,\mathrm{Re}_{\mathrm{b}}^{0.82}\,\mathrm{Pr}_{\mathrm{b}}^{0.5}\left(\frac{\rho_{\mathrm{w}}}{\rho_{\mathrm{b}}}\right)^{0.3}\left(\frac{\overline{c_p}}{c_{p,\mathrm{b}}}\right)^{n} \tag{2}$$

In addition, the conventional Dittus-Boelter correlation [Winterton (1998)]:

$$\mathrm{Nu}_{\mathrm{b}} = 0.023\,\mathrm{Re}_{\mathrm{b}}^{0.8}\,\mathrm{Pr}_{\mathrm{b}}^{0.4} \tag{3}$$

can also be used at supercritical pressures, even though it is known to be very inaccurate. The Dittus-Boelter correlation is currently used in SMABRE also for supercritical pressures.

The pseudo-phase-transition occurs in the codes, when the thermodynamic state of a calculation node passes an extension of the saturation line called the pseudo-critical line. The pseudo-critical line consists of the points (p, T_{pc}), where the pseudo-critical temperature T_{pc} is defined as

$$T_{\mathrm{pc}}(p) = \arg\max_{T} c_p(p,T), \quad p > p_c \tag{4}$$

Its purpose is to transfer the mass from the numeric liquid phase to the numeric gas phase when the enthalpy rises (or vice versa when the enthalpy decreases), to ensure that the mass is assigned to the correct phase when the pressure drops to subcritical levels.

In SMABRE, an artificial pseudo-two-phase region is defined along the pseudo-critical line. In this 200 kJ/kg broad region the fluid is numerically handled as a mixture of supercritical liquid and gas and void fraction changes from zero to one.

3.2 Simulation models

Simulation models of the current HPLWR design were created for APROS and SMABRE. The models represent the internals of the reactor pressure vessel with the three-pass core flow configuration. Attention was paid to make the models as mutually similar as possible.

Boundary conditions are used for feedwater mass flow rate and enthalpy and for pressure at the end of the main steam line. Main steam lines are modelled from reactor pressure vessel (RPV) outlet to the turbine valve. A main steam isolation valve (MSIV) is located 8 m from the RPV outlet. For calculation of the main steam line break transients the broken line is modelled separately and the three intact lines are lumped as one.

No neutronics feedback is included in either simulation model.

