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Calculation of Unmitigated Small Break Loss of Coolant Accident for the IRIS Reactor

Siniša Šadek, Davor Grgić, Petra Strmečki

Summary **— Preliminary probabilistic risk assessment (PRA) analyses have shown that the IRIS (International Reactor Innovative and Secure) reactor has very low core damage frequency, but in the frame of evaluating accident sequences in IRIS relevant for revising the need for relocation and evacuation measures some severe accident sequences should be defined. Systematic approach based on PRA results was historically used for identification of the sequences and, subsequently, explicit deterministic calculation of a representative sequence was then performed. Calculation methodology is based on using the coupled RELAP5-GOTHIC code to provide the boundary conditions for the severe accident calculation by means of the ASYST code.**

The limiting severe accident scenario analyzed in the paper was hypothetical reactor pressure vessel break at the active core bottom elevation with passive safety systems available. Preliminary studies demonstrated that the accident sequence is highly dependent on the break position along the reactor pressure vessel outside surface and by moving the break downwards the core loses more water and, hence, its temperature rises faster. The break size is 4-inch in diameter which corresponds to the size of the piping in the chemical volume and control system. The reactor core heat-up, cladding oxidation, core degradation and core melt progression processes are similar to those obtained in the analyses severe accident progression in light water reactors.

Keywords **— IRIS reactor, severe accident, SB LOCA, RELAP5, GOTHIC, ASYST**

I. Introduction

IRIS (International Reactor Innovative and Secure) concept has been primarily focused on establishing a design with innovative safety characteristics. The first line of defence in IRIS reactor was moved toward elimination of event initiators that could potentially lead to core damage. This concept is called the “safety by design” approach. Its application in addition to improved safety minimizes the number and complexity of the safety systems and required operator actions.

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Two limiting small break loss of coolant accident (SB LOCA) design bases analyses have already been performed for IRIS reactor [1]. In that accompanying paper the most critical failures were analyzed: the complete rupture of a chemical volume and control system (CVCS) pipe, and the double-ended rupture of the direct vessel injection (DVI) line. The analyses confirmed the validity of the IRIS design in mitigating the consequences of a postulated small break LOCA by maintaining the core covered with water throughout the whole duration of the transient, thus preventing a significant increase in the cladding temperatures.

In the analysis, presented in this paper, a hypothetical 4-inch break of the reactor pressure vessel (RPV) wall in the lower plenum was considered. Present reactor safety analyses are not addressing this event due to its extremely low probability. For the IRIS reactor with overall core damage frequency from internal events of the order of 10^{-8}, this highly unlikely event was used as a calculation exercise needed to develop simple severe accident scenario. All active systems were assumed unavailable. On the other hand, passive safety systems like emergency heat removal system (EHRS) and long-term gravity makeup system (LGMS) were assumed operable. The break was located below the bottom of the active core. That means that is not possible to keep the core covered with water. RPV inventory will decrease very fast and no natural circulation, normally present in IRIS reactor, will be established. Unavailability of the CVCS and the normal residual heat removal system will accelerate core uncovery process and lead to a possible core melt.

The severe accident research, which in the last 20-30 years has been extensively carried out for light water reactors, has recently been performed for SMR reactors with codes such as RELAP5/SCDAP [2], MELCOR [3], [4] and ASTEC [5], [6]. Here, the ASYST code [7] is used for the calculation of a severe accident, while the RELAP5/Mod3.3 [8] and GOTHIC [9] codes are used for the analysis of a design basis sequence. The RELAP5 code is used for the analysis of the processes in the reactor vessel and the GOTHIC code for the analysis of the processes in the containment. RELAP5 and GOTHIC are explicitly coupled and exchange mass and energy at the points of contact between the reactor vessel and the containment. This approach has already been used for previous analyzes of IRIS [10], [11] and other types of reactors [12], [13].

II. IRIS design of passive safety systems

Descriptions of reactor coolant system (RCS) and the containment, as well as their nodalizations, are presented in [1]. The unique feature of the IRIS reactor is its arrangement of the passive safety systems; whose operation enables efficient cooling of the core in the case of a loss-of-coolant accident. Therefore, a more

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detailed description of these systems is given here.

