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Analyses for MARIA Research Reactor with RELAP/MOD3 Code

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ABSTRACT

This paper deals with the application of the RELAP5/MOD3 code to the transient analyses for MARIA research reactor. Poland's MARIA Research Reactor is water and beryllium moderated, water-cooled reactor of a pool type with pressurized fuel channels containing concentric multi-tube assemblies of highly enriched uranium clad in aluminium.

The RELAP5/MOD3 input data model includes the whole primary cooling circuit of the MARIA reactor. The model was qualified against the reactor data at steady state conditions and additionally against the existing reliable experimental data for a transient initiated by the reactor scram. The RELAP transient simulation was performed for loss of forced flow accidents including two scenarios with protected and unprotected (no scram) reactor core. Calculations allow estimating time margin for reactor scram initiation and reactivity feedbacks contribution to the results.

1 INTRODUCTION

Poland’s MARIA Research Reactor is a high-flux multi purpose reactor which is water-cooled and moderated with both water and beryllium. Standard U-Al alloy HEU (80%) fuel assemblies (FAs) are of the M6-type which consists of six concentric circular fuel tubes clad in aluminium. Water flows downward in the three outer coolant channels and returns upward in the four inner coolant channels. The fuel assemblies are located within beryllium matrix on square. Reactor power depends on the core configuration, but is typically of the order of 20 MW.

The compact fuel assembly structure and special configuration of the cooling channels make particular demands on the description of the thermohydraulic behaviour and the technical safety measures for the MARIA reactor. In the case of accidents in such a reactor, the transients, opposite to the case of the power reactor, occur in the range of seconds. This is due, not least, to the high heat flow densities of about 1.5 MW/m^2 and to the system pressure, temperatures and flow rates that are quite low in comparison to power reactors.

Extensive research work has been performed world wide for several years to develop and verify large thermohydraulic system codes such as RELAP and ATHLET for analyzing the thermohydraulics in light water reactor system during various transients and accident conditions. Owing to the specific features of the research reactor mentioned above it is not possible to apply these codes, which were originally developed for power reactors, in safety analyses of the research reactors without considerable effort related to the preparation of reliable input data model, verification or even modification of the code [1].

The first approach to the RELAP5 application for MARIA reactor was done within the ERTR (ANL) Program in 1999 [2]. The main effort of ANL analysis concerned determination of the neutronic safety parameters for the reference core configuration using several advanced physical codes. These parameters were applied to define the point neutron kinetic model of the MARIA core for RELAP5/MOD3 code calculation of the reactivity insertion transients.

The present paper was motivated by the need to adequately develop the RELAP5 input deck model for the purpose of MARIA reactor analysis under both loss of flow and reactivity insertion transients.

2 RELAP5 INPUT MODEL OF MARIA REACTOR

A RELAP5/MOD3 input model of the MARIA research reactor core developed in IEA reflects a real status of a reference core configuration consisting of 16 M6-type fuel assemblies with 80% ^{235}U enrichment as in Ref. [2]. The model used in RELAP5 includes one fuel assembly (FA) to represent the aggregate of 15 average FAs and one FA to represent the peak power assembly.

Nodalization scheme is shown in Fig. 1 for the average and peak power fuel assembly. It can be seen that the identical nodalization approach was applied to both fuel assemblies representing the reactor core. The model includes 304 hydrodynamic nodes, 302 junctions and 272 heat structures with 2496 mesh points.

