Thermal hydraulic Influence on the Severe Accident Progression in a Small Modular Reactor

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ABSTRACT

System dynamics in the small modular reactor (SMR) is determined by the large amount of water in the reactor vessel and the interaction with the small volume of the containment. The amount of decay heat is also smaller due to the low nominal power. If a significant pressure reduction is ensured in the initial phase of the accident, the core will be submerged for a long period of time. Fluid loss through the pressurizer valves will not significantly reduce the reactor coolant system inventory, but coolant leakage through a potential rupture site will be minimal due to the low pressure in the reactor vessel.

The analyses conducted in this study focus on the influence of thermal hydraulic boundary conditions on the heat removal capability in critical situations. The influence of safety systems is limited, and the initial conditions are conservative. The probability of a severe accident in a small modular reactor is low, but the accident should still be taken into consideration in order to provide a basis for the preparation of technical documentation, e.g. severe accident management guidelines, or emergency planning zones. Thermal hydraulic calculations are performed with coupled RELAP5-GOTHIC code and the severe accident calculations with the ASYST code, which combines a detailed thermal hydraulic model in design basis conditions and a mechanistic approach in assessing the progression of a severe accident. The core quench tracking, hydrogen production, core melt, molten pool progression and other main severe accident phenomena are studied. The reactor system chosen is IRIS, a classic example of a light water SMR with an integral reactor vessel and a small containment with built-in passive safety systems. The numerical model is complex and enables a realistic simulation of the power plant behaviour.

Keywords: *small modular reactor, safety analysis, IRIS, severe accident, RELAP5, GOTHIC, ASYST, passive safety systems*

1 INTRODUCTION

Recent safety analyses of the IRIS reactor [1] show a strong impact of the operation of passive safety systems on the possibility of successful core cooling [2, 3]. Reactor pressure vessel (RPV) depressurization is the main safety measure to ensure the long-term integrity of the reactor core during a potential loss-of-coolant accident (LOCA). The reduction in pressure causes a reduced loss of coolant through the break, and in combination with the removal of decay heat through the emergency heat removal system (EHRS) and the injection of water from the containment tanks, ensures a stable water level in the RPV. The injection from the long-term gravity make-up system (LGMS) is only possible when the containment pressure is larger than the RPV pressure. The conditions for its activation are therefore ensured only after the primary system has been depressurized, which means that depressurization must occur as early and as quickly as possible.

Calculations were performed with thermal hydraulic system code RELAP5, containment code GOTHIC and severe accident analysis code ASYST [4]. The explicitly coupled RELAP5/GOTHIC code version [5] was used for the design basis transient sequence. If the accident propagates beyond the design basis, boundary conditions obtained by calculations with RELAP5/GOTHIC are used as initial conditions for the ASYST calculation.

Small modular reactors were developed with the aim of increasing safety compared to conventional nuclear power plants. A large break LOCA is eliminated by design, and the consequences of a small break LOCA are mitigated by the action of passive safety systems. Multiple analyses have confirmed that the probability of a severe accident is very small, even in the event of a single failure of the safety systems [2, 6]. The integrity of the core is threatened only in case of multiple failure of the safety class systems [7, 8], or possibly if there is a direct rupture of the RPV at low elevation [3].

What complicates the system dynamics a bit is that the core cooling is based on natural circulation. The occurrence of thermal hydraulic instabilities accompanied by high amplitude fluctuations in the RPV pressure and the core water level [9] can affect heat transfer and threaten efficient heat removal. Additional studies are thus performed to check the overall thermal hydraulic system performance depending on the RPV depressurization rate. The size of the system, occurrence of natural circulation, and interaction between the RPV and the containment also pose challenges for numerical modelling. Therefore, IRIS nodalization is complex, and the selected codes (RELAP5, GOTHIC, ASYST) ensure the physical accuracy of the performed analyses.

2 NUMERICAL MODEL

The nodalization of the reactor coolant system (RCS) is shown in Figure 1 [10]. IRIS reactor features an integral vessel with built-in eight steam generators (control volumes 201-281), eight reactor coolant pumps (control volumes 191-198) and the pressurizer at the top of the RPV (control volume 130). The reactor core has 89 fuel assemblies, each having 264 fuel rods (geometry type 17×17). There is a large riser section (control volumes 120-124) above the core to facilitate natural circulation during accidental conditions. The core is divided into four thermohydraulic channels (control volumes 107-110) to meet ASYST requirements for calculating the radial spread of core melt. The annular downcomer (control volume 101) has a large volume, which prevents the core from drying out quickly during LOCA.

Passive safety systems include emergency boration system (control volumes 604-612), emergency heat removal system (control volumes 501-511), long-term gravity make-up system (control volumes 643-647) and automatic depressurization system (control volumes 682-693). The reactor cavity is connected by the direct vessel injection line (DVI) to the RPV (control volumes 651-657) but calculations have shown that there is no water injection through that line.