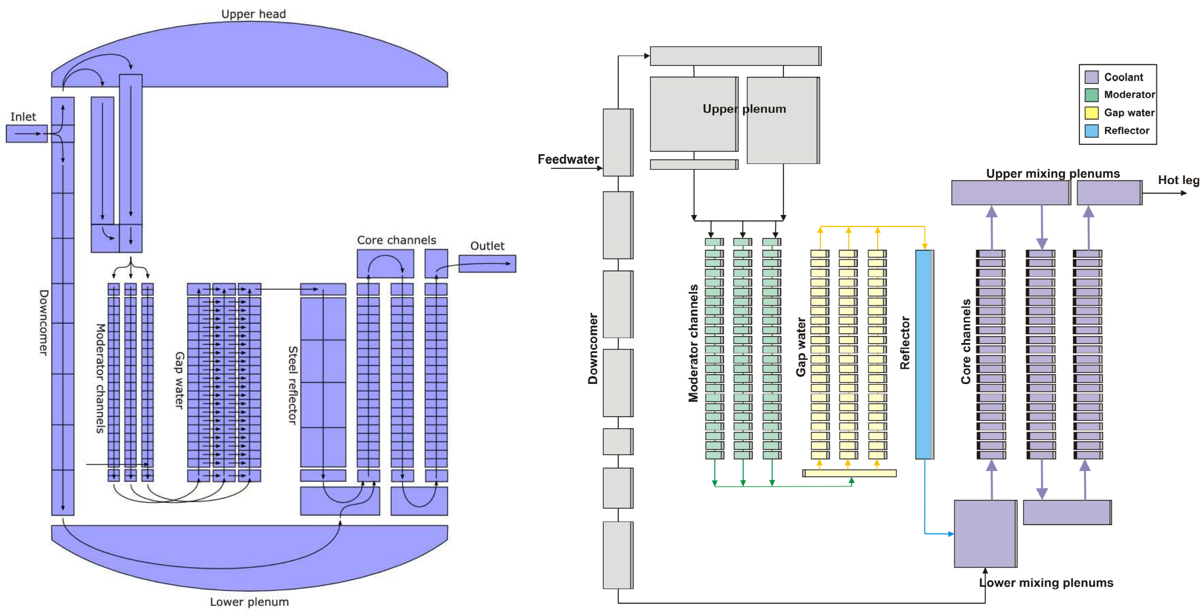

Figure 2. Thermal hydraulic flow channel nodalisations of the HPLWR reactor pressure vessel models for the system codes APROS (left) and SMABRE (right).

4 TRANSIENT ANALYSES

The transient to be analysed was a main-steam line break (MSLB), i.e. rupture of one of the steam pipes between the reactor pressure vessel outlet and the turbine plant. A guillotine break of the steam line was analysed with APROS and a small 8.8 % break in the steam line with SMABRE. In these postulated accidents, the following events take place:

- A break orifice opens in one of the four steam lines, just before the main steam line isolation valve (MSIV).
- The break is detected when difference between feedwater mass flow rate and steam flow rate to turbine exceeds 200 kg/s and a signal for reactor scram is send. The closure of the main steam line isolation valves is initiated by a low pressure signal at 22.5 MPa. The reactor scram time and the MSIV closure time are both assumed to be 3.5 seconds. After scram the power level decreases to decay heat.
- The inlet water is kept constant at 1179 kg/s until the feed-water tank runs empty. In normal operation, the feed-water tank with volume of 300 m^3 is filled with water at 155 °C at 0.55 MPa, and thus empties in 230 seconds after the closure of MSI valves.
- In normal operation, the feed-water is heated from the 155 °C to 280 °C in pre-heaters between the feed-water tank and the reactor inlet. This pre-heating stops when the turbine-line is closed. The time estimated for the feed-water temperature to decrease to 155 °C is 60 seconds after the MSIV closure.
- Low-pressure coolant injection (LPCI), an active safety system, starts to inject cold water (at 40 °C) to the reactor inlet at flow rate of 140 kg/s after the pressure at the inlet drops below 6.0 MPa.

The feed-water keeps the fuel rods cool as long as there is a constant flow rate at the feed-water line. After this, the fuel rod claddings start to heat up because of the decay heat from the fuel. The LPCI system should be designed such that the pressure vessel will be re-flooded in time before the cladding temperatures rises above the acceptance level.

5 RESULTS AND DISCUSSION

5.1 Large break in main steam line

Results of the analysis simulation of a guillotine break in main steam line, as calculated with APROS using the supercritical heat transfer correlation of Jackson & Hall, for the first 10 minutes after the break are presented in Figure 3.

As expected, the break opening causes a very rapid depressurization to the subcritical region. However, pressure at the reactor pressure vessel inlet stays slightly above 6 MPa for more than a minute, and because of this, the LPCI injection isn't actuated but only 90 seconds after the start of the transient. After the depressurization, the hot liquid contents of the pressure vessel tend to evaporate, as can be seen from the void fraction graph, but the evaporation rate is limited by the sonic velocity at the break orifice.

The feed-water flow, which also turns into steam as pressure decreases, keeps the fuel rods sufficiently cooled until the feed-water tank runs dry at 230 seconds. After this, the vessel is almost completely filled with hot vapour, and the relatively low pumping power of the LPCI system is unable to re-flood the reactor core with cold water fast enough: the cold injection water is partially vaporised by the heat from the hot vapour, and the rest just accumulate at the bottom of the vessel and in the downcomer. The cladding temperatures start to rise, and will eventually reach the melting point.