Removal of decay heat is enabled by means of natural circulation between the core and steam generators (SG). The heat is removed by the passive emergency heat removal system through the SG heat transfer surface. The EHRS consists of four independent subsystems, connected to SG feed and steam lines, where each has a heat exchanger immersed in a water pool of the refuelling water storage tank (RWST). In the initial IRIS design the heat exchangers had horizontal pipes. In the newest design they have vertical pipes. By removing heat from the reactor vessel, the EHRS lowers the pressure of the primary system, thus reduces coolant loss and enables earlier activation of water injection from dedicated tanks (emergency boration and gravity make-up tanks) and the reactor cavity inside the containment. These water reservoirs are connected to the RPV via DVI lines. The emergency boration tanks (EBT) are at the RPV pressure, while the LGMS tanks are at the containment pressure. Thus, EBT tanks start to inject water immediately at the start of the accident. Water from the LGMS tanks will be injected when the primary pressure drops below the containment pressure.

There is also possibility to inject water from the suppression pools directly to the RPV using separate DVI lines. This is enabled using the new design of the safety systems where the LGMS tank is split in two separate water tanks. The layout of the passive safety systems for the two configurations analyzed in the paper is shown in figures 1 and 2. The early design with a dual function long-term gravity make-up system, and the emergency heat removal system with horizontally positioned heat exchangers in the refuelling water storage tank is shown in Figure 1, while the newer design with the suppression pools connected to separate DVI lines, and the EHRS with the vertical heat exchanges in the RWST is shown in Figure 2.

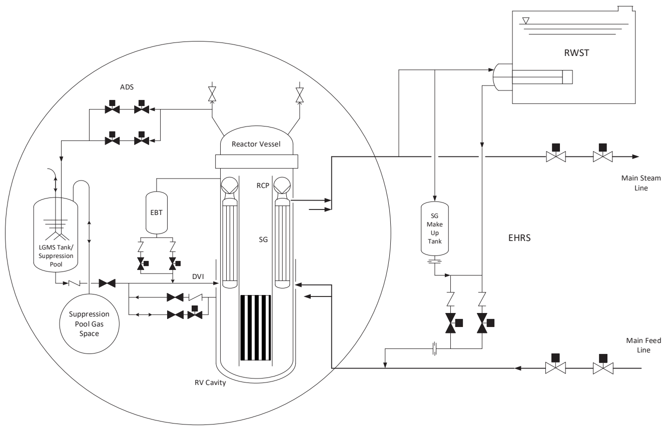

Fig. 1. Design of passive safety systems for the older configuration

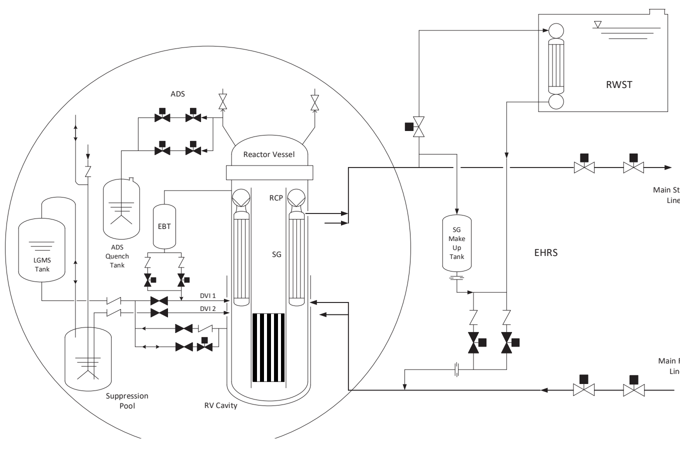

Fig. 2. Design of passive safety systems for the newer configuration

III. Numerical Model and Nodalization

RELAP5 model is shown in Figure 3. The RELAP5 input data set is quite large. The IRIS reactor has been developed over the years and its design and nodalization have changed accordingly. The discretization of the reactor system components is sufficiently detailed in order to take into account all the important thermal-hydraulic phenomena. The total number of volumes is 1673 and junctions 1724. Most of the calculation nodes have a linear size in the range of 0.2 m to 0.5 m. All relevant heat structures are modelled. The number of heat structures in the current nodalization is 625 with the total number of mesh points being 3574.

