Development of the Numerical Model of the IRIS Reactor for Severe Accident Analysis

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ABSTRACT

The interest in the IRIS (International Reactor Innovative and Secure) reactor is revived by today's popularity of small modular reactors. The IRIS reactor project, led by Westinghouse Electric Corporation, was active in the first decade of the 21st century. Different groups of institutions such as nuclear manufacturers, academic institutions, national laboratories, etc. from 10 countries around the world participated in the IRIS team.

IRIS is an integral, modular, medium sized (1000 MWt) pressurized water reactor. The IRIS reactor pressure vessel houses, beside the reactor core, also other major reactor coolant system components such as the pressurizer, reactor coolant pumps and steam generators. The lack of large pipes ensures high safety of the IRIS power plant and eliminates many causes of major accidents. This principle is known as “safety-by-design” approach.

Polytechnic of Milan and the University of Zagreb were leading institutions in performing safety analyses for the IRIS reactor. The explicit coupling of RELAP5 and GOTHIC codes has been set up to cover the sequence of most probable LOCA transient events. This was necessary because the reactor vessel and the containment, once when the LOCA is initiated, become one hydraulic system with strong interaction. They exchange mass and energy which affects both systems in short time period and therefore cannot be treated separately as in a conservative analysis of a classic PWR nuclear power plant. In addition, the ASYST code model was recently developed to cover possible severe accident sequences. The core heat structures were replaced with SCDAP components to simulate core degradation. A couple of different GOTHIC models were developed to represent various arrangements of passive safety systems. A steady state analysis was performed to confirm the applicability of the IRIS numerical model in the safety analyses.

Keywords: *IRIS reactor, severe accident, RELAP5, GOTHIC, ASYST*

1 INTRODUCTION

IRIS (International Reactor Innovative and Secure) is an integral, medium power (1000 MWt), light water reactor [1], [2]. It is characterized by enhanced safety and improved economics. Although it features innovative and advanced engineering, it is based on the proven technology of pressurized water reactors (PWR). A “safety-by-design” approach is implemented in the IRIS system which objective is to prevent severe accidents from occurring rather than to mitigate their consequences. In order to enable such a strategy, all components of the reactor coolant system (RCS) are enclosed in a reactor vessel which eliminates large break loss-of-coolant accidents (LOCA) since there are no large piping or vessel penetrations. If the water leaks from the reactor vessel, the small containment will provide the necessary back-pressure to limit the loss of water, and in the later phase of the accident, the long-term cooling due to the innovative strategy of depressurization and retention of water inside the vessel.

Recently, there has been a considerable interest in small and medium sized modular reactors for many reasons: enhanced safety performance by using passive safety systems, the capability for flexible power generation, high availability, lower price comparing to larger units of nuclear power plants of second and third generations due to modular construction of several smaller units, etc. [3], [4]. Passive safety systems, such as the passive decay heat removal and the long-term gravity core cooling systems, used in the IRIS reactor are common in many small modular reactor (SMR) designs today, in a similar or slightly modified form [5], [6].

The IRIS reactor project, led by Westinghouse Electric Corporation, was active in the first decade of the 21st century. During the implementation of the project, 22 organizations from 10 countries participated in the IRIS team [7]. These organizations represented leading nuclear manufacturers, academic institutions, national laboratories and power producers, and covered different phases of the project (plant components design, licensing, testing, instrumentation and control, safety analyses, advanced cores, neutronics, source term, radioactive waste management, maintenance, utility perspective, etc.).

One of the more important activities of the IRIS project were the safety reactor analyses, coordinated by the Polytechnic of Milan and the University of Zagreb. As part of this activity, a preliminary safety assessment report (PSAR) was issued in 2003 [8]. A standard set of accidents was analyzed according to the requirements of the Chapter 15 of the Safety Analysis Report. In the section related to loss-of-coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary, two failures were singled out as limiting LOCA events: double ended ruptures of a chemical volume and control system (CVCS) 4-inch pipe and the direct vessel injection (DVI) 2-inch pipe. The analyses have shown that the 10 CFR 50.46 acceptance criteria will not be exceeded for these small break (SB) LOCA scenarios.