The containment nodalization is presented in Figure 2. The mathematical model of the containment is composed of nine control volumes. Control volumes 1 and 2 represent the drywell, control volume 3 represents the reactor cavity, control volume 4-7 represent two long-term gravity make-up system (LGMS) water tanks and vent pipes that connect these tanks to their respective drywells. Control volumes 8 and 9 are air tanks connected to the LGMS tanks. The connections to the RELAP5 volumes, by means of coupled numerical routines, are also indicated. The automatic depressurization system (ADS) connection is realized by the flow path 12, the flow path 1 is the break connection, flow paths 8 and 9 are LGMS connections to the DVI lines, and flow paths 10 and 11 are connections between the cavity and the RPV.

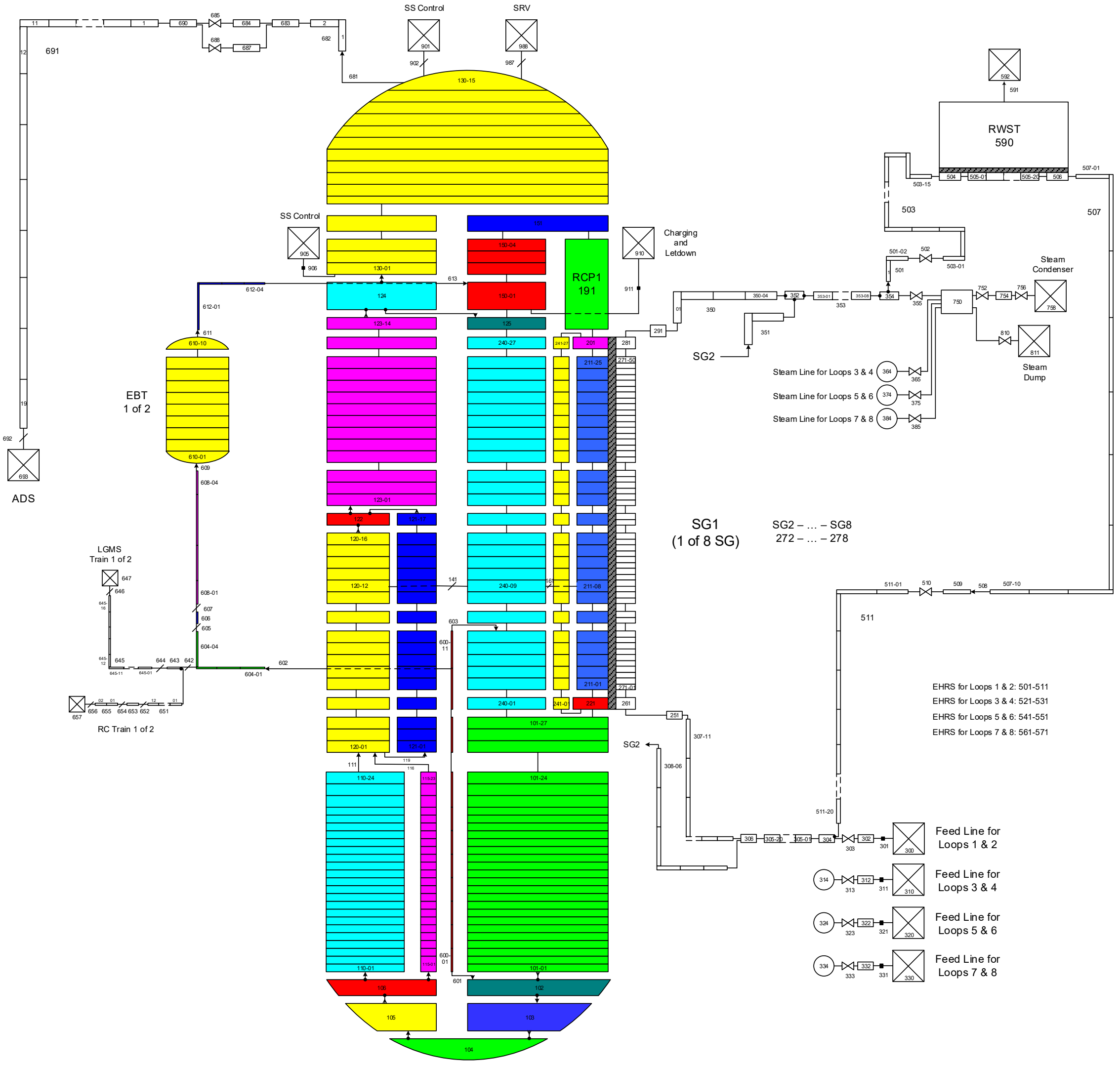

Figure 1: Reactor coolant system nodalization

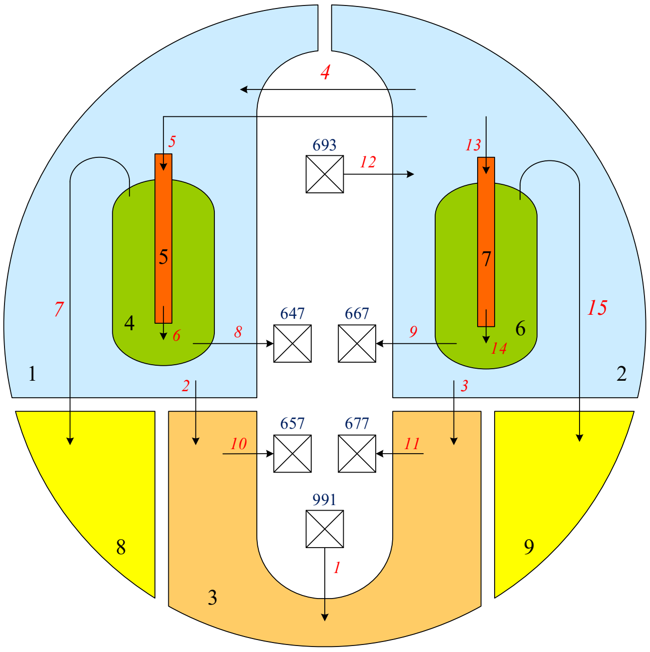

Figure 2: Containment nodalization

3 CALCULATIONS AND DISCUSSION

Initial safety analyses covered the most critical failures at the IRIS facility: pipeline breaks in the DVI line and the chemical and volume control system (CVCS) line [11]. Calculations were performed to confirm the operability of the safety systems to prevent the core temperature from rising in the short term and ensure conditions for natural circulation. All safety systems were active. Due to the complex model and limitations of the computers of the time, calculation times were short (about 1000 s), but in a situation where all plant systems were operating without failure, there was no need for long-term calculations. Today, it is also possible to conduct long-term calculations that test