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Research Article

Decay Heat Removal and Transient Analysis in Accidental Conditions in the EFIT Reactor

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The development of a conceptual design of an industrial-scale transmutation facility (EFIT) of several 100 MW thermal power based on accelerator-driven system (ADS) is addressed in the frame of the European EUROTRANS Integral Project. In normal operation, the core power of EFIT reactor is removed through steam generators by four secondary loops fed by water. A safety-related decay heat removal (DHR) system provided with four independent inherently safe loops is installed in the primary vessel to remove the decay heat by natural convection circulation under accidental conditions which are caused by a loss-of-heat sink (LOHS). In order to confirm the adequacy of the adopted solution for decay heat removal in accidental conditions, some multi-D analyses have been carried out with the SIMMER-III code. The results of the SIMMER-III code have been then used to support the RELAP5 1D representation of the natural circulation flow paths in the reactor vessel. Finally, the thermal-hydraulic RELAP5 code has been employed for the analysis of LOHS accidental scenarios.

1. INTRODUCTION

Within the EURATOM Sixth Framework Program (FP6), the EUROTRANS integrated project [1] is expected to provide a significant contribution to the demonstration of the industrial transmutation through the accelerator-driven system route. The goal will be reached through two phases: the realization of a detailed design for an experimental facility of 50 to 100 MWth power which shows the technical feasibility of transmutation in an ADS (XT-ADS), and at the same time, the development of a conceptual design of a generic European Transmutation Demonstrator (ETD) of several hundreds of MWth, to be realized in the long term (EFIT).

The EFIT reactor should be able to produce energy at reasonable costs with enhanced transmutation performances, maintaining as much as possible the high safety level. Modifications introduced in the plant layout with respect to smaller facilities contribute for a more compact primary system and a higher core power density: the elimination of intermediate loops by installation of steam generators inside the primary vessel and the implementation of mechanical pumps for forced circulation.

In order to assure a high safety level, a DHR system provided with four independent inherently safe loops is installed in the primary vessel to remove the decay heat by natural convection circulation under accidental conditions which are caused by a loss of heat removal by the secondary side through the steam generators.

In the present study, performed in the frame of a collaboration between ENEA and the University of Palermo, the multi-D SIMMER-III code has been applied to confirm the adequacy of the DHR system design and for the calibration of the 1D RELAP5 model to be used for T/H transient analysis of EFIT.

2. THE EFIT REACTOR

The EFIT reactor [2] is a pool-type reactor which uses pure melted lead as primary coolant in forced circulation by

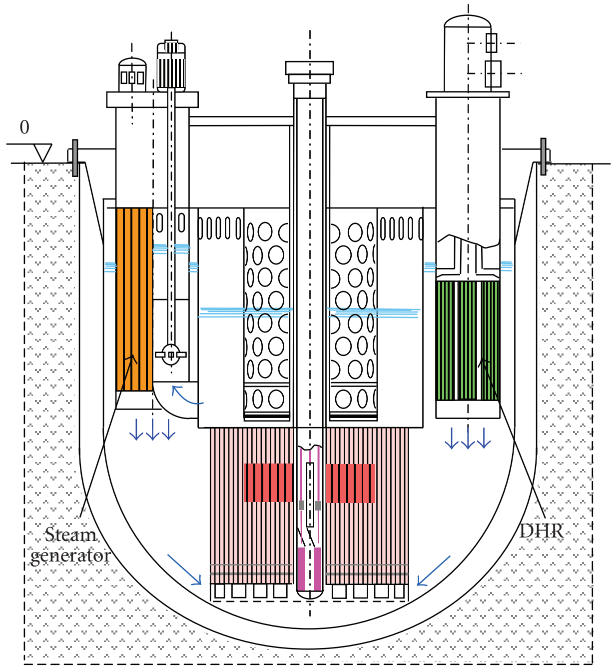


Figure 1: Scheme of the EFIT reactor block.

means of 4 mechanical pumps placed in the hot collector zone. The reactor thermal power, which is approximately 400 MW, is removed in normal operation by 8 helical-coil tube bundle steam generators, which are located in the upper part of the primary vessel to enhance natural circulation in case of loss of primary pumps. The scheme of the EFIT reactor block is depicted in Figure 1.

The main thermal-hydraulic parameters of the EFIT reactor are given in Table 1. The core power is removed by 32.3 tons of lead working between 673 K and 753 K. The primary coolant at the core outlet is sucked by pumps straight inside the steam generators, then comes out and flows towards the vessel bottom for coming back and cooling the core. So the vessel will be in contact with the coolant at its minimum temperature (673 K), avoiding mechanical stresses due to high lead temperature at core outlet.

