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RELAP5/MOD3 model and transient analyses for the MARIA research reactor in Poland

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Abstract The RELAP5/MOD3 input data model of the MARIA research reactor has been developed to provide the capability for the analysis of the reactor core under loss of flow and reactivity insertion transients. 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 results obtained with the code agree well with the experimental data. 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. The presented input data model should be treated as the first step for developing of the model including the whole primary cooling circuit of the reactor.

Key words research reactor • transient analysis • neutronic parameters • qualification

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Introduction

The MARIA Research Reactor in Poland is a water and beryllium moderated, water-cooled reactor of pool type with pressurized fuel channels containing concentric multi-tube assemblies of highly enriched uranium clad in aluminium. The MARIA reactor currently uses 6-tube fuel assemblies in which water flows downward in three outer coolant channels and returns upward in four inner coolant channels.

The high power densities in the MARIA reactor core are achievable due to a fuel assembly concept in which the active parts of the multi-tube fuel elements are divided by narrow annular cooling channels. The heat generated by fission in the fuel tubes is released via short heat conduction paths to the cooling water. With such arrangement of the fuel elements the driving differential pressure for water flowing through all annular cooling gaps is practically identical; it depends on the cooling water requirements of the most heated channel. The maximum power of the fuel channel is selected in such a way that there is a large subcooling of water at the channel outlet relative to the minimum possible coolant pressure.

The compact fuel assembly structure and special configuration of the cooling channels make particular demands on the description of the thermohydraulic behavior 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 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 [3].

This paper deals with the application of the advanced thermal hydraulic code RELAP5/MOD3, developed at AT&G Idaho Falls [7], to the MARIA research reactor safety analyses.

The first approach to the RELAP5 application for the MARIA reactor was done within the ERTR (ANL) Program in 1999 [1]. The main effort of ANL analysis concerned determination of the neutronic safety parameters (reactor kinetic parameters, reactivity feedback coefficients, and control rod reactivity worth) 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. The input model was used to calculate the reactor response to fast and slow reactivity insertions with both low and high enriched uranium fuels.

The present paper was motivated by the need to adequately develop the RELAP5 input deck model for the purpose of MARIA reactor core analysis under both loss of flow and reactivity insertion transients. The input model was established for MARIA reactor core parameters (geometry, thermal-hydraulic parameters, protection system set points, etc.) taking into account the reference initial and boundary conditions. The reference core configuration and the set of the neutronic safety parameters for the fresh 80% enriched fuel were assumed according to Ref. [1]. Before performing accident analyses, priority was given to qualification of the developed input model against the reactor data at steady state conditions and, additionally, against reliable experimental data for a transient [4].

RELAP5 input model of MARIA reactor

A RELAP5/MOD3 input model of the MARIA research reactor core developed in the 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. [1]. The fuel assemblies are located within a beryllium matrix on a square grid. The MARIA reactor, fuel assemblies, and operational characteristics are described in Refs. [2] and [5].

The input deck was developed with emphasis on well-arranged structure providing sufficient space for remodeling of the core configuration and including extension to the primary cooling circuit representation. Presented status of the RELAP5 input deck allows analysis of the MARIA research reactor under loss of flow and reactivity insertion transients.

The MARIA reactor 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. The attempt is a compromise between efforts to determine specific conditions of a particular FA behavior and a reasonable size of the input deck. All the coolant channels and fuel tubes in the FA are explicitly represented. Suction and delivery collectors of the primary cooling circuit are also included in the model. A variable number of hydraulic nodes is introduced to simulate the boundary conditions required in each transient calculation. Conductive heat losses are considered either to the pool or between the modeled volumes, specifically across the wall along the inlet and outlet part of pressurized channels.

Thermal-hydraulic model

The model was developed considering the guidelines of the code User Manual [7] as well as the previous experience gained during participation in several IAEA and OECD projects related to the code assessment.

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. The basic nonhomogeneous nonequilibrium two-phase flow model is applied uniformly to all hydrodynamic nodes and junctions.

Nodalization of the fuel assemblies reflects specific features of FA layout including concentric multi-tube channels and the downward and upward coolant flow within the FA. The fine mesh model was used in the nodalization of the active part of fuel assembly to reproduce its topological complexity and to represent correctly the expected thermal hydraulic phenomena. A relatively coarse nodalization was selected for long inlet and outlet parts of the pressurized channels above six-tube fuel assemblies.