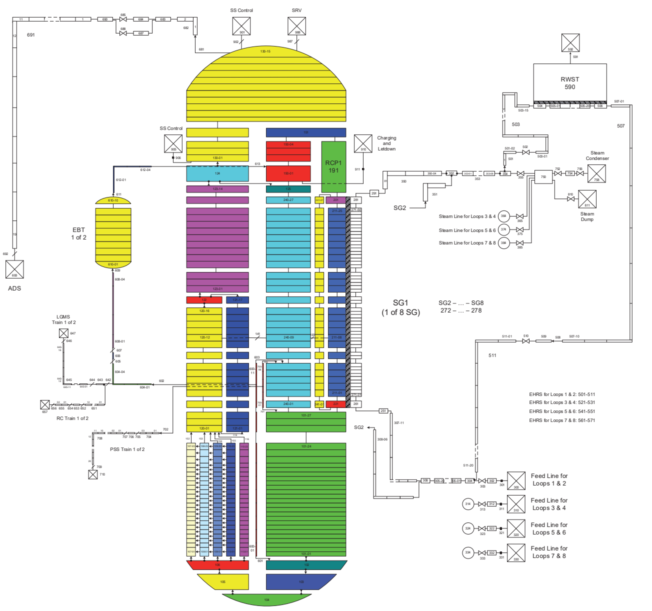

Fig. 3. RELAP5 model of the reactor vessel and RWST connections

IV Iris Behaviour during a LOCA Event

The strategy to mitigate LOCA implemented in the IRIS reactor is shown in Figure 4. It should be emphasized though that this is the theoretical sequence of events. The actual accident sequence depends on the specific boundary conditions as it is shown in the next section.

After the LOCA initiation, the reactor vessel depressurizes and loses mass to the containment vessel causing its pressure to increase. The RPV depressurization is supported by EHRS heat removal and release of steam to the suppression pool by the automatic depressurization system (ADS). At the end of the blowdown phase, pressures of the RPV and the containment become equal. The break flow stops and gravity makeup of borated water to the RPV from the suppression pool becomes available. The coupled RPV/containment system is then depressurized by continued operation of the EHRS. In this phase the break flow actually reverses since heat is removed directly from inside the reactor vessel and not from the containment. This, however depends on the position of the break. If the break position is low at the RPV wall, then the flow reversal is less likely. As the containment pressure is reduced, a portion of suppression pool water is pushed out through the vents and assists in flooding the vessel cavity. The depressurization phase is followed by the long term cooling phase where the RPV and containment pressures are slowly reduced as the core decay heat decreases.

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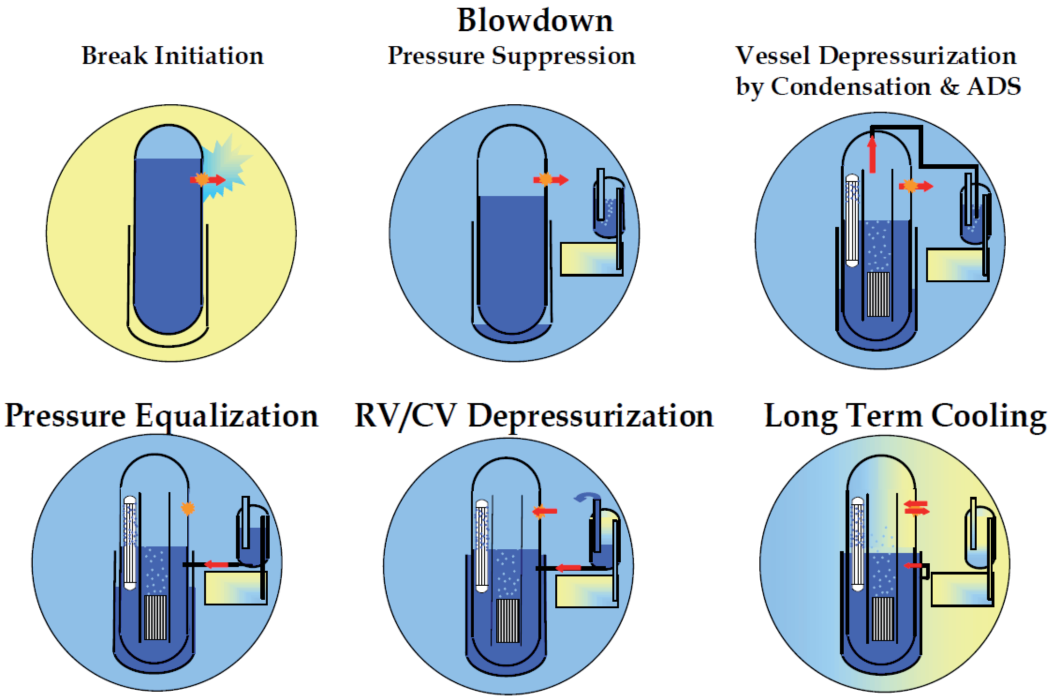