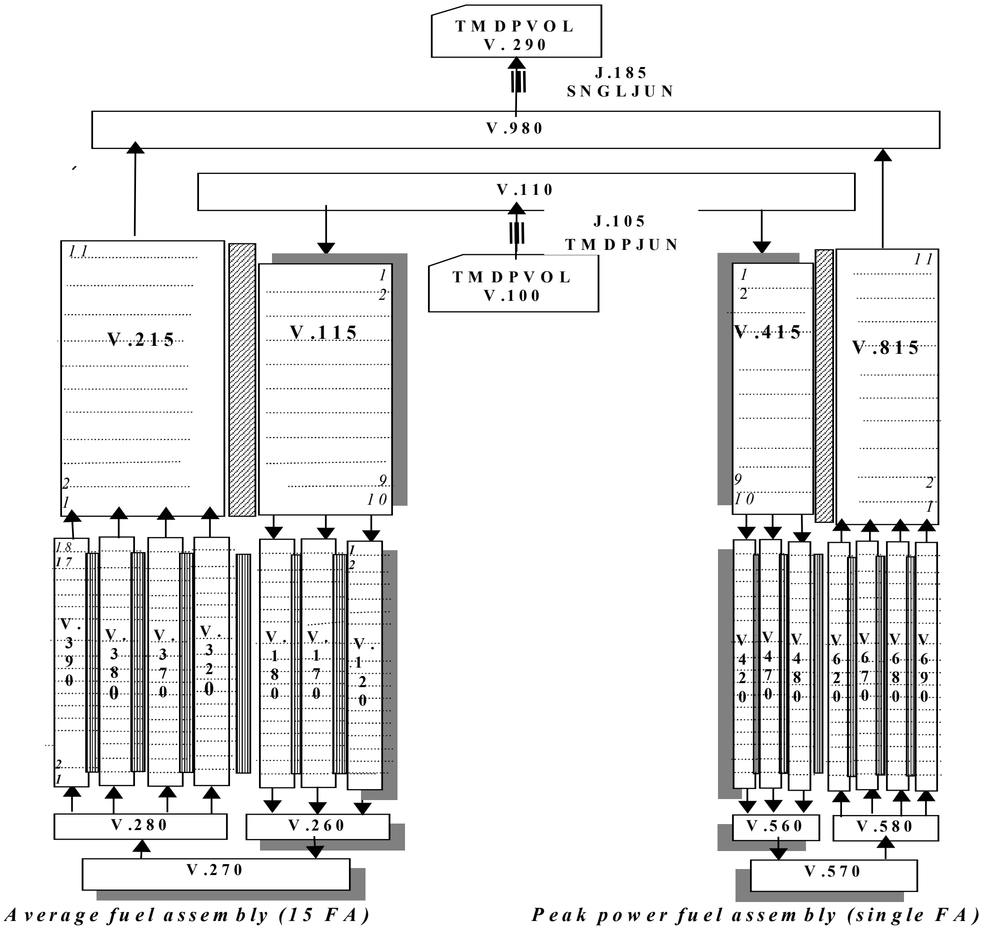

Fig.1 MARIA core nodalization for RELAP calculations

The lower plenum of the FA is subdivided into three layers, providing a quasi two-dimensional representation. This model allows reflecting the existence of the flow direction change and it should be appropriate for prediction of natural circulation conditions within FA during transient simulation.

No particular techniques were used to define pressure loss coefficients in the junctions. Later, during input stabilization, the coefficients were adjusted in order to obtain the proper pressure drop.

The point reactor kinetic approximation available in RELAP5 code was used to compute the power of the MARIA reactor. Once the total core power has been determined, it is then distributed among the fuel heat structures in an invariant manner.

The immediate fission power plus decay product power option was used for calculating the total power. The ANS 79-1 standard fission product data were used with a fission product yield factor of 1.2 to provide a conservative margin.

Based on the results of the three-dimensional diffusion calculations for the MARIA reactor 16 fuel assembly core, reported in [2], an adequate format of the reactivity feedback data was defined in the RELAP5 input deck. Weighting factors are input to specify the reactivity contribution of each hydrodynamic volume and heat structure to the total.

For the MARIA reactor, all the reactivity feedback coefficients are negative except for the beryllium temperature feedback coefficient, which is positive [3]. Unfortunately, beryllium influence on the total reactor reactivity could not been considered in the RELAP calculations since in-core and ex-core beryllium matrix immersed in the pool water was not explicitly represented in the input deck model.

Power distributions needed for transient analyses were determined for case with the safety rods withdrawn and with the bottom of the control rod absorbers located at the core midplane. The limiting power value for MARIA assembly equal to 1.8 MW [4] was assumed in the peak power assembly model.

3 QUALIFICATION OF INPUT DATA

A number of calculations were performed for qualification of the input deck against the reactor data at steady state conditions as well as to simulate transient data for which reliable data exists.

3.1 Steady State Calculations

Basic parameters of the system under steady state conditions are summarized below.