In the axial direction, each of the seven parallel fuel channels within FA is divided into 18 nodes. The first and the last axial nodes represent the non-heated part of the core. The other 16 nodes belong to the heated part of the core. The two top heated volumes have a reduced length for better simulation of scenarios with partial core uncovering. The lower plenum of the FA is subdivided into three layers, providing a quasi two-dimensional representation. This model allows to reflect the existence of the flow direction change and it should be appropriate for prediction of natural circulation conditions within FA during transient simulation.

The roughness of the surface was defined in the input so that the pressure drop distribution along the channels could be calculated by the code. The abrupt area change option available in the RELAP code was selected to calculate additional losses resulting from abrupt expansions, abrupt contractions and orifices. No particular techniques were used to define pressure loss coefficients in the junctions. Nevertheless, some reasonably chosen pressure loss coefficients were introduced when additional losses were needed besides the wall friction and abrupt area change losses, specifically for both the inlet and outlet nozzles of pressurized channels and for regions with flow

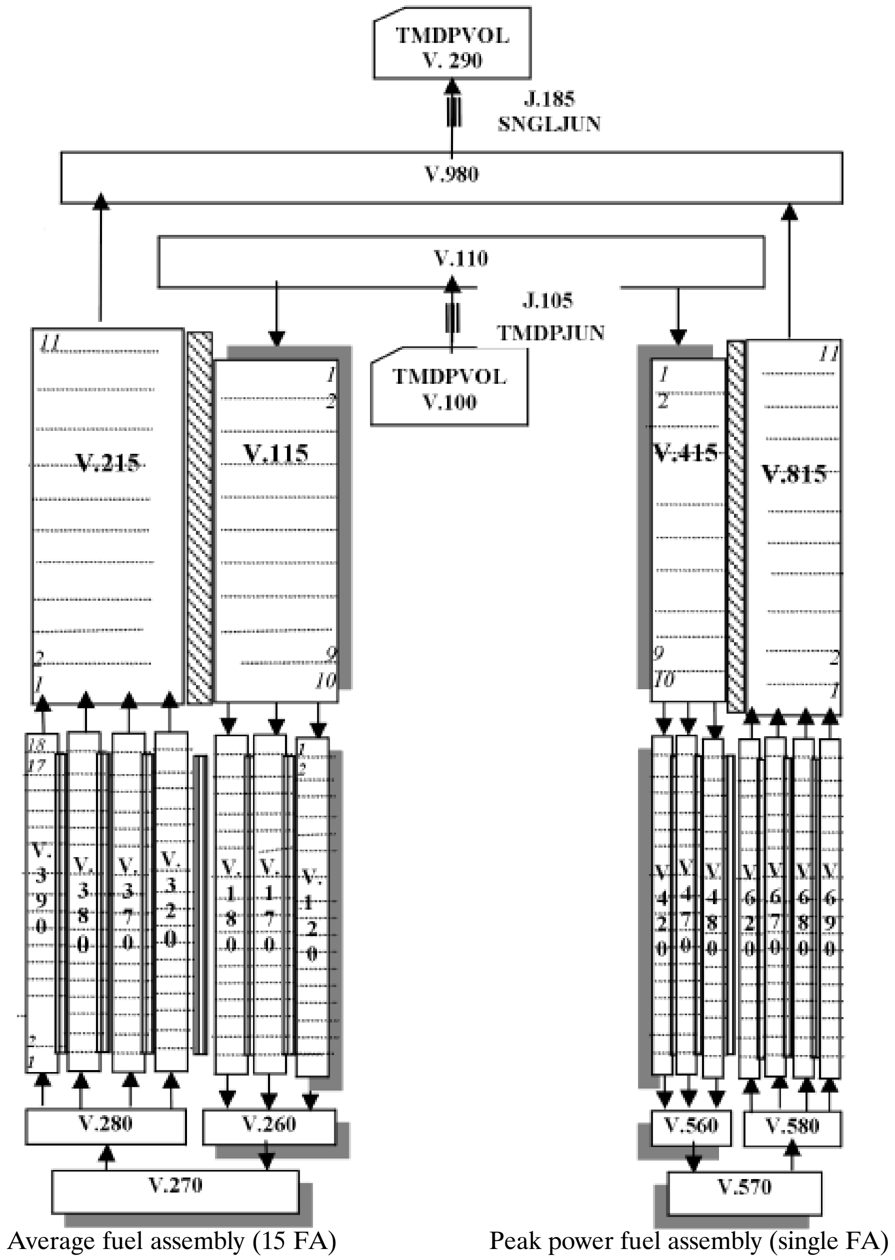

Fig. 1. MARIA core nodalization for RELAP calculations.

direction change. Later, during input stabilization, the coefficients were adjusted in order to obtain the proper pressure drop.

Concerning conduction heat transfer, three types of heat structure composition of cylindrical geometry are considered in the input model. They represent nuclear fuel tubes, walls of pressurized channels connected with the pool water and inner structures of pressurized channels along the inlet/outlet channels above the fuel assembly.

In the axial direction, each of the six fuel tubes within the active core part of FA is divided into 16 heat structures. Temperatures are computed in 18 mesh-points along the radial coordinate within each heat structures. In the input deck, the total power determined from the reactor kinetics calculation was distributed among the fuel heat structures using the weighing factors such that the sum of weights for all heat structures is one. Then the source power in the particular heat structure is obtained by multiplying the total power by the weighting factor. A standard set of heat transfer correlations available in RELAP5/MOD3 code was applied to represent boundary conditions between the heat structures of the core and hydrodynamic nodes. The correlations cover various modes of heat transfer [7]. The heat transfer may take place under natural or forced convection conditions, with provisions for nucleate, transition, or stable film boiling.

A coarser discretization was applied for the heat structures representing walls of pressurized channels (4 mesh-point) and for the inner structures along the inlet/outlet part of pressurized channels (7 mesh-points).