In the upper annular space between the inner cylindrical vessel and the main reactor vessel, the heat exchangers of the DHR system are placed for core decay heat removal under accidental conditions.

2.1. *The decay heat removal system*

The DHR system is conceived for inherently safe decay heat removal in accidental conditions by means of natural circulation and with passive mode actuation. The system consists of 4 independent loops filled with organic diathermic fluid (oil) that dissipate the decay heat to the atmosphere by natural convection circulation. The functional scheme of one DHR loop is depicted in Figure 2. Each loop consists of a dip cooler (decay heat eXchanger, (DHX)) immersed in the lead, where the oil partially vaporizes (oil boiling point determined by superimposed pressure of inert gas), and an air-vapour

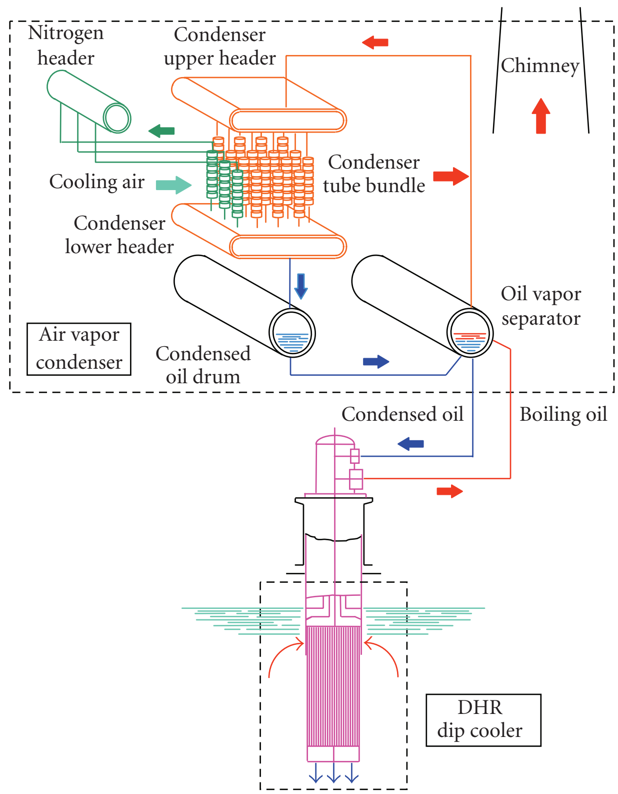


Figure 2: Functional scheme of one DHR loop.

Table 1: Main EFIT thermal-hydraulic parameters.

Reactor power (MW)	384
Core lead mass flow rate (kg/s)	32300
Core lead temperature (inlet/outlet, K)	673/753
Primary circuit total pressure drop (MPa)	About 0.08
Core pressure drop (MPa)	About 0.04
Secondary steam pressure (MPa)	14
Secondary steam temperature (K)	723
Feedwater temperature (K)	608
DHR system power (4 units, MW)	26.6

condenser with stack chimney and interconnecting piping. The lead enters radially the DHX through a window in the cylindrical shell and leaves axially the DHX through the open bottom end of the shell.

At normal operating conditions, the oil is below its boiling point and the DHR system removes the heat losses from steam generators and inner vessel (a few 100 kW) to keep cold the upper part of the reactor vessel. In accidental conditions (e.g., LOHS), when the lead temperature increases in the annular space where the DHXs are located, the oil starts to boil enhancing heat transfer in the DHR and thus favouring natural circulation on both primary and secondary sides. Each DHX is rated at approximately 6.7 MW in decay heat removal conditions.

3. SIMULATION OF DHR OPERATION WITH THE SIMMER-III CODE

The SIMMER-III code [3], jointly developed by JNC (J), FZK (D), and CEA (F), is an advanced safety analysis computer code originally developed to investigate postulated core disruptive accidents in liquid-metal fast reactors. SIMMER-III is a two-dimensional, three-velocity field, multiphase, multicomponent, Eulerian, fluid-dynamics code coupled with a space-dependent neutron kinetics model. By integrating all the original physical models, SIMMER-III is now applicable to a large variety of reactor calculations and other complex multiphase flow problems. The code is provided with a turbulent diffusion model to evaluate the effects of recirculating flows. This model, which is simpler than classical turbulent models of CFD codes, has been applied in the present analysis.