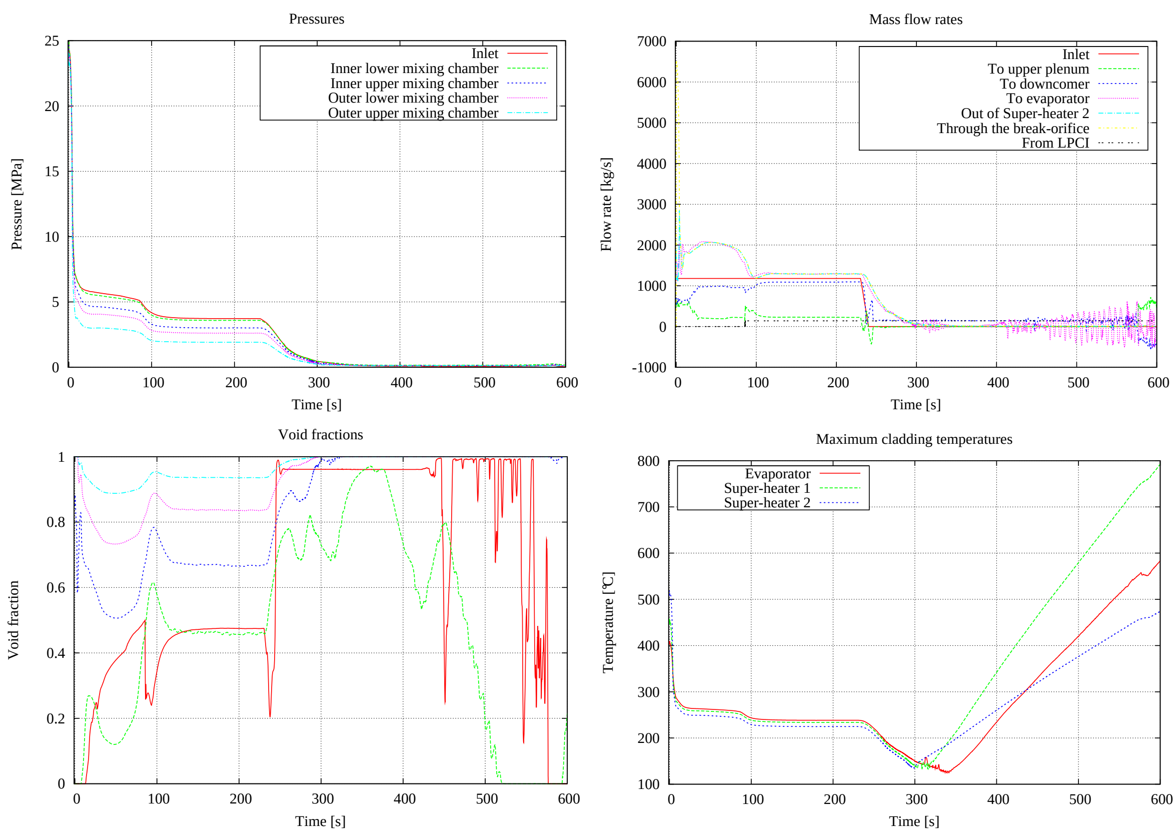

Figure 3. Simulation results from APROS: pressures (top-left), mass flow rates (top-right), void fractions (bottom-left) and hottest cladding temperatures (bottom-right). The legend is the same for the pressure and for the void fraction graph.

5.2 Effect of the heat transfer correlation

Because the heat transfer correlations developed for the supercritical pressure region are known to be inaccurate and to give highly deviating results in some situations, the uncertainty of the simulation results poses a serious concern. To get some insight into gravity of this uncertainty, the transient simulation was calculated with APROS using the three different heat transfer correlations, and the results were analysed.

First, effect of the heat transfer correlation on the cladding temperature distribution in the steady-state was assessed. In figure 4, the cladding temperatures in the three core regions, as calculated by the three different heat transfer correlations, are presented. The curves starting from the left-bottom corner of the graph are the temperature distributions in the evaporator region. and the curves ending to the right-top corner of the graph are the temperature distributions in the super-heater 2 region, while the middle curves represent the distribution in the super-heater 1 region.