Development of the IRIS numerical models for the codes RELAP5 [9], GOTHIC [10] and ASYST [11] is presented in the paper. The RELAP5 code is used for the analysis of the processes in the reactor vessel, the GOTHIC code for the analysis of the processes in the containment and the ASYST code for the severe accident calculation. RELAP5 and GOTHIC codes are explicitly coupled in order to correctly calculate mass and energy exchange between the reactor vessel and the containment [12]. A steady state analysis is performed, and the calculations of CVCS and DVI pipe breaks are repeated, in order to qualify the model at both levels: the steady state and the transient level.

2 IRIS SYSTEM DESIGN

IRIS consists of eight internal cooling loops. It has eight small, spool type, reactor coolant pumps and eight modular, helical coil, once through steam generators. The pressurizer is located in the reactor pressure vessel (RPV) upper head. The steel reflector surrounds the core and improves neutron economy, as well as it provides additional internal shielding. IRIS vessel is shown in Figure 1.

The steel containment of the IRIS reactor is spherical in shape, Figure 2, and relatively small in volume due to the integral design of the reactor vessel. Since the containment volume is reduced, compared to high power PWR power plants, its design pressure is high, 1.4 MPa. The automatic depressurization system (ADS), emergency boration tanks (EBT), pressure suppression pools, the long-term gravity make-up system (LGMS) and the associate pipelines, valves and other necessary supporting equipment are located inside the containment. The emergency heat removal system (EHRS) connection to feed and steam lines is made outside the containment, thus the refuelling water storage tank (RWST), containing EHRS heat exchangers, is also placed outside the containment.