Table 1: Nominal value and output results for the steady state analyses

Nr.	Input Parameter	Units	Nominal Value	Output Value
1.	Steady state reactor power	MW	17.0	17.0336
2.	Coolant flow rate through core	kg/sec	145.1	145.1
3.	Inlet pressure (V.110 at Fig. 1)	MPa	1.400	1.406
4.	Inlet core coolant temperature	^{0}C	54.0	54.0
5.	Temperature increase in core	^{0}C	26.2	26.16
6.	Peak-to-average power density	-	2.815	2.816
7.	Total power of peak power FA	MW	1.8	1.8080
8.	Peak power FA flow rate	kg/sec	9.0687	9.0687
9.	Pressure drop in FA	MPa	0.54	0.539676

The main effort during steady state calculation was concentrated on obtaining pressure loss characteristics that correspond to the available experimental data for fuel assemblies [5]. A final correction of the pressure distribution along the particular part of FA was done by reasonable adjusting of pressure loss coefficients in the junctions. The resulting values of pressure drop from RELAP5 steady state calculations (see Fig. 2) are quite acceptable. The first 200 sec of calculation showed that there were no significant difficulties to reach steady state conditions.

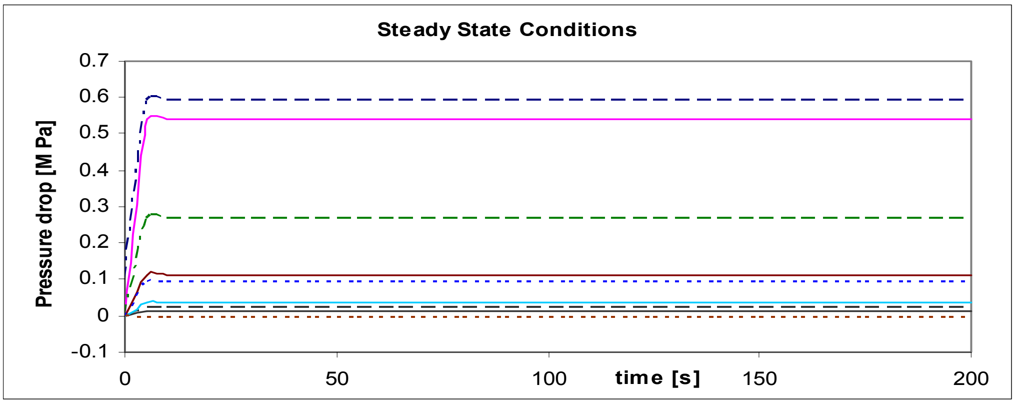

Fig. 2 Local pressure drops within fuel assembly

It can be seen from Fig. 3 that the calculated heat loss to the reactor pool does not exceed 7% of the total reactor power.

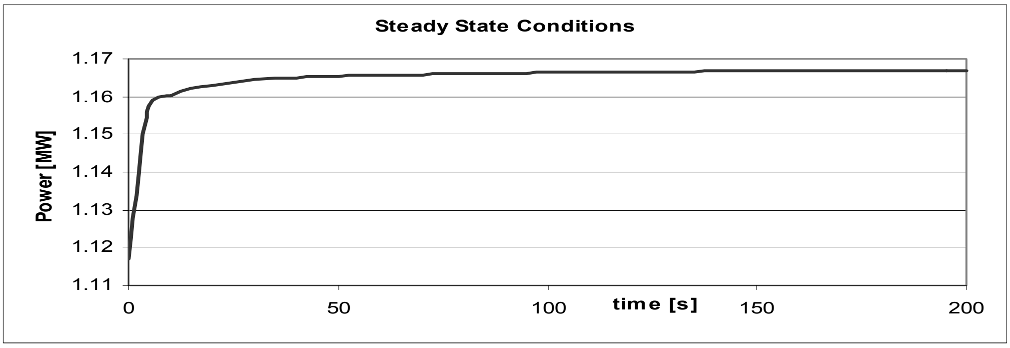

Fig. 3 Heat loss to the reactor pool

3.2 Qualification against Experimental Data

The experiment with reactor scram and without pump costdown [5] has been used to check the input deck. Evaluation was limited to the time trends of coolant temperatures in two points of the peak power fuel assembly, namely at the outlet (node1 of V. 815 in Fig. 1) and at the lower plenum (V.570 in Fig. 1). Comparisons in Fig. 4 showed that the agreement with the experimental data is quite good.