The most-notable difference is between the distribution calculated with the Dittus-Boelter correlation and the two other distributions: the Dittus-Boelter correlations seems to underestimate the heat transfer coefficient, which manifests itself as higher cladding temperatures throughout the core region. The distributions calculated with the Jackson-Hall and Bishop correlations, on the other hand, are very consistent, except for the slight deviation that occurs in the middle of the evaporator region, where the pseudo-critical temperature is passed.

The transient simulation, on the other hand, is virtually unaffected by the choice of the supercritical-pressure heat transfer correlation, due to the fact that the pressure drops to subcritical region very early in the simulation, and thus the supercritical-pressure heat transfer correlations are used only a very brief time.

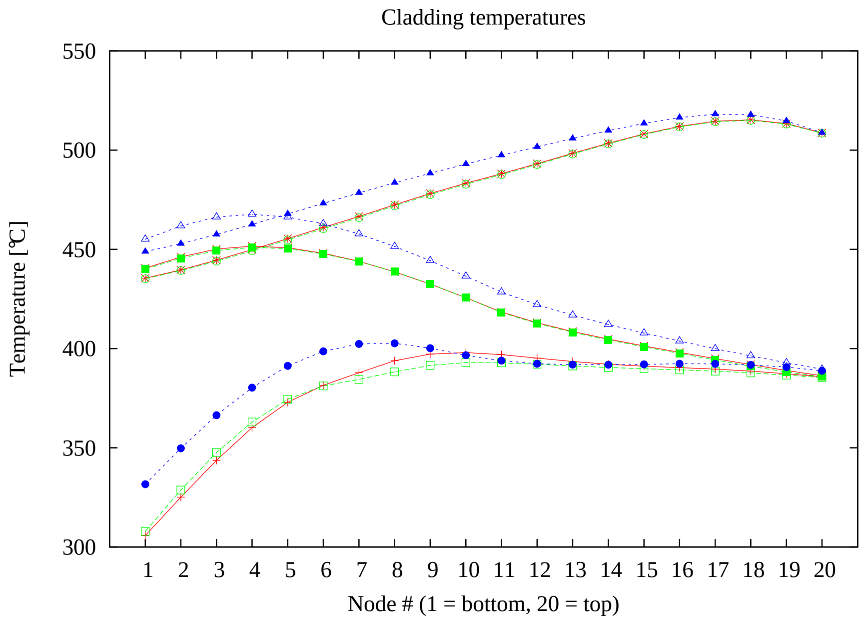

Figure 4. Cladding temperatures in the core region calculated with the heat transfer correlations of Jackson & Hall (red), Bishop (green) and Dittus & Boelter (blue).

5.3 Small break in main steam line

Results from the analysis simulation of a small break in main steam line, as calculated with SMABRE for the first 25 minutes of the transient, are shown in Figure 5.

At the beginning of the transient, the mass flow rate from the 8.8 % break stabilizes at 250 kg/s preventing depressurization until scram is initiated. The following slow pressure decrease to 9.6 MPa has little impact on mass flow rate through the core and therefore temperatures and void fractions in the core decrease. At 234 seconds the feedwater tank is emptied, which causes the pressure to drop rapidly to 1.4 MPa and the mass flow rates to decrease almost to zero. However, fluctuation of void fraction in the evaporator part of the core induces strong oscillation of the mass flow in evaporator, upper plenum and downcomer, which provides sufficient cooling in the core and prevents the cladding temperatures from increasing even to the level of normal operation.

The LCPI system is assumed deactivated during the transient.