the system's behaviour after a single or multiple failure of safety systems, or for less likely LOCA sequences. Such analyses enable not only the quantification of the possible loss of core integrity, but also a better insight into the process of natural circulation and system stability.

3.1 CVCS Line Break

The largest pipes in the IRIS plant have a cross-section of 4 inches (10.2 cm) and their rupture is considered the most severe possible LOCA scenario. Such is a CVCS pipe connected to the upper plenum of the reactor vessel. Its rupture is a design basis accident for which the core coolability is proven by the IRIS design [11]. Previous studies showed a different outcome if one of the passive safety systems is unavailable [2]. It was determined that the most important safety system is the EHRS. The lack of residual heat removal via the EHRS, combined with the CVCS line failure, will cause the core to start to dry out after 19000 s, and the accident will propagate into a severe accident.

3.1.1 Unavailability of the Decay Heat Removal

The EHRS removes decay heat by forcing natural circulation between the core and the heat exchangers in the RWST and lowers the RCS pressure by condensing water vapour inside the reactor vessel. Thus, by lowering the pressure, it limits the coolant loss through the break. If there is no EHRS available, the core is quenched as long as there is enough water in the RPV. The water inventory is sufficient for 19000 s of successful heat removal. After that, the temperature increases (Figure 3) as the collapsed core water level decreases (Figure 4).

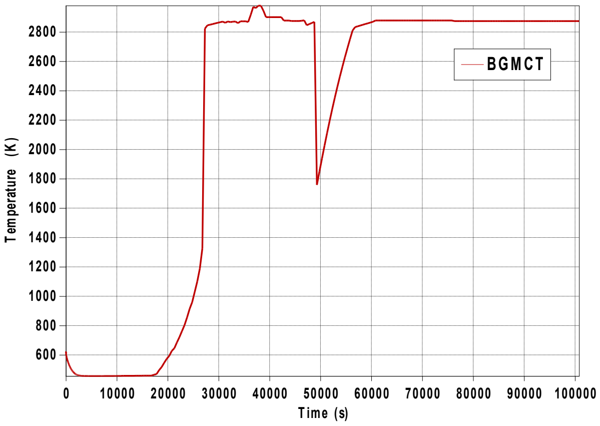

Figure 3: Peak cladding temperature (CVCS break, no EHRS)