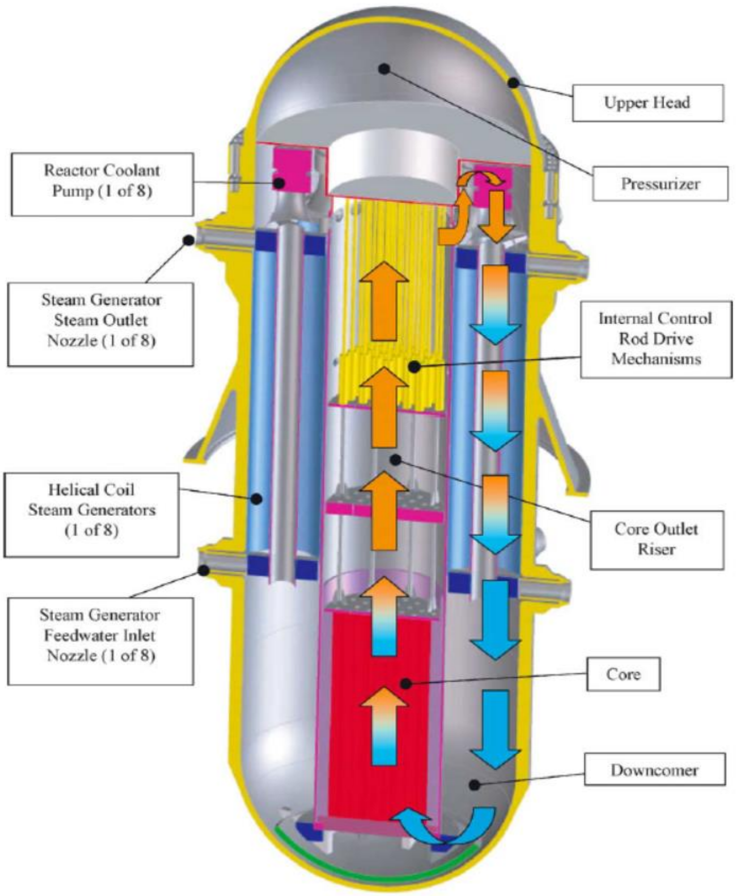

Figure 1: Reactor pressure vessel of the IRIS reactor

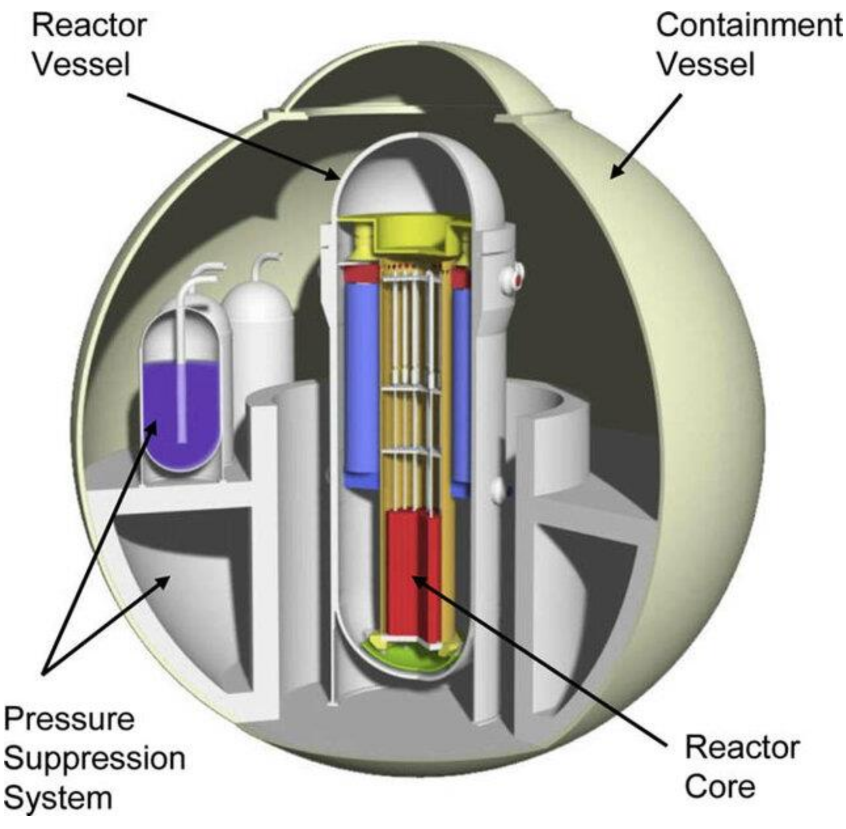

Figure 2: Layout of the RPV and the containment

3 NODALIZATION OF THE REACTOR VESSEL AND THE CONTAINMENT

3.1 Calculation models of primary and secondary systems

The RELAP5 nodalization is shown in Figure 3. The nodalization contains numerical models of reactor core flow channels, downcomer, lower plenum, core bypass, upper plenum, pressurizer, reactor coolant pumps, steam generators, emergency boration tanks, emergency heat removal system, ADS line, feedwater lines, steam lines and the DVI connections to LGMS tanks, reactor cavity and the pressure suppression system (PSS) tanks. The model also includes necessary reactor protection system functions for trips and their actuations.