The EFIT reactor has been modelled in 2D cylindrical geometry with SIMMER-III (see Figure 3). The core is represented by 6 radial fuel rod rings plus the reflector and bypass zones, and discretized in 14 axial nodes. The primary pump section, the steam generators, and the DHR heat exchangers are represented by annular zones with equivalent cross flow area. The simulation of DHR loops is limited to the in-vessel heat exchangers, and the power removal is calculated as a function of the temperature difference between the primary flowing lead and the boiling oil in the secondary side. As a conservative assumption in the accident analysis, the DHR system is considered in degraded conditions with 3 out of 4 loops in service.

A protected LOHS scenario has been simulated with the SIMMER-III code in order to evaluate the capability of the DHR system for decay heat removal in transient conditions by natural circulation in the primary system. Stagnant lead is assumed inside the primary vessel at transient initiation with simultaneous loss of primary pumps and steam generator heat removal function. Initial lead and core temperatures (clad and fuel) are set according to RELAP5 steady-state results for reactor operation at 384 MW nominal power.

The distribution of lead temperature in the primary vessel calculated by SIMMER-III is represented in Figure 4 from the beginning up to 1 hour transient. Initially, the release of heat from hot fuel rods leads to lead heatup and strong recirculation in the upper plenum. Due to the different density and gravity effect, hot lead flowing down at the steam generator outlet moves upward in the annular external region between the inner cylindrical vessel and the reactor vessel where the DHR heat exchangers are located. A natural circulation then starts inside the DHX where the lead is cooled by efficient heat transfer to the boiling oil of the secondary circuit. A mixing region of cold and hot fluids is evidenced below the DHX and the steam generator resulting in temperature increase of the cold lead flowing down at the DHX outlet towards the core inlet. Enhanced temperature stratification is evidenced after 1 hour transient when quasi steady-state conditions are reached, and the core decay power is efficiently removed by natural circulation through the core and the DHR system with limited lead temperature increase.

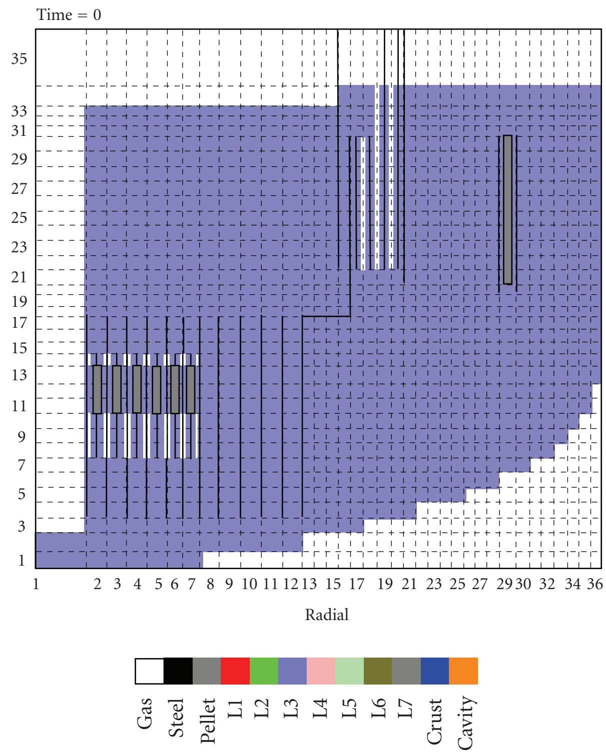


Figure 3: SIMMER-III nodalization scheme of the EFIT reactor.

At present, the SIMMER-III code is not validated for this kind of analysis. The results of lead-bismuth natural circulation experiments, which are foreseen in the integral CIRCE facility at ENEA/Brasimone research centre, could be used to confirm the capabilities of the code in this area and propose possible and beneficial code model improvements.

4. CALIBRATION OF THE RELAP5 MODEL ON SIMMER-III RESULTS

The RELAP5 code [4], developed by INEEL for the US-NRC, is a lumped parameter code used for best-estimate thermal-hydraulic (T/H) transient analysis in light water reactors. The RELAP5 code has been modified by ANSALDO (including lead properties) to be used for lead-cooled ADS analysis. Lead thermophysical property data and heat transfer correlations for heavy liquid metal in different geometries have been taken from the literature and available studies and introduced in the modified code [5]. Conservative assumptions have been considered for heat transfer in core bundle geometry, and new correlations have been developed for helical tube geometry of SGs [6]. The so modified RELAP5 code has been applied in previous ADS plant transient analysis such as PDS-XADS [7], and the results have been successfully compared with other codes. Furthermore, the code has been validated against experimental data from tests conducted on the CHEOPE and CIRCE facilities at ENEA/Brasimone Centre [8, 9]; a new experimental program to be conducted on the large-scale integral facility CIRCE will provide further valuable data for code validation and