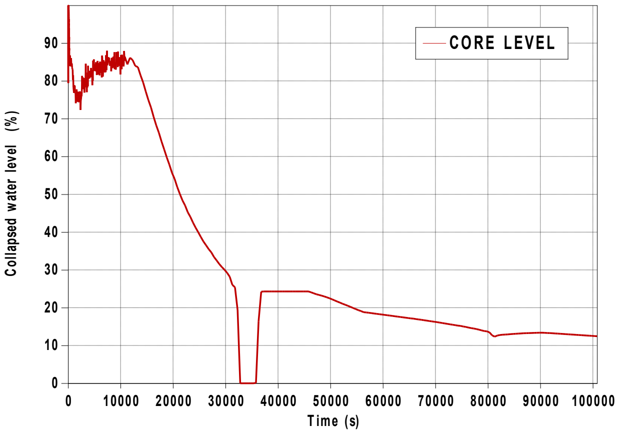

Figure 4: Collapsed core water level (CVCS break, no EHRS)

Rising temperatures and water evaporation will cause fuel rod claddings to oxidize, leading to core melting. The mass of hydrogen produced and the equivalent radius of the molten pool, within the core, are shown in Figures 5 and 6, respectively. The increase of oxidation rate at 28000 s coincides with the increase of temperature. The kinetics of oxidation is such that hydrogen production and heat released by oxidation depend exponentially on temperature, so this relationship is expected.

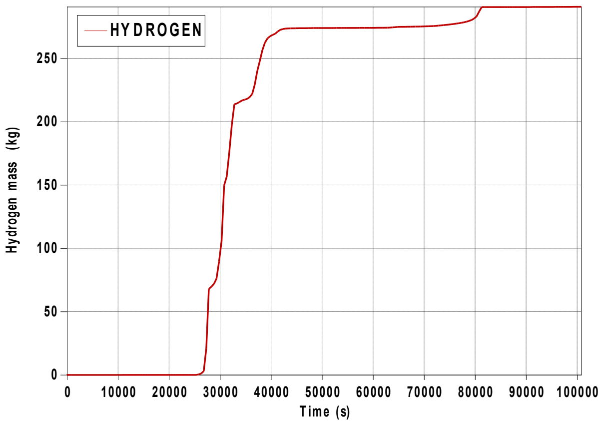

Figure 5: Hydrogen production (CVCS break, no EHRS)

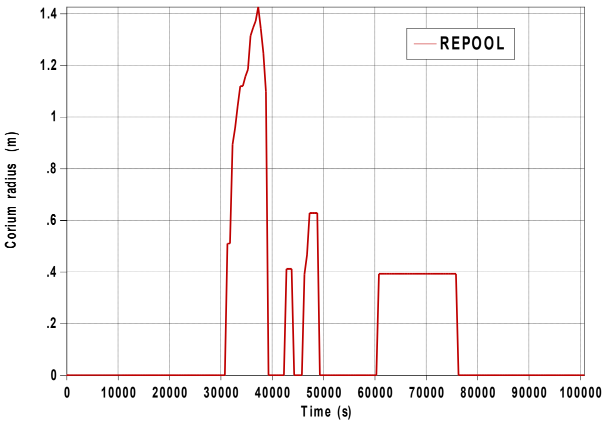

Figure 6: Equivalent radius of the molten pool (CVCS break, no EHRS)

A correlation between the variables shown in Figures 3-6 is also visible. A drop in maximum cladding temperature at 50000 s is due to a large corium relocation into the lower head. Several peaks in hydrogen production rate between 30000 s and 40000 s are caused by a core-wide melting of fuel assemblies and exposing metallic zirconium to high-temperature steam. In the same period there is a complete absence of liquid phase in RELAP5 core control volumes, which again is a typical indication of extensive core melt. The final hydrogen mass of 290 kg corresponds to 65% oxidation of Zircaloy tubes.

The mass of accumulated molten core materials (UO_2, ZrO_2, Ag) in the RPV lower head is shown in Figure 7. The corium relocated to the lower head, through the bypass channel, several times between 30000 s and 80000 s. The mass of relocated material corresponds to 69% of the core inventory. Thus, the fraction of cladding oxidation (65%) and corium displacement (69%) are similar. Damage to the fuel rods supports the oxidation process because it uncovers unoxidized zirconium.

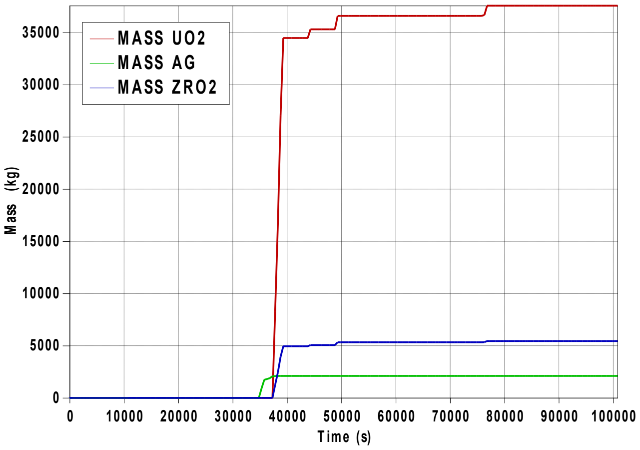

Figure 7: Accumulated material in the RPV lower head (CVCS break, no EHRS)