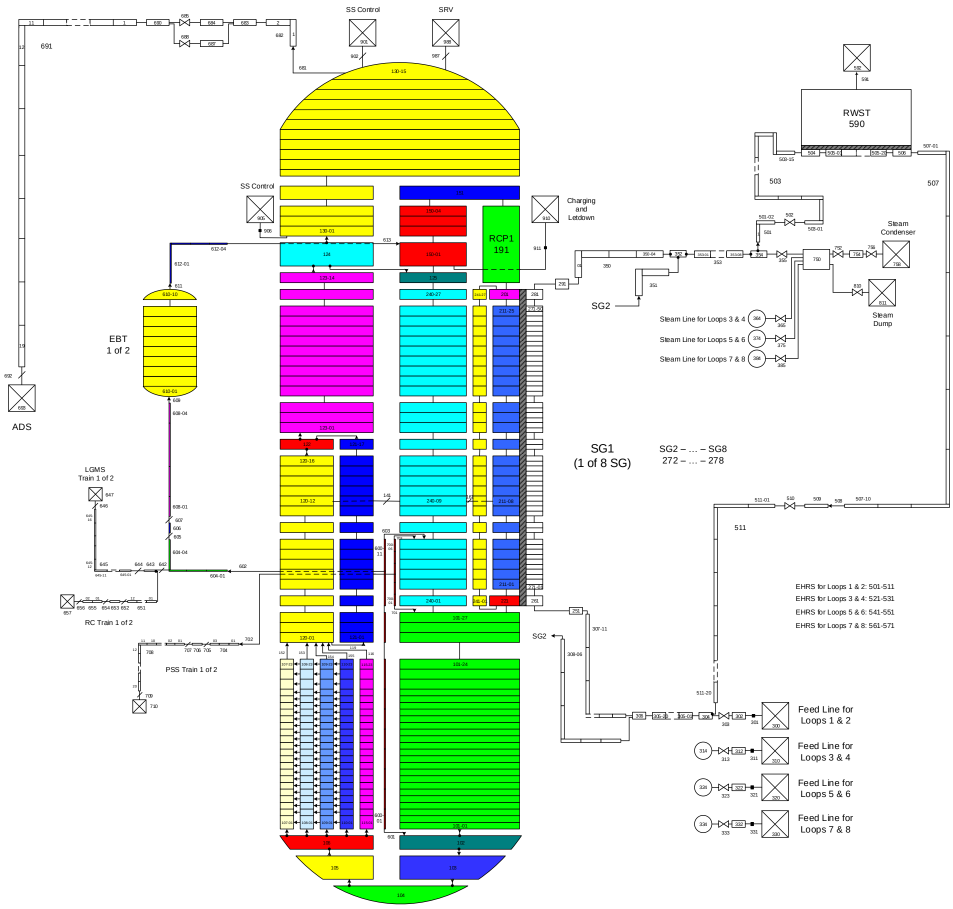

Figure 3: RELAP5 nodalization

The core structures: fuel rods, control rods, grid spacers and the core barrel are modelled with the ASYST code responsible for the simulation of a severe accident. Fuel assemblies are divided into four regions, Figure 4. The fuel assemblies in which the control rods are located are marked with an x. The radial division enables a more accurate temperature calculation that takes into account the differences in the fuel temperature in the centre and the periphery of the core. This temperature profile basically causes the central fuel assemblies to start melting before the assemblies in the outer core region. Each of the four ASYST fuel rod components is located in a separate RELAP5 thermal-hydraulic channel. The channels are interconnected by radial junctions to ensure lateral coolant flow in the event of a channel blockage due to core melting.