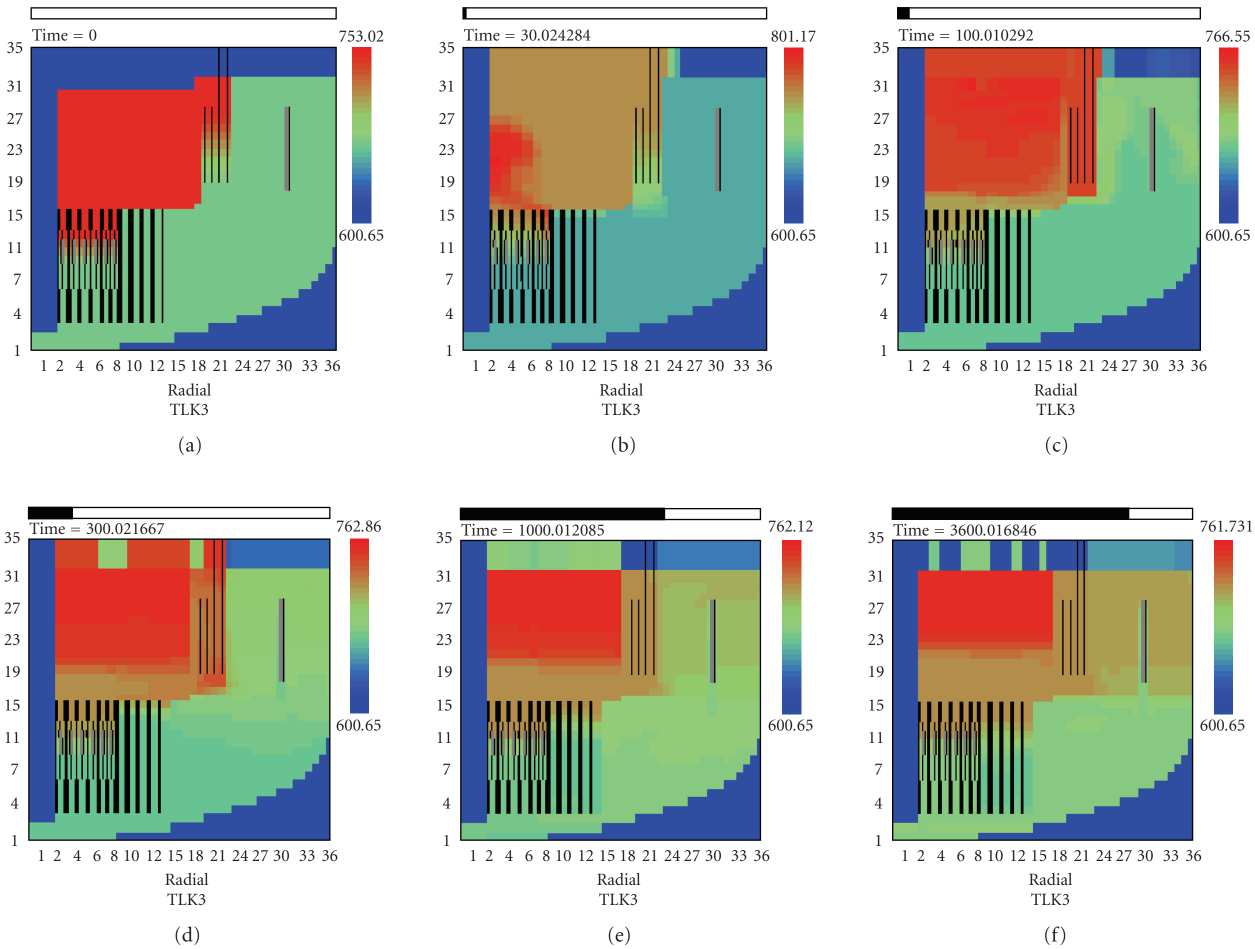


FIGURE 4: Time evolution of lead temperature (K) distribution in the primary vessel.

verification. This modified version of RELAP5 code is used in the framework of a collaboration between ENEA Centre and the Department of Nuclear Engineering (DIN) of the University of Palermo for T/H transient analysis of EFIT.