3.1.2 Impact of RCS Depressurization with EHRS unavailable

Without decay heat removal via the EHRS, the CVCS line break accident will proceed to a severe accident. IRIS is equipped with ADS to ensure fast depressurization in order to limit the coolant losses from the RCS and to establish water injection from the tanks in the containment, as soon as possible. The system consists of a pipe that connects the pressurizer region of the RPV to the containment. Its cross-section is small (diameter 8.4 cm), in accordance with the IRIS system design logic of eliminating large cross-section pipes to prevent possibility of a large break LOCA.

Even though the analyzed accident is very demanding from an accident management perspective, combination of a small break LOCA and unavailability of EHRS should still result with consequences that are not severe. Therefore, additional calculations were performed with larger diameter ADS pipe. The pipe is connected at the top of the RPV, and the fluid is modelled to flow directly into the containment. This is de facto a medium sized LOCA. The goal of the calculation was to determine whether it was possible to successfully cool the core under conditions where water leakage from the system was reduced and steam leakage through the ADS was increased.

Two scenarios were analyzed, with five (diameter 18.8 cm) and ten times (diameter 26.5 cm) the cross-sectional area of the design ADS pipe. The results are shown in Figures 8-10. For comparison, results of the original scenario calculated with RELAP5 are also shown. Since RELAP5 cannot simulate a severe accident, results slightly differ from the scenario presented in the previous section.

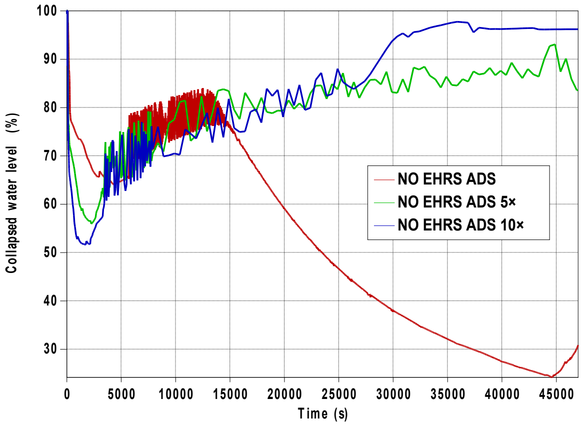

Figure 8: Collapsed core water level (CVCS break, no EHRS, different ADS pipe cross-sections)

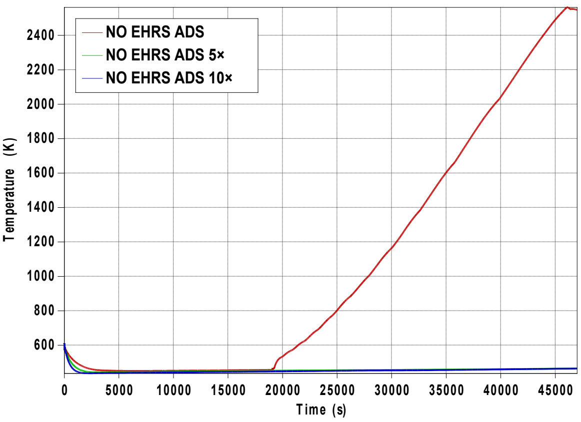

Figure 9: Peak cladding temperature (CVCS break, no EHRS, different ADS pipe cross-sections)

By increasing the diameter of the ADS pipe, indeed the break flow decreased since there was a faster decrease of the RCS pressure. Of course, on the other hand, the steam leakage from the top of the RPV has increased.

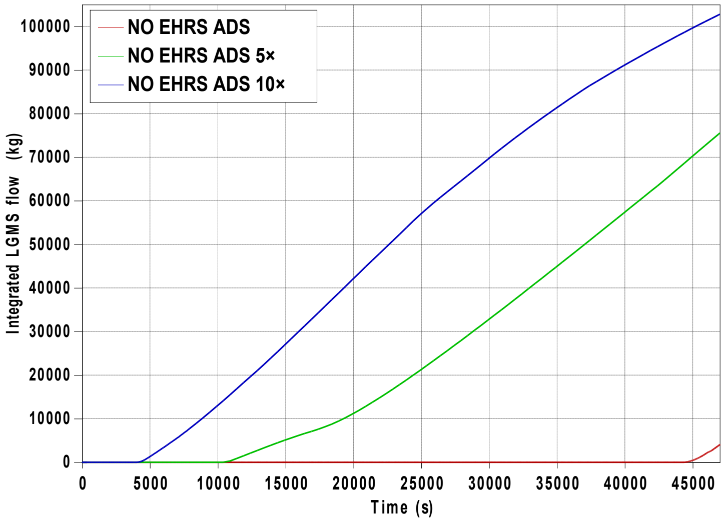

Figure 10: Integrated water flow from the LGMS – one line (CVCS break, no EHRS, different ADS pipe cross-sections)