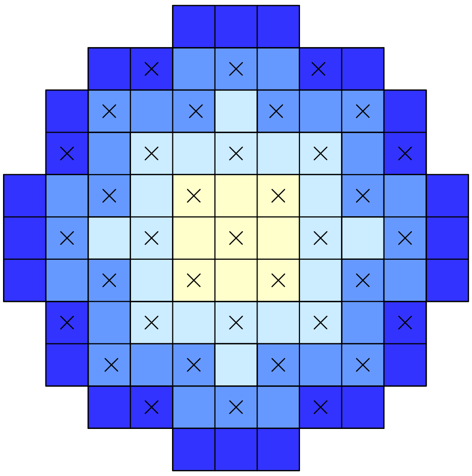

Figure 4: Radial distribution of fuel assemblies inside the core

3.2 Containment models

The GOTHIC nodalizations of the containment are shown in figures 5 and 6. Nine control volumes and 15 flow paths are used in the original containment model, Figure 5. Drywell containment space is split in two parts (volumes 1 and 2) connected by the flow path 4. At the bottom, they are connected to the reactor cavity with two flow paths to simulate mixing between the volumes. Two LGMS tanks/suppression pools are connected to the air spaces (tanks) below them. Volumes 5 and 7 simulate vent pipes that connect the suppression pools to the containment atmosphere. The flow path 1 is used for break modelling. It connects reactor vessel with the cavity in which the coolant, released from the primary system, accumulates. Flow paths 8 and 9 are used to connect gravity make-up lines to the LGMS tanks. Flow paths 10 and 11 are used to connect gravity make-up lines to the reactor cavity. All those flow paths are connected to the DVI lines. The numbers at the other end of the lines represent RELAP5 time-dependent volumes, that is, boundary connections to the RELAP5 model.

The second nodalization which corresponds to the latest containment configuration is shown in Figure 6. The main difference to the old model is splitting of each of the two LGMS tanks/suppression pools in two separate water pools (4, 6, 8, 9). The suppression pools can also be connected by flow paths 25 and 26 to a separate direct vessel injection lines feeding water directly to the downcomer of the reactor vessel.

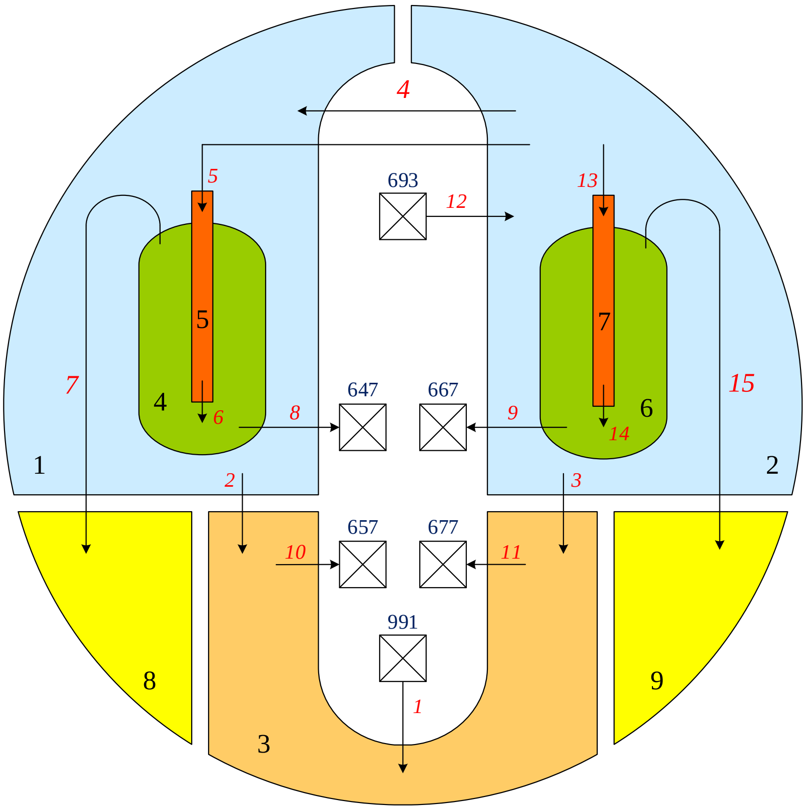

Figure 5: GOTHIC containment nodalization for the older safety system configuration