The RELAP5 nodalization scheme of the EFIT reactor employed for T/H transient analysis is shown in Figure 5. In the model, the lead mass inventory distribution and the major flow paths are represented. The core is discretized in three zones, each one represented with average and hot channel with equivalent flow area coupled with corresponding pin thermal structures. The reflector and bypass zone is represented by an equivalent hydraulic channel, while the target unit is not simulated. Primary pumps are modelled according to preliminary EFIT pump design. Steam generator primary (shell) and secondary side (straight and helical tubes) are modelled in details, while just boundary conditions are used for the secondary loop. The DHR system model is limited to the DHX primary side imposing the exchanged power as a function of the lead temperature at the DHX inlet.

In case of loss of primary pumps and LOHS scenario with operation of the DHR system, the natural circulation flow paths in the 1D RELAP5 representation of EFIT are arbitrarily defined in the input nodalization scheme. Therefore, mixing effects at steam generator and DHX outlets and inside plenum recirculation phenomena, observed with more detailed SIMMER-III analysis, are not taken into account. These phenomena may tend to reduce the lead mass flow rate in natural circulation through the core and the DHR system in accidental conditions, as demonstrated by comparing SIMMER-III and RELAP5 results under the same transient conditions. As expected, RELAP5 predicts larger lead mass flow rate than SIMMER-III through the core and the DHR system.

The RELAP5 model has been calibrated in order to reproduce as close as possible the SIMMER-III results. According to nomenclature in Figure 5 (green characters), fluid mixing in the volume below steam generator and DHX is defined by the fraction of lead mass flow rate ($x = 17\%$) entering this volume from the DHX outlet. This fraction is deduced from SIMMER-III results by mass and energy balances for the volume according to

$$y = \frac{m_C(\mathrm{T}_{Ci} - \mathrm{T}_{Do})}{(\mathrm{T}_{Di} - \mathrm{T}_{Do})}, \qquad x = y + m_D - m_C, \tag{1}$$