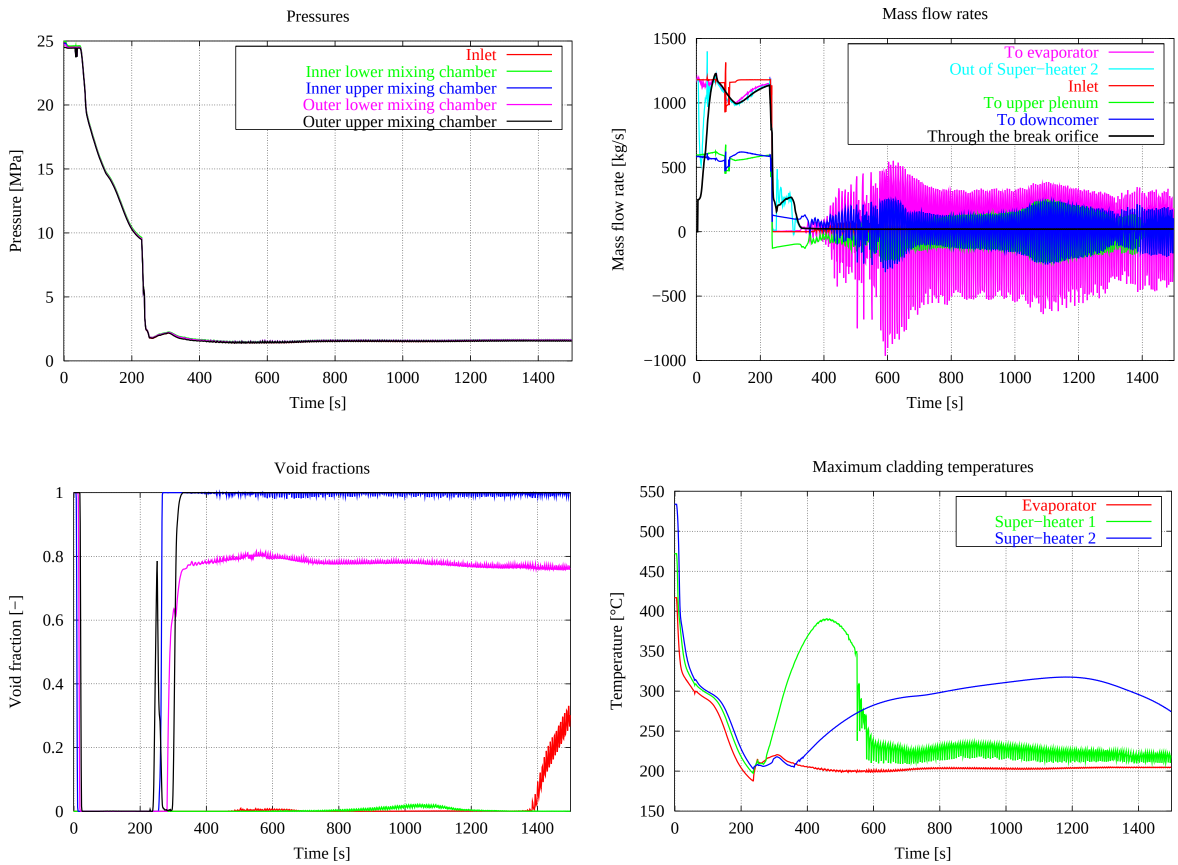

Figure 5. Simulation results from SMABRE: pressures (top-left), mass flow rates (top-right), void fractions (bottom-left) and hottest cladding temperatures (bottom-right). The legend is the same for the pressure and for the void fraction graph.

6 CONCLUSION

VTT's thermal hydraulic system codes APROS and SMABRE have been updated for modelling of water at supercritical pressures by introducing a pseudo-phase-transition in the supercritical pressure region and extending the steam tables to cover higher pressures.

Detailed simulation models of reactor pressure vessel internals of the High Performance Light Water Reactor were created for APROS and SMABRE. Feedwater mass flow rate and enthalpy and outlet pressure were applied as boundary conditions.

Preliminary transient analyses of main steam line breaks in the HPLWR were performed using APROS and SMABRE. Analysis of a guillotine break in the main steam line showed a need for more efficient low pressure coolant injection after the feedwater tank has emptied. In case of a small break in the main steam line, the reactor was sufficiently cooled even with the LPCI system deactivated.