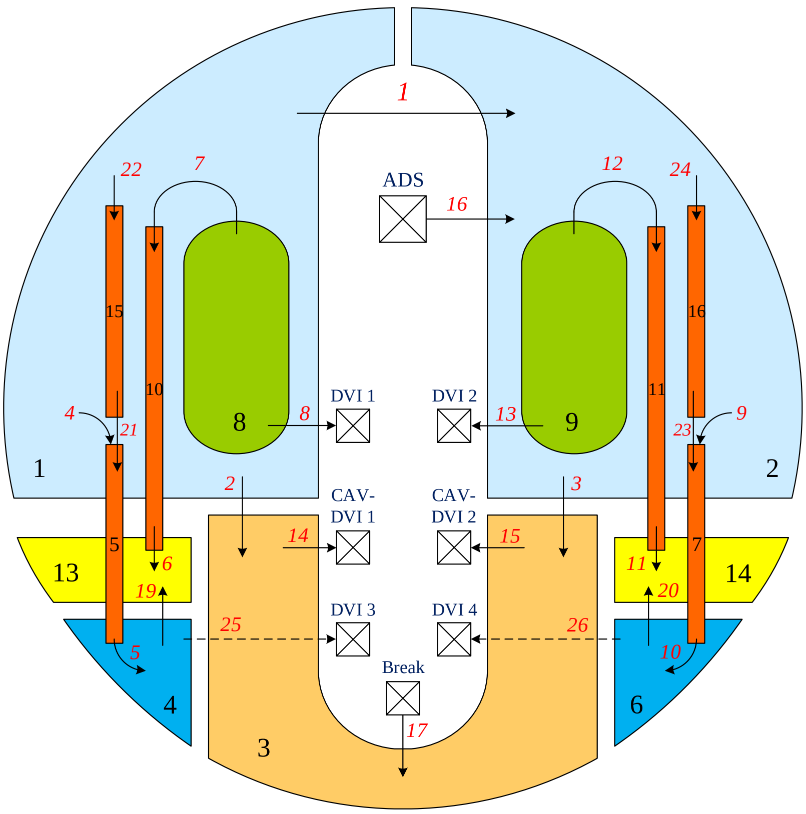

Figure 6: GOTHIC containment nodalization for the newer safety system configuration

4 STEADY STATE CALCULATION

In order to tune up the model and achieve correct initial conditions for the transient analysis, a steady state calculation for the reactor coolant system was performed with the RELAP5 code. The calculation simulated 2000 s of steady state operation with a time step of 0.05 s. Results are shown in Table 1. Comparison with the operational plant data shows that the model is qualified at the steady state level since the differences between the most important calculated parameters (pressurizer and SG pressures, reactor coolant and steam flow rates and temperatures, power transferred in steam generators) and reference values are less than 1%.

Table 1: Results of the steady state calculation

Parameter	**Reference**	**RELAP5**	**Difference [%]**
Pressurizer pressure [MPa]	15.5	15.512	0.08
Steam generator pressure [MPa]	5.8	5.784 – 5.809	–0.28 – 0.16
Core inlet temperature [K]	565.2	564.44	–0.13
Core outlet temperature [K]	603.15	602.18	–0.16
Reactor vessel mass flow [kg/s]	4707	4724	0.36
Core mass flow [kg/s]	4504	4524	0.44
SG steam exit temperature [K]	590.2	589.45 – 591.43	–0.13 – 0.21
Total steam flow [kg/s]	502.8	502.78	–0.004
Core pressure drop [kPa]	52.0	54.08	4.0
SG pressure drop [kPa] – primary	72.0	71.69	–0.43
SG pressure drop [kPa] – secondary	296.0	297.01	0.34
Total SG power [MW]	1002.0	1001.2	–0.08

5 TRANSIENT ANALYSIS

The calculations of the two limiting breaks at existing pipes are performed again with the intention of qualifying the numerical model at the transient level. The first break is the complete rupture of a CVCS 4-inch pipe, connected to the upper annular pump suction plenum of the reactor vessel. The second break is the double ended rupture of one DVI 2-inch pipe located in the lower annular region surrounding the steam generators.

The reactor coolant system pressure decreases due to the loss of coolant at the break and through the ADS, and due to heat removal by the EHRS. Natural circulation is established between the core and the steam generators; the steam condenses on the steam generator tube surface. More heat is removed by the EHRS via the steam generators than is produced in the core. Thus, the core temperature is decreasing. While the RCS pressure is decreasing, the pressure in the containment is increasing. When the containment pressure becomes greater than the pressure in the DVI lines, the safety injection from the long-term gravity make-up system tanks is activated.