The time dependent junction in addition to a time dependent volume (J.105 and V.100 in Fig. 1) specify inlet flow boundary of the model. This type of boundary conditions is analogous to a positive displacement pump where the inflow rate is independent of the system pressure. Pressure boundary conditions were modeled using a time dependent volume V.290 in which the pressure and thermodynamic state variables at suction collectors of the primary cooling circuit were specified.

Reactor kinetics

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 power model was defined by setting up a certain number of physical parameters for the reference core configuration with 16 fuel assemblies of fresh 80%-enriched fuel [1]. For the transient analyses given in this paper, i.e. the decay heat characteristics, reactivity feedback coefficients, control and safety rod reactivity worth, and power distribution characteristics were applied according to Ref. [1].

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. Effective delayed fission neutron fractions and decay constants for six delay groups were taken according to ENDF/B-VI data provided by the MC code in ANL and reported in [1].

Of the three reactivity feedback options provided in the RELAP5 code, the SEPARABLE option was used in the calculations. Reactor kinetic feedback due to coolant void/density, coolant temperature, and fuel temperature (Doppler effect) was assumed to be separable. With this option, a change in one of the three parameters does not affect the others.

Based on the results of the three-dimensional diffusion calculations for the MARIA reactor 16 fuel assembly core, reported in [1], an adequate format of the reactivity feedback data was defined in the RELAP5 input deck. Reactivity data for feedback from coolant void/density, coolant temperature, and fuel Doppler as well as volume and heat structure weighting factors were prepared for the core with fresh HEU fuel and for the case when the control rods were fully withdrawn. Weighting factors are the 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 [1]. Unfortunately, beryllium influence on the total reactor reactivity could not be considered in the RELAP calculations since the in-core and ex-core beryllium matrix immersed in the pool water was not explicitly represented in the input deck model.

In the RELAP5 input deck, the scram curves are obtained from a general table defining reactivity as a function of time. The table has an associated trip number. If the trip is true, the search argument is the current time minus the time at which the trip last turned true. The same type of tables was used to describe reactivity changes due to control rod motion for simulation of reactivity insertion transients. The same reactivity worth characteristics were used in scram modeling throughout all the cases.

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 [1]. The limiting power value for the MARIA assembly [5] equal to 1.8 MW was assumed in the peak power assembly model.