Fig. 4. Accident sequence during a design basis LOCA

V. Analysis of a Severe Accident

The rupture of the pipes of the existing pipelines will not have serious consequences and the accident will be a design basis event [1]. In order to force the core damage one needs to assume hypothetical rupture of the reactor vessel at the lower elevation, or the unavailability of the passive safety systems. The severe accident sequence is initiated by opening a 4-inch break at the reactor vessel, with a surface area of $8.1 \cdot 10^{-3}$ m^2 (which corresponds to the cross-section of the widest pipe connected to the reactor vessel), in the lower plenum on the RELAP5 volume 102.

Following the initiating event, the LM (LOCA mitigation) signal is rapidly actuated on a coincident low pressurizer pressure and high containment pressure. On a LM signal the following actions are initiated:

1. The reactor and the reactor coolant pumps are tripped,
2. Containment penetrations are isolated,
3. The four EHRS subsystems are actuated by closing the main feed and steam isolation valves, and by opening the fail-open valves in the EHRS return lines from the EHRS heat exchangers connected to the SG feedlines,
4. The ADS valves and the emergency boration tank discharge isolation valves are actuated to open.

Reactor coolant is discharged rapidly through the break. Passive safety systems are working but the water injection is insufficient to maintain the core water level at the value that would support adequate core cooling. The drying of the core causes the temperature to increase, Figure 5. At the beginning of the transient, up to 750 seconds, the temperature decreases because the EHRS removes more heat through the steam generators than the decay heat is produced in the core. After that, as the coolant is lost from the reactor vessel and the natural circulation slows down, thermal-hydraulic conditions are reversed and less heat is removed than is produced in the core. Due to the intensive oxidation of the Zircaloy cladding there is a fast temperature increase at 2500 s. The water injection from the safety systems is not enough to ensure long-term cooling. More heat is produced inside the core than is removed by the steam. Thus, the temperature does not decrease which prolongs cladding oxidation and the core degradation.