The pressure inside the reactor pressure vessel decreases faster for the CVCS line break (Figure 7) because that pipe has a larger diameter than the DVI pipe, so more fluid is lost from the RPV to the containment. At 2500 s for the DVI pipe break, the reactor coolant system pressure decreases below the containment pressure and the higher pressure in the suppression pool gas space forces water to the reactor coolant system through the intact make-up line from the LGMS tank to the reactor vessel. There is no injection from the LGMS tank in the case of the CVCS line break.

Water injection from the emergency boration tanks compensates for water loss through the break. In the case of a CVCS pipe break the collapsed core water level is higher than in the case of a DVI pipe break (Figure 8). This is because the water from the emergency boration tank connected to the damaged DVI pipe is directly discharged into the containment, which means that only one EBT is available, compared to the first case where both tanks are available. Both cases are design basis events since the acceptance criterion for the LOCA accident is met: the calculated maximum fuel rod cladding temperature is well below the threshold value of 1477 K (Figure 9) as defined by 10 CFR 50.46 LOCA regulations.

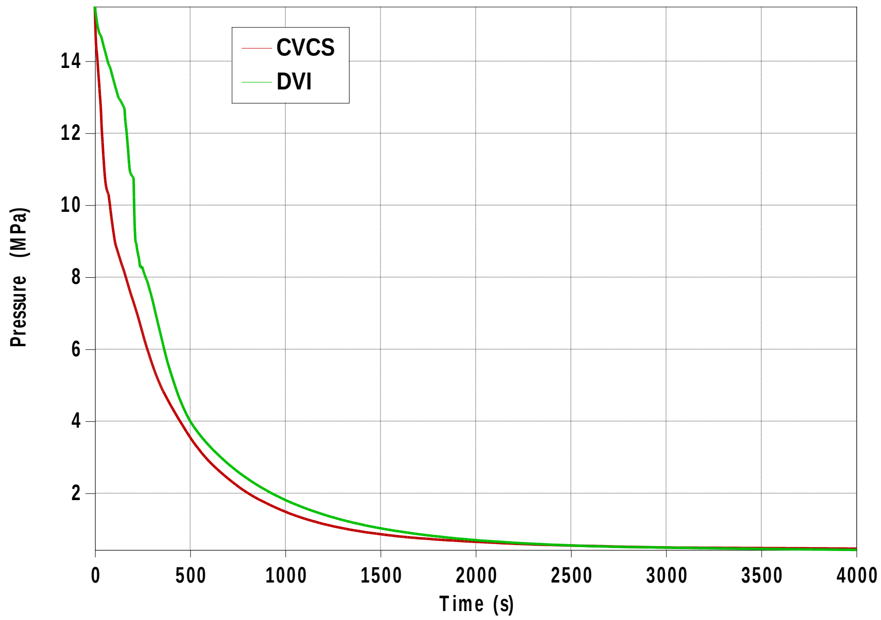