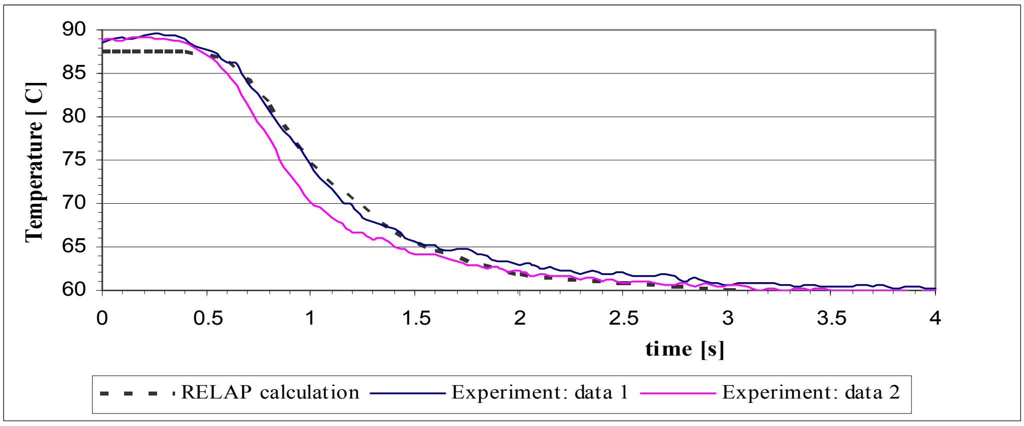

Fig. 4 Coolant temperature in lower plenum of the fuel assembly

4 TRANSIENT SIMULATION

4.1 Loss of Flow Accident. Basic Case

The initiating event of the accident is the simultaneous tripping of all reactor coolant pumps due to loss of electrical power supply.

In the analysis, the reactor scram signal was assumed to appear when reactor coolant mass flow rate decreases below 70% of the initial value. Time delay of the reactor protection system equal to 0.1 sec was applied. Reactivity inserted into the core during scram was determined assuming the constant acceleration of the safety rods (2.7 m/sec^2). Data related to the pump characteristics were prepared based on Ref.5.

The neutronic power trend can be explained considering two time periods (see Fig. 5). Reactor scram signal appears at 1.3 sec into the transient. Up to 1.4 sec: the fission power essentially depends upon the neutronic-thermal hydraulic feedback. It continues to decrease and at 1.4 sec becomes about 94% of the nominal power value owing to negative reactivity feedback effects with increase of both fuel and coolant temperature. After 1.4 sec: the scram forces calculated power to decay values.

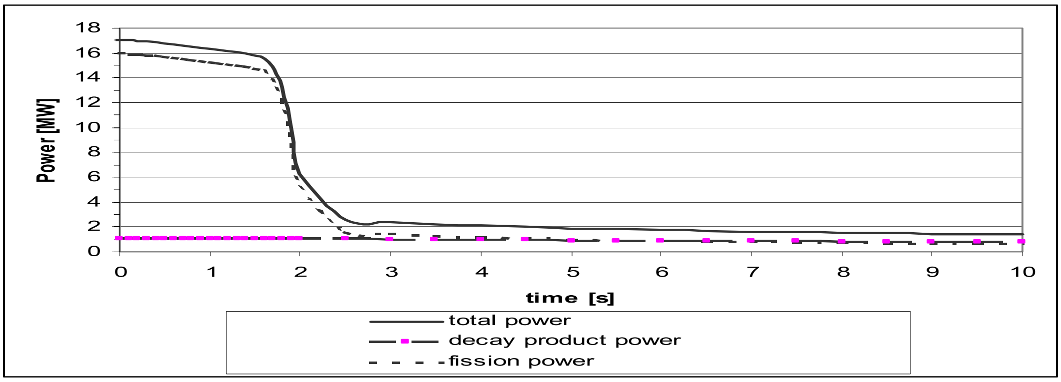

Fig. 5 Total core power, fission power and decay product power

The close relation between fuel/clad temperature trends and heat power balance within fuel elements can be seen easily from Figs. 6 and 7. The temperatures continue to increase from the beginning of the transient up to 1.69 sec, and then the heat generation rate within fuel element exceeds the heat power transferred to the coolant. Exactly at 1.69 sec, the cladding and fuel maximum temperatures are reached and just after this the heat transfer power becomes significantly higher than the heat generation rate, which results in a rapid decrease of fuel/cladding temperatures.