While the known inaccuracy of the supercritical-pressure heat transfer correlations is a major concern in any safety analysis calculation, and poses a serious question on the validity of the simulation results in the general case, it doesn’t pose a problem in transient calculations where the pressure drops to subcritical levels very early on the simulation, as might have been expected

Symbols

c_p	isobaric heat capacity	J/(kg °C)
p	pressure	Pa
T	temperature	°C
D	diameter	m
u	flow velocity	m/s
ρ	density specific	kg/m^3
λ	thermal conductivity	W/(m °C)
μ	dynamic viscosity	Pa s
v	kinematic viscosity (μ/ρ)	m^2/s
Nu	Nusselt number (hD/λ)	-
Pr	Prandlt number ($c_p\mu/\lambda$)	-
Re	Reynolds number (uD/v)	-

REFERENCES

Bishop, A., Sandberg, R. and Tong, L. 1965. Forced convection heat transfer to water at near critical temperatures and supercritical pressures. Westinghouse electric corporation, atomic power division. WCAP-5449.

Hänninen, M. and Ylijoki, J. 2008. The one-dimensional separate two-phase flow model of APROS. VTT Tiedotteita – Research Notes 2443.

Jackson, J. and Hall, W. 1979. Forced convection heat transfer to fluids at supercritical pressure. Turbulent forced convection in channels and bundles, Vol 2, edited by Kakaç, S. and Spalding, D.

Koehly, C. 2009. HPLWR reactor design concept. Proceedings of 4th Internaltional Symposium on Supercritical Water-Cooled Reactors, Paper No. 37, Heidelberg, Germany, March 8-11.

Kurki, J. 2008. Simulation of thermal hydraulics at supercritical pressures with APROS. Master's thesis, Helsinki University of Technology.

Miettinen, J. 1999. A Thermohydraulic Model SMABRE for Light Water Reactor Simulations. Licensiate Thesis, Helsinki University of Technology.

Miettinen, J. and Hämäläinen, A. 2000. Development and validation of the fast running thermohydraulic model SMABRE for simulator purposes. Proceedings of ICONE8, Baltimore, MD USA, April 2-6, 2000. 12 p, CD-ROM Proceedings, ASME, New York, NY (2000).

Pioro, I. and Duffey, R. 2004. Heat transfer to supercritical fluids flowing in channels – empirical correlations (survey). Nuclear engineering and design, Vol. 230. P. 69-91.

Pioro, I. and Duffey, R. 2005. Experimental heat transfer in supercritical water flowing inside channels (survey). Nuclear engineering and design, Vol. 235. P. 2407-2430.

Seppälä, M. 2008. Neutronic and thermal-hydraulic modelling of High Performance Light Water Reactor. Master's thesis, Helsinki University of Technology.

Schulenberg, T. and Starflinger, J. 2006. Three Pass Core Design Proposal for a High Perfor-mance Light Water Reactor. Proceedings of INES-2, Yokohama, Japan, November 26-30.

Squarer, D., Schulenberg, T., Struwe, D., Oka, Y., Bittermann, D., Aksan, N., Marachy, C., Kyrki-Rajamaki, R., Souyri A. and Dumaz, P. 2003. High performance light water reactor. Nuclear Engineering and Design, Vol. 221. P. 167-180.

Starflinger, J., Schulenberg, T., Marsault, P., Bittermann, D., Marachy, C., Laurien, E., Lycklama, J. A., Anglart, H., Aksan, N., Ruzickova M. and Heikinheimo, L. 2007. European research activities within the project: "High Performance Light Water Reactor Phase 2" (HPLWR Phase2). Proceedings of ICAPP 2007, Paper No. 7146, Nice, France, May 13-18.

Winterton, R. 1998. Where did the Dittus and Boelter equation come from? International journal of heat and mass transfer, Vol. 41. P. 809-810.