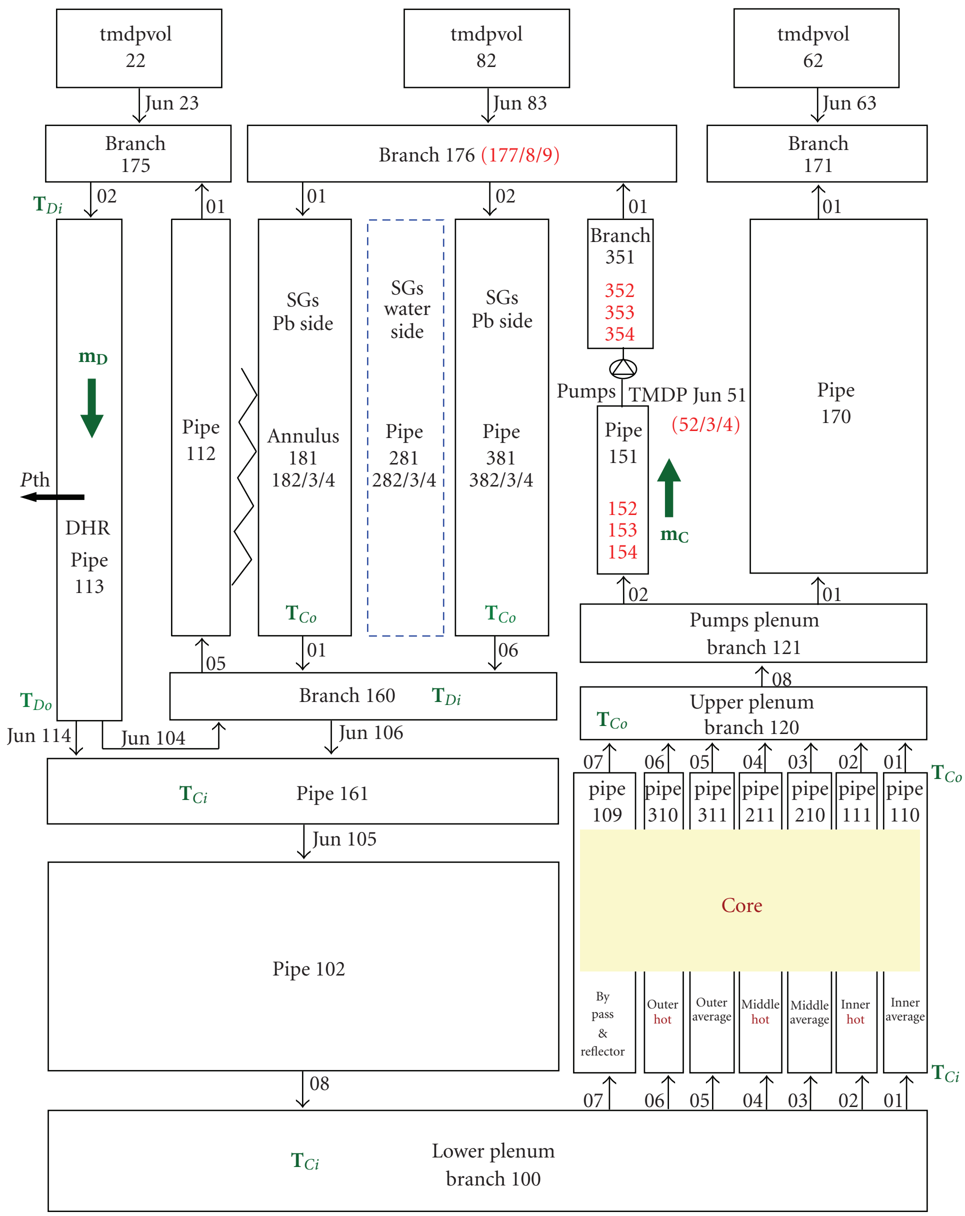


FIGURE 5: RELAP5 nodalization scheme of EFIT reactor.

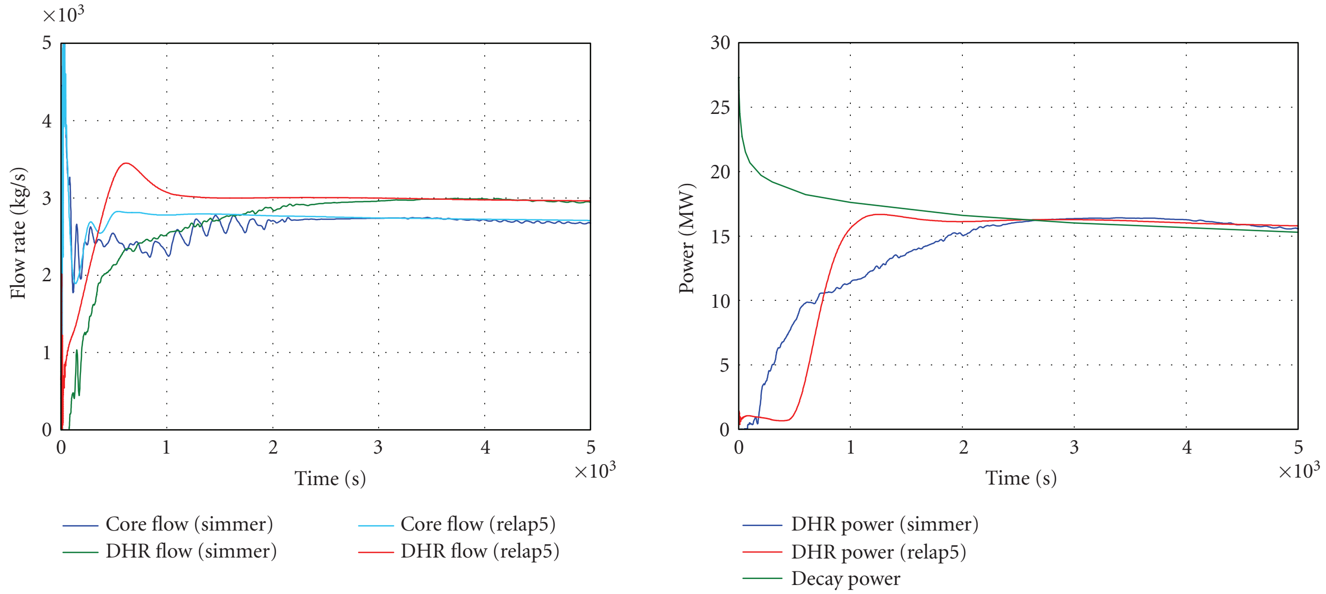
where C and D denote, respectively, core and DHX parameters (inlet/outlet temperatures and mass flow rate of lead).

Additional pressure drop coefficients have been implemented in the revised RELAP5 model of the primary system (at pump and DHX outlet locations) to reproduce the natural circulation mass flow rates calculated by the SIMMER-III code in the medium term.

The lead mass flow rate and decay heat removal in the DHR system calculated by the two codes are presented in Figure 6. Both codes predict efficient removal of decay power after about 2000 seconds. Mass flow rates through the core and the DHR heat exchangers are equivalent in the medium term. Code results differ in the initial transient owing to the different modelling.