The net leakage from the RCS in the case of the wider ADS pipe was slightly smaller, but this did not play a significant role, as can be seen in Figure 8. The collapsed core water level was kept high enough to ensure successful core quenching (Figure 9) during the first 15000 s. Since this is a relatively slow transient, the dynamics of the system in the long term is more important. Therefore, the injection of water from the LGMS tanks plays a crucial role (Figure 10). The system for the long-term cooling is activated much earlier when the RCS depressurization is faster. The pressure build-up in the containment will force the water into the reactor vessel. The pressure inside the containment for all three scenarios is approximately the same, so the rupture of a pipe with a larger cross-section does not pose a threat to the containment from excessive pressure. The LGMS activation in the early phase of the transient thus prevents the core to dry out and maintains the long-term cooling. In a case with the narrowest ADS pipe, the RCS depressurization was much slower than in the cases when the ADS pipe was five, or ten times wider. Once the containment pressure became higher than the RCS pressure, the core already dried out, cladding significantly oxidized and the molten pool formed inside the RPV. The goal of this analysis is certainly not to influence the design of the IRIS system, but to quantify the complex thermal hydraulic processes in the SMR and their impact on the reactor safety.

3.2 RPV Break in the Lower Plenum

The results of the previous analysis motivated us to recalculate the scenario of a severe accident that occurs after the rupture of the RPV in the lower plenum [3]. Two additional calculations were again performed, with five and ten times the cross-sectional area of the ADS pipe. The EHRS was assumed to be available.

In the initial case, when the break is positioned at the RELAP5 control volume 102 (lower downcomer, elevation below the active core), the core heatup starts already at 1000 s and the core melt is formed as early as an hour after the start of the transient. Once again, the RCS depressurization helps to stabilize the system, but the system behaviour depends on the pressure reduction rate, i.e. the width of the ADS pipe. Only in the case of the largest cross-section (diameter 26.5 cm) we clearly have successful core cooling. In the case of the medium pipe cross-section (diameter 18.8 cm), the maximum cladding temperature reaches 2200 K (Figure 11), which is admittedly lower than the fuel melting temperature, but is high enough to cause increased cladding oxidation. The collapsed core water level is shown in Figure 12.

The transient is fast. The pressure in the containment becomes higher than the RCS pressure after a little less than 1000 s, which is when limited injection via the LGMS begins, but during this period, injection from the emergency boration system (EBS) tanks is much more significant. However, this is not enough to quench the core in the case of a small or medium-width ADS pipe. The reason is a large water leak through the break in the lower plenum, which the safety injection cannot compensate for. In the third scenario with the largest ADS pipe width, this leakage is the smallest due to fast RCS depressurization and the smallest pressure force acting on the fluid (Figure 13). As the width of the ADS pipe increases, steam leakage from the top of the RPV increases, but this leakage is generally lower than the water leakage in the lower plenum. Therefore, the water level in the RPV is generally higher the larger the pipe cross-section at the top of the vessel. For the largest ADS pipe size, there is enough water in the core to provide efficient cooling.

After an initial sudden drop in the RPV water level, the level rises in the period between 1000 s and 2000 due to the joint action of the EBS and LGMS, but then drops again until approximately 5000 s, for scenarios with five and ten times the width of the ADS pipe. The drop in the water level is the result of evaporation of previously injected water which causes the RCS pressure to rise above the containment pressure. This temporarily stops the LGMS injection, while at the same time, the EBS tanks are almost completely emptied. Restarting the LGMS injection at 5000 s, after another change in the RCS and containment pressure trends, causes the water level to rise. However, in the medium cross-section ADS pipe scenario, this does not occur early enough to prevent the fuel temperature from rising above 2200 K.

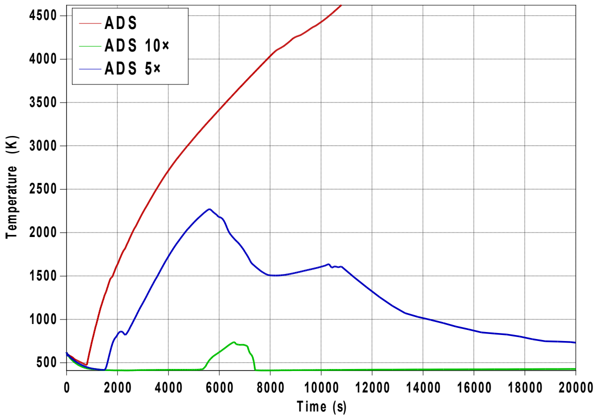

Figure 11: Peak cladding temperature (lower plenum break, different ADS pipe cross-sections)