Figure 7: Pressurizer pressure

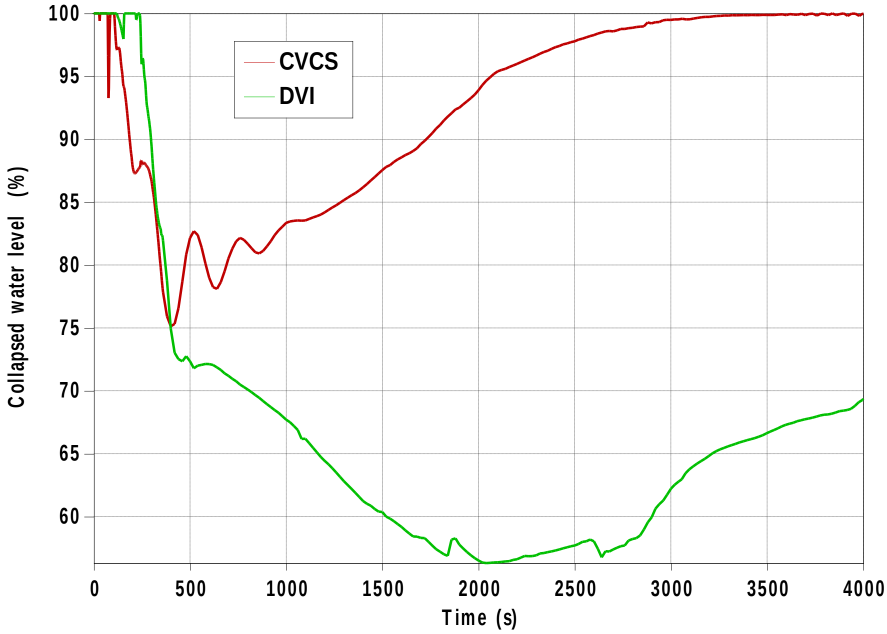

Figure 8: Collapsed core water level

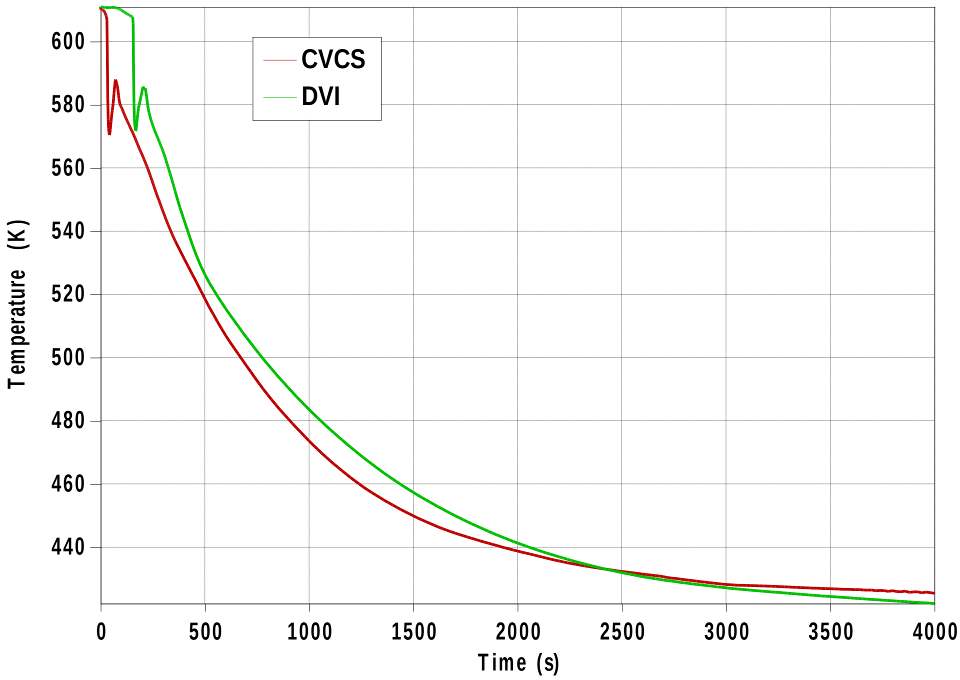

Figure 9 Maximum fuel rod cladding temperature

6 CONCLUSION

The best-estimate computer codes are needed for comprehensive safety analyses of complex thermal-hydraulic systems such as IRIS. The physical processes of natural circulation, core water level tracking, thermal-hydraulic coupling of the reactor vessel and the containment are correctly analyzed only if detailed nodalizations of reactor coolant system and containment are used. The transient calculations of CVCS and DVI pipe breaks show that overheating of the core and more serious consequences will not occur with proper operation of passive safety systems. The injection of borated water from the emergency boration tanks is a sufficient measure to control thermal-hydraulic conditions in the reactor vessel in the case of rupture of existing pipelines. Injecting water from the containment tanks (LGMS tanks, PSS pools) is a secondary safety measure since the main intention of the IRIS design is to maintain reactor coolant system inventory in the case of a design basis accident, rather than to rely on water injection from the external systems.

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