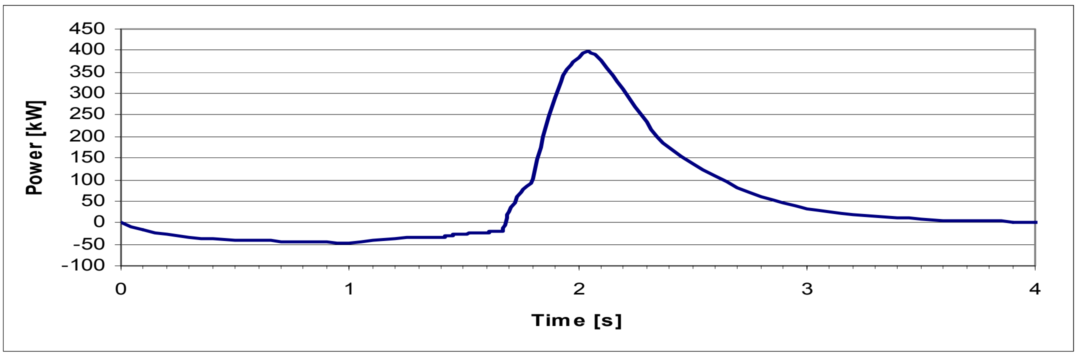

Fig. 6 Increase of heat transfer power above source power at the peak FA

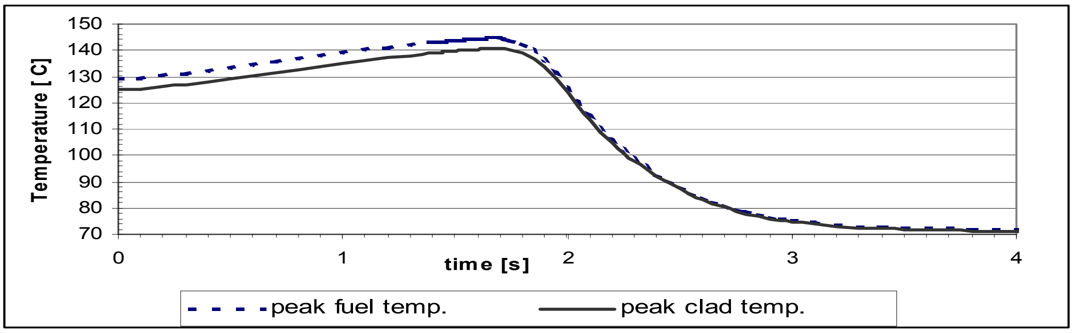

Fig. 7 Cladding and fuel maximum temperature

Additionally, the conservative case without contribution of the reactivity feedback was recalculated to estimate the influence of such simplification on the prediction of the maximum cladding temperature.