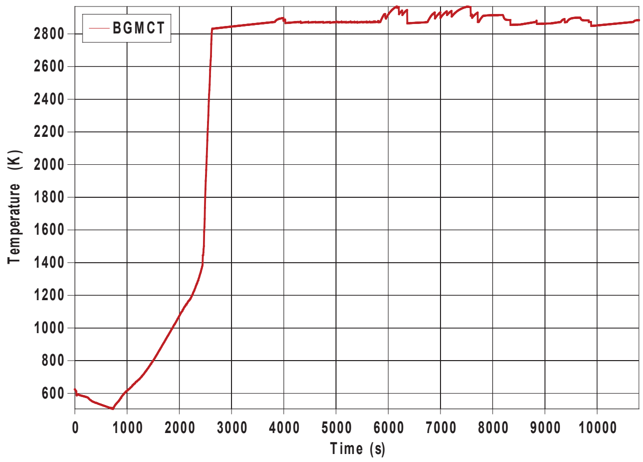

Fig. 5. Maximum fuel rod cladding temperature

Production of hydrogen is shown in Figure 6. The oxidation rate depends on several factors: the core temperature, the amount of water and steam, and the state of fuel assemblies. The oxidation process starts at 2100 s when the maximum core cladding temperature reaches 1200 K. In the next 400 s, the rate of oxidation is low, but after that it increases significantly because of its exponential dependence on the temperature and the recrystallization of the ZrO_2 layer on the outside cladding surface at 1853 K, which facilitates steam penetration through the cladding. The total calculated mass of hydrogen is 135 kg, which corresponds to 66% oxidation of the Zircaloy cladding. Theoretically, the total mass of hydrogen produced would be 205 kg if the fuel rod claddings were completely oxidized.

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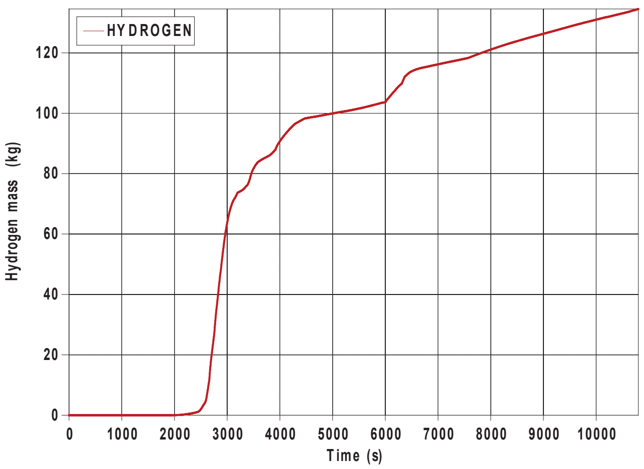

Fig. 6. Production of hydrogen

The extent of core degradation during the progression of a severe accident is described by the equivalent radius of the molten pool of core material, Figure 7. Damage kinetics assumes a change of core state from the intact fuel rods, through partially liquefied debris to local melting, which finally evolves into a molten pool. In the period between 3800 s and 7500 s, the volume of the molten pool increases continuously due to the melting of core structures. This process, which lasts about an hour, is partially stopped by relocation of the molten material into the lower head, but the degradation process continues because the core temperature does not decrease. Relocation of the corium to the lower plenum occurs after the melt breaks through the core barrel cylindrical structure, at 7500 s. The severe accident progresses rapidly. The fuel assemblies almost completely melted in a time span of 3 hours. This example shows that an accident with severe consequences is possible, even with passive safety systems available.

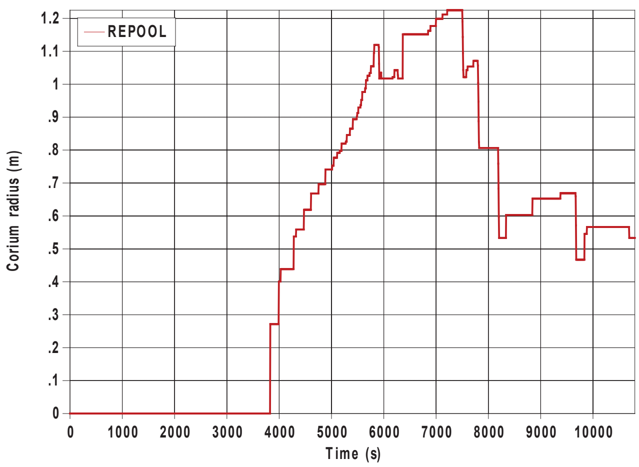

Fig. 7. Equivalent radius of the molten pool in the core

Figure 8 shows the water distribution in the reactor cavity, the containment dry compartment and in the pressure suppression system (PSS) tank. Water discharged from the RCS through the break enters into the cavity. Because the RCS pressure is higher than the cavity pressure this water does not flow back into the primary system, although the cavity water level becomes higher than the break elevation already at 200 s. Therefore, the core is rapidly drying out, while the cavity is being filled with water. When the water level in the cavity reaches the dry compartment bottom elevation, water spills into the containment. Injection of water from the PSS tanks into the RCS after the pressure equalization leads to the water level decrease in the PSS tanks.