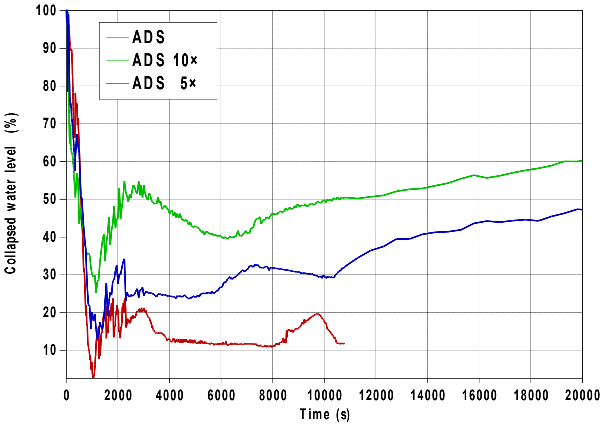

Figure 12: Collapsed core water level (lower plenum break, different ADS pipe cross-sections)

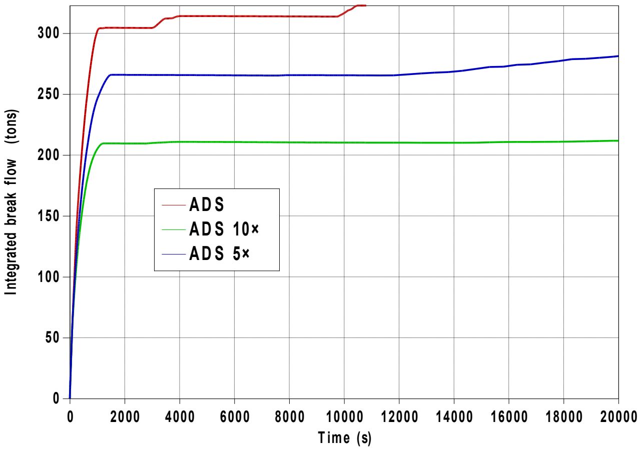

Figure 13: Integrated coolant flow at the break (lower plenum break, different ADS pipe cross-sections)

Finally, for the 10 times size of the ADS pipe, a scenario without the EHRS was analyzed. Results indicate that this is a beyond design basis sequence for which the limited core damage and substantial cladding oxidation cannot be excluded. The maximum temperature is above 2000 K which marks occurrence of a severe accident. The core water level is of course lower when there is no EHRS. The EHRS does not only provide decay heat removal but also lowers the RCS pressure by condensing the steam inside the RPV. Thus the loss of coolant through the break and through the ADS pipe is lower with the EHRS available. We therefore see that in the case of the RPV break in the lower plenum, the EHRS plays an important role in mitigating the consequences of the accident, regardless of how quickly the depressurization of the RCS takes place by releasing steam from the pressurizer.

3.3 Discussion

In the early days of SMR development, the focus was on conducting design basis event analyses to verify SMR safety in general and, specifically, the effectiveness of passive safety systems. Detailed numerical models of the facilities have been developed using stand-alone or coupled computer codes [12, 13]. The calculation times were a few thousandths of a second, which was enough to confirm the reactor design. However, the CPU time was high due to hardware limitations of computers 20-30 years ago.

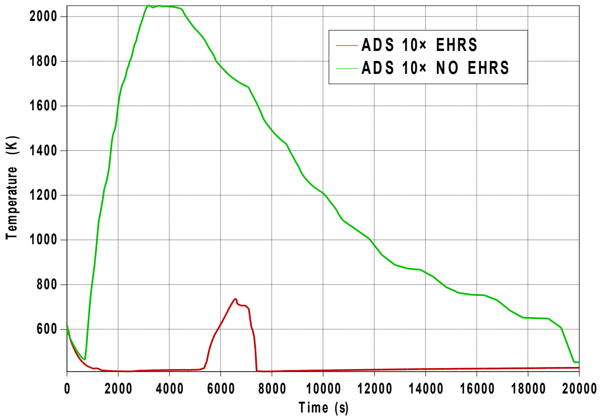

Figure 14: Peak cladding temperature (lower plenum break, influence of EHRS)

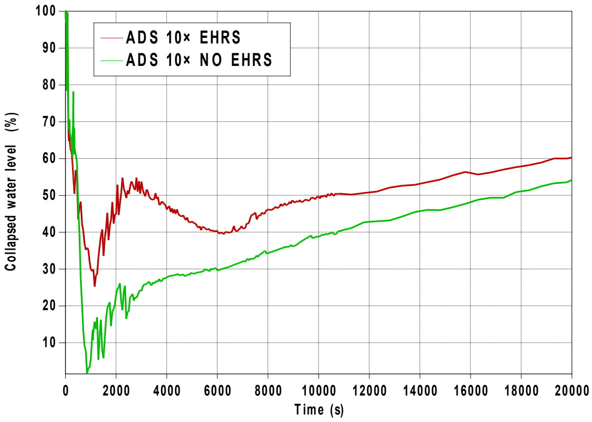

Figure 15: Collapsed core water level (lower plenum break, influence of EHRS)