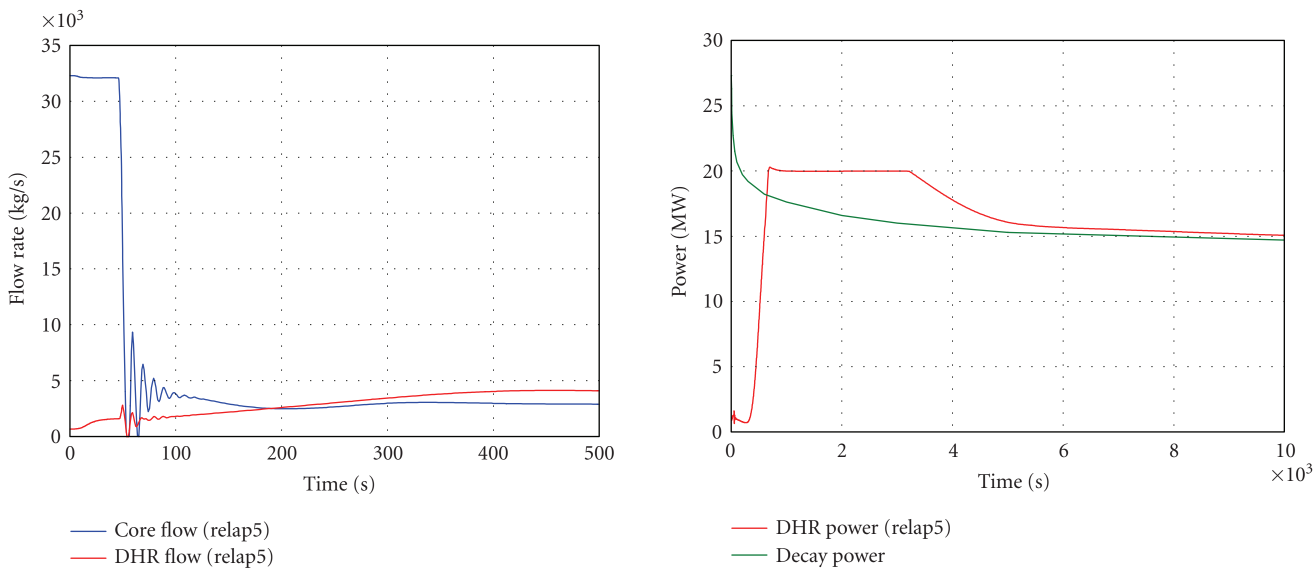
5. LOHS ACCIDENT ANALYSIS WITH RELAP5

The LOHS accidental transient has been analysed with the revised RELAP5 model. The initiating event is the total loss of feedwater of secondary loops. The following lead temperature increase in the primary circuit is supposed to lead to reactor trip on high core outlet temperature signal by the protection system. Reactor trip, consisting in the proton beam switch-off, is assumed with 1 second delay after the average core outlet temperature exceeds 773 K (20 K above the nominal outlet temperature). As a conservative assumption, primary pumps trip is assumed to occur at the same time as the reactor trip and no pump inertia is considered (pump design is not precisely defined yet), in



(a) (b)

FIGURE 6: Lead mass flow rates (core and DHR) and heat removal in the DHR system (3 units).



(a) (b)

FIGURE 7: Lead mass flow rates (core and DHR) and heat removal in the DHR system (3 units).

order to maximize core peak temperature just after reactor trip.

The results of the analysis are presented in Figures 7 and 8. Reactor trip is calculated by RELAP5 at 46 seconds. After some initial oscillations induced by free level movements (see Figure 7), the lead mass flow rates through the core and the DHR system become stable and the DHR attains maximum performance (20 MW for the 3 units supposed to be in operation) after about 700 seconds.

The peak clad temperature reaches 862 K in the hottest channel of outer core zone as shown in Figure 8 (the clad safety limit in normal operation of 823 K is exceeded for few seconds only), then all temperatures stabilize starting from 5000 seconds at acceptable values. The reactor vessel wall temperature reaches a maximum value of 724 K around 2200 s during the transient, then it stabilizes at 713 K in the medium term, below the maximum acceptable value of 723 K. This vessel temperature limit has been defined to assure the vessel integrity for the entire life of the plant. The vessel wall temperature peak of 724 K is not a critical issue, because its duration is limited in time. Furthermore, an improved DHX solution now implemented in the EFIT