The new arrangement of the passive safety systems with separate DVI lines connected to the suppression pools ensures longer period of successful core cooling. More water is injected into the reactor vessel in the new model, which is a direct consequence of the better cooling by the EHRS. In addition to the fact that in the new model the water comes from the PSS tanks, which did not exist before, the amount of water injected from the LGMS is also larger. Nevertheless, the new design of the emergency heat removal and the pressure suppression systems cannot prevent a severe accident in the event of a RPV wall rupture at a low elevation, Figure 9. However, what it does provide is additional time for the operator to start active water replenishment systems. Their equipment is not of safety class, so their operation is not guaranteed in all the environmental conditions, but sufficient spare equipment can ensure high availability of active systems.

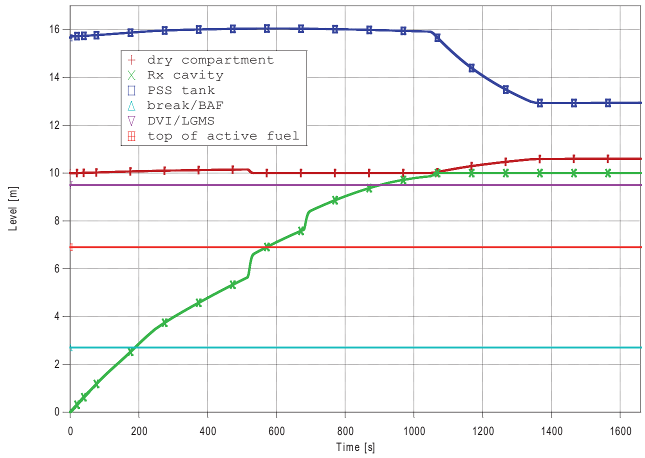

Fig. 8. Reactor cavity, dry compartment and PSS tank water elevations

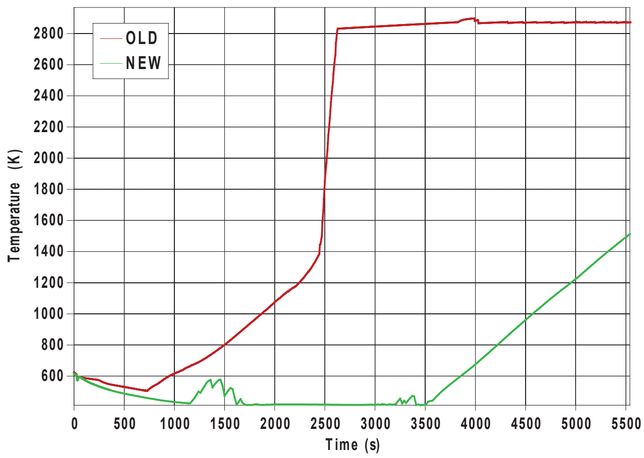

Fig. 9. Maximum fuel rod cladding temperature, comparison of the calculations with the older and newer passive safety system configurations

VI. Conclusion

Core damage frequency for the IRIS reactor is very low. Nevertheless, severe accident study has to be performed to evaluate the relocation and evacuation measures. Hypothetical reactor pressure vessel break at the active core bottom elevation with passive safety systems available is the accident scenario analyzed in this paper.

Results show that injection of water from the passive safety systems is not enough to quench the core. Too much water is lost through the break to maintain the water level in the core at an acceptable level. A large amount of water accumulated in the cavity is unavailable for core cooling because the RCS pressure is higher

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than the cavity pressure and, therefore, no fluid can enter from the cavity to the reactor vessel through the break, so the core dries out in a very short time. Fuel assembly temperatures rise to almost 3000 K which leads to enhanced oxidation and core damage. The core melting process takes place rapidly, and given the sufficient amount of water that evaporates in the core, the fuel rod cladding oxidation will result in a large amount of hydrogen produced. By improving the safety systems, the time period during which decay heat is successfully removed increases, whereby the water tanks in the containment play a key role in supplying the water.

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