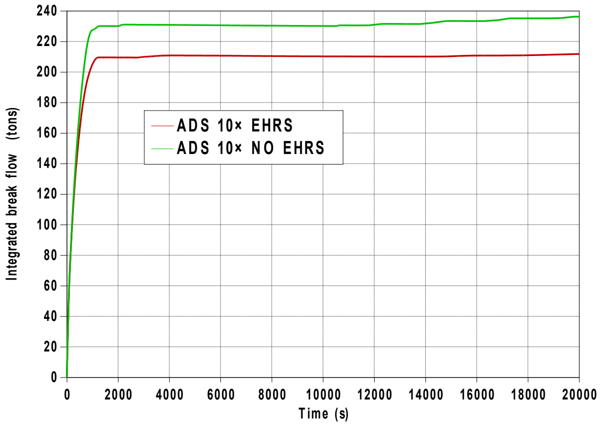

Figure 16: Integrated coolant flow at the break (lower plenum break, influence of EHRS)

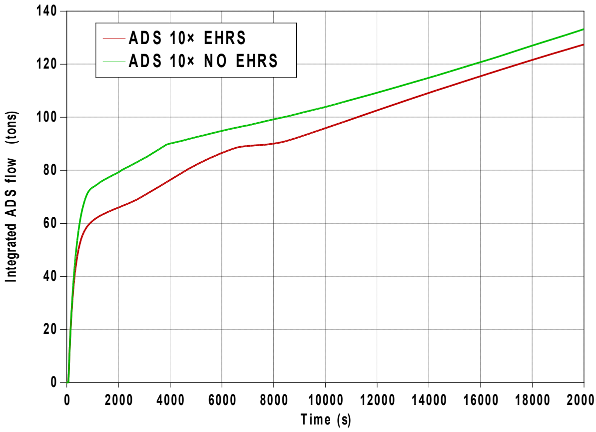

Figure 17: Integrated coolant flow through the ADS pipe (lower plenum break, influence of EHRS)

Today, research is moving towards creating a database for less likely, but more demanding, failures, or combinations of failures with aim to support development of severe accident management guidelines and creation of emergency planning zones [14]. The severe accident analysis would be a starting point for further estimation of radionuclide release and transport, and dose rate quantification. The quantification of complex accident scenario which tests the limits of the safety systems to maintain core coolability requires long computation times and fine adjustment of the minimum time-step. The IRIS integral vessel contains two to three times more water than the conventional nuclear power plant. The process of water evaporation is thus rather slow and numerically challenging to simulate, especially with the one-dimensional system codes. Three-dimensional finite volume or finite element simulations are still impractical to perform. Therefore, the experience gained in modelling SMRs such as IRIS remains crucial for future numerical simulations of new types of SMRs.

4 CONCLUSION

Reactor system depressurization plays a major role in stabilizing thermal hydraulic conditions following a potential hazardous accident, such as LOCA. The more significant the RCS pressure drop, the earlier the long-term core cooling system is activated. When the break is at a high elevation, the transient is slower, and the greater the likelihood that the consequences will be minor. Besides, the only way a severe accident can occur is if the most important safety system, the EHRS, is unavailable. In that case, since the ADS pipe has a small cross-section and provides a limited RCS pressure decrease, depressurization rate is not sufficient for the LGMS to start in time, before the core degradation process begins. However, when the size of the opening at the top of the RPV is increased, the LGMS will start much earlier, due to the sudden pressure drop, and the core will be successfully cooled (feed and bleed strategy) at all times.

As the break position moves lower across the reactor vessel, the transient accelerates, and the dynamics of the processes of coolant loss, RCS pressure reduction, containment pressure increase and decay heat removal becomes increasingly complex. When the break is located in the lower plenum, there is no possibility to ensure core cooling without the EHRS and, even with the EHRS operational, this is only possible when the ADS pipe cross-sectional area is large. The RCS depressurization must ensure a small coolant loss because it takes some time before the LGMS starts delivering full water flow. What is de facto crucial is what happens in the system during the first 1000 s. If the minimum water level is maintained above 30% during that period, the long-term water injection that will then follow will successfully remove heat from the core through the natural circulation process.

ACKNOWLEDGMENTS

This work was supported by the Croatian Science Foundation under the project number HRZZ-IP-2024-05-4011.

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