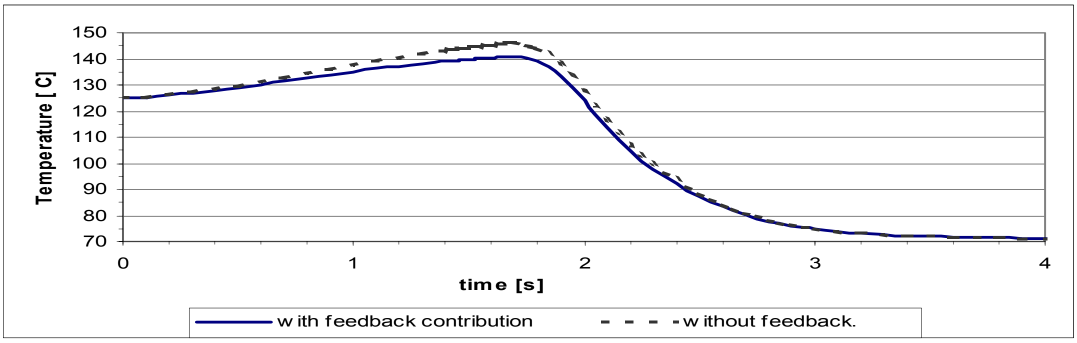

Fig. 8 Comparison of peak cladding temperature

Fig. 8 shows that overestimation of the maximum cladding temperature due to neglecting of reactivity feedback contribution is about 5 ^{0}C. Even under the pessimistic assumptions the calculated maximum temperature of the surface cladding is still evidently below saturation temperature associated with the system pressure.

4.2 Loss of Flow Accident for Unprotected Core

The hypothetical scenario of loss of flow accident with unprotected core (no scram) was chosen to study the time margin limit for scram activation. Two cases with and without reactivity feedback contributions were recalculated. According to the calculation with reactivity feedback the cladding temperature starts to increase rapidly at about 16 s after the pump trip while in the case of transient simulation without reactivity contribution the cladding temperature excursion appears 10 s earlier (see Fig. 9).

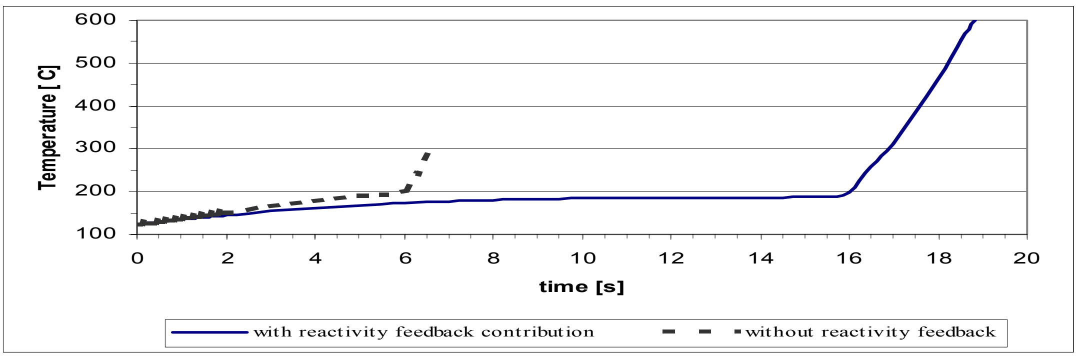

Fig. 9 Unprotected core: maximum cladding temperature

At the moment of cladding temperature excursion the power is well below 50% of the initial power (Fig. 10). Such a rate of power decrease is determined by negative reactivity feedback coefficients for coolant temperature, the coolant void, and the fuel Doppler coefficient.

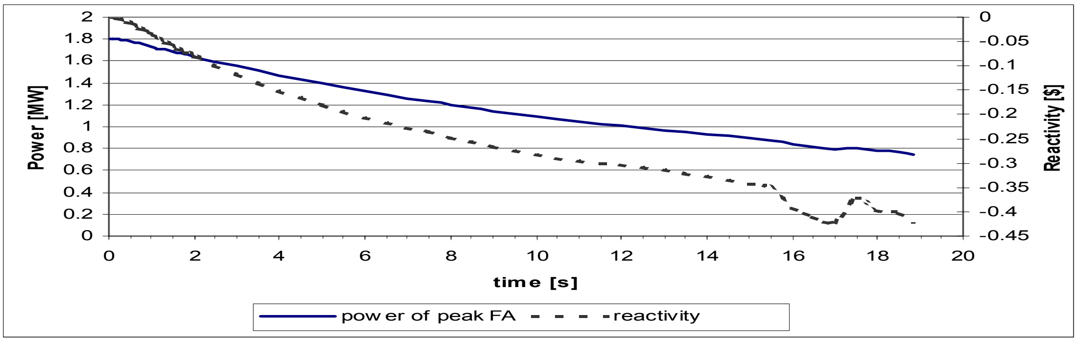

Fig. 10 Unprotected core: Power of the peak FA and total core reactivity

Due to impairment of forced cooling by failure of the coolant pumps, flow excursion arise in the narrow cooling channels (Fig. 11) owing to steam formation, which results in critical heat flux load being exceeded within a few seconds (Fig. 12). The flow instability is caused by the fact that although the coolant is subcooled, steam is formed in the thermally highly stressed channels and this steam requires a high volume fraction owing to the low pressure. If the steam present in the channel exceeds a critical value, then the associated additional acceleration and friction pressure losses lead to such an unstable situation that the pressure loss will increase with decreasing flow rate. However, since the pressure drop over the channel is constant owing to the parallel arrangement of the remaining channels, this leads to a further drop in the flow rate in the affected channel