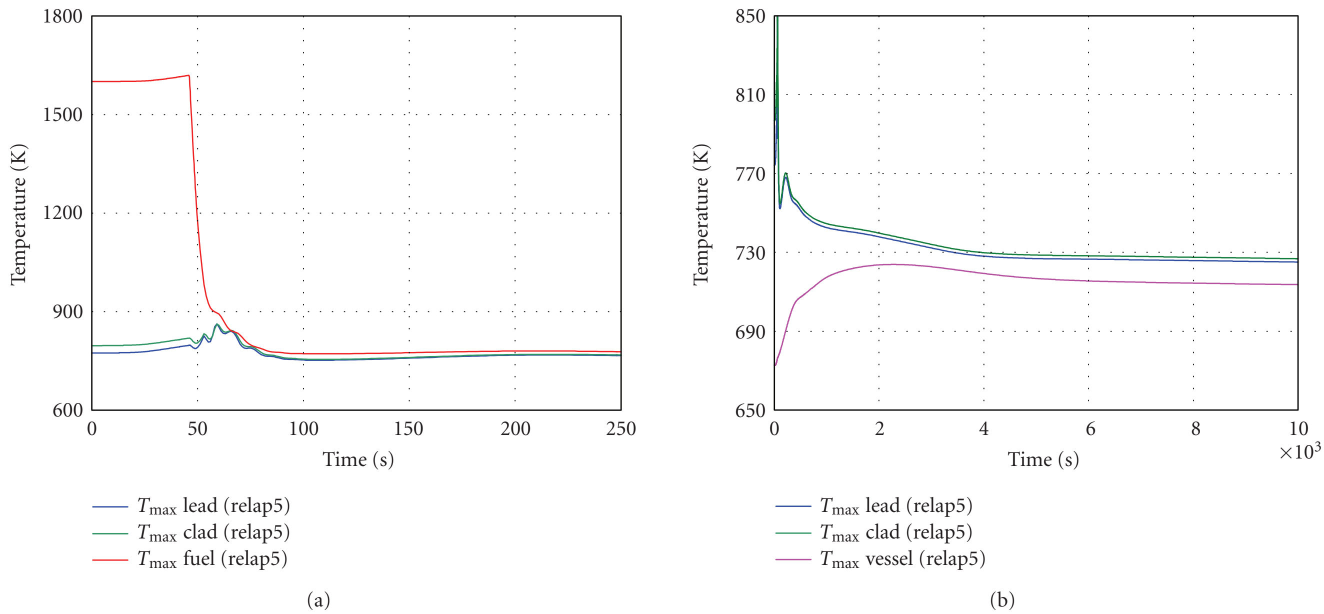


FIGURE 8: Maximum core (lead, clad, and fuel) and vessel wall temperatures.

design, with increased length of the external wall and reduced pressure drops, which facilitate the natural circulation of lead through the DHX, foresees an increase of about 15% of the lead mass flow rate, thus reducing significantly the thermal load on the vessel wall.

The flow path imposed in the RELAP5 model accelerates the startup of natural circulation through the DHX with respect to the SIMMER-III simulation (see comparison of mass flow rates calculated by the two codes in Figure 6); however, the integral power removed by the DHR system at 2200 seconds calculated by SIMMER-III is consistent with the RELAP5 value, therefore, no appreciable increase of the peak vessel wall temperature is exhibited at that time by SIMMER-III with respect to RELAP5 analysis.

6. CONCLUSIONS

The performances of the DHR system provided in the EFIT reactor have been confirmed by SIMMER-III and RELAP5 analyses of accidental conditions with complete loss of heat removal by the secondary system. In particular, natural circulation in the primary circuit through the core and the DHR system stabilises in less than one hour, and three out of four DHR units are sufficient to adequately remove the core decay power under LOHS transient conditions.

The 1D RELAP5 model of EFIT has been successfully calibrated on the basis of the SIMMER-III results by evaluating the amplitude of hot and cold lead mixing at steam generator and DHX outlets and the effects of turbulence phenomena and recirculating flows in the upper and lower plena of the reactor vessel. Except for the initial transient, the results obtained with the revised RELAP5 model are close to the SIMMER-III results. However, code validation for this kind of applications is still in progress, and limited for SIMMER-III code. Therefore, the results of the DHR system performance analysis will be precised after further validation of the codes at the CIRCE facility or other appropriate facilities.

Finally, the application of RELAP5 to the LOHS accident analysis has shown that the DHR system is able to face up accidental situations with sudden total loss of heat removal by the secondary side with limited increase in core temperature and brings the plant in safe conditions in the medium term. Also the vessel wall temperature increase is limited below acceptable value in the medium term with adequate margin, taking into account more recent DHX design improvements not addressed in this study.

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