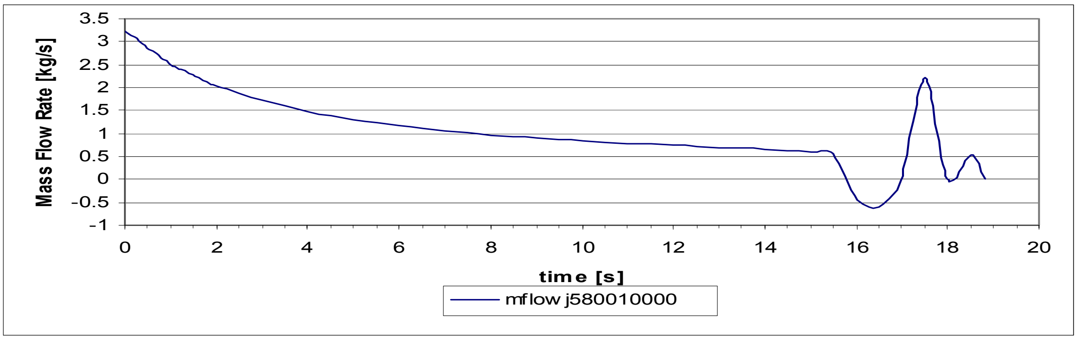

Fig. 11 Unprotected core: Inlet mass flow rate to hot channel (V. 620) of the peak FA

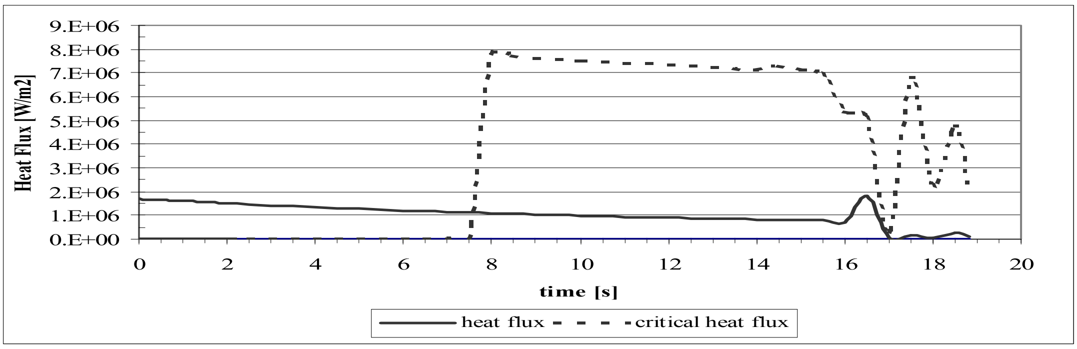

Fig. 12 Unprotected core: Heat flux and critical heat flux at hot place

5 CONCLUSION

The RELAP5/MOD3 input data model of the MARIA research reactor has been developed to provide the capability for the analysis of the reactor under loss of flow and reactivity insertion transients. General concept was to create a best estimate model, generic for future applications. The developed input deck, which models a reference core status as in Ref. [2], can be easily modified according to the investigated core configuration as well as boundary and initial conditions.

The model was qualified against the reactor data at steady state conditions and additionally against the existing reliable experimental data for a transient initiated by reactor scram. The results obtained with the code agree well with the experimental data. With the elaborated nodalization of fuel assemblies, the main effort during steady state calculation was concentrated on obtaining pressure loss characteristics that correspond to the available experimental data.

The RELAP transient simulations of loss of forced flow accidents were performed including two scenarios with protected and unprotected (no scram) reactor core. Calculations allow estimating time margin for reactor scram initiation. Great sensitivity of the results to reactivity feedback was found.

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