It should be noted that, depending on the core status and analyzed case, the user should prepare adequate neutronic safety parameters to apply in the RELAP5 input deck.

Input data deck overview

The RELAP5/MOD2 input for the MARIA research reactor was developed according to general requirements provided in the RELAP5 code manuals and satisfies the final recommendation of the International Code Assessment Program initiated by the US NRC. In the preparation of the input data deck, special attention was given to the flexibility of the input. The logical structure of the data deck and the numbering system provide sufficient space for remodeling or extensions (e.g. extension to the total primary loop representation).

The input deck is composed of the following groups of data: the job control cards, trip system data, hydrodynamic component data, heat structure data and control variables. A brief description of these data groups is provided below.

Job control data includes miscellaneous control, time step control and printed output organization data. The normal 'TRANSNT' run option is used for both the steady state and the transient simulations. The beginning of the transient simulation is assumed at 1000 s. At least 100 s is required for the pre-transient stabilization of the input model.

Time step control cards, inserted at the beginning of the basic input deck, should be treated as an example. For each individual job, the user should define such parameters as the minimum-maximum time steps, time step control option, frequencies of restarts and major-minor edits separately.

Trip system data is defined in terms of system variables or control variables through arithmetic and logic operations. Logical variables associated with each trip are applied in order to activate various model components.

Hydrodynamic component data is supplied for all hydrodynamic components including geometric data, relative elevation and junction orientation data, form loss coefficients, surface roughness data, volume related initial thermodynamic data, and junction velocity data. In the basic input deck, the hydrodynamic model includes 304 control volumes (nodes) connected by 302 junctions.

Heat structure data consists of the geometric data, thermophysical data, and heat source data for solid portions of the thermal-hydrodynamic system. 272 heat structures are thermally connected to the hydrodynamic control volumes. For nuclear fuel tubes, the axial power and pellet radial power distributions are provided according to Ref. [1]. In the basic input deck, steady state nominal conditions are assumed for the definition of the core power. Required thermophysical quantities include the thermal conductivity and heat capacity (both as a function of temperature) for each material composition included in heat structures.

All control variables defined in the input deck are divided into several groups that can be recognized through

Table 1. Input parameters and output results for the steady state analyses.

No.	Input parameter	Unit	Nominal value	Output value
1	Steady state reactor power	MW	17.0	17.0336
2	Coolant flow rate through core	kg/s	145.1	145.1
3	Inlet pressure (V.110 in Fig. 1)	MPa	1.400	1.406
4	Inlet core coolant temperature	°C	54.0	54.0
5	Temperature increase in core	°C	–	26.16[1)]
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/s	9.0687	9.0687
9	Pressure drop in FA	MPa	0.54[2)]	0.539676

[1)] Heat to the pool across FA walls taken into account. [2)] The data taken from [2].

the different range of numbers assigned to each of them. These groups include: pressure differences and gradients, temperatures (maximum, average, difference), heat balances (sources, heat transfer, heat losses), coolant mass or volume, collapsed and mixture levels.

It should be noted that all RELAP calculations required certain changes to the basic input deck. It is recommended that all accident specific modifications should be specified at the end of the basic input deck. Such an approach allows for quick access to and verification of the modified parts of the model.

Qualification of input data

Before performing accident analysis for the MARIA reactor it is highly recommended to run a number of calculations for qualification of the input deck against the reactor data under steady state conditions as well as to simulate transient data for which reliable data exists.

Steady state calculation

Basic parameters of the system under steady state conditions are summarized in Table 1.

The steady state calculation was done without using special RELAP5 mode "STDY-ST" with reduced thermal inertia of the system. The problem option "TRANST" was applied. Stabilization time was 200 s.

The curves of the main parameters during steady state calculations are shown in Figs. 2 up to 6. The first 200 s of calculation showed that there were no significant difficulties to reach steady state conditions.

The main effort during steady state calculation was concentrated on obtaining pressure loss characteristics that correspond to the available experimental data for fuel assemblies [2]. 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 (forward and reverse). The resulting values of pressure drop from RELAP5 steady state calculations (see Fig. 5) are quite acceptable.

Heat loss to the reactor pool was calculated on the assumption of constant pool water temperature (45°C) and the value of the heat transfer coefficient estimated as 8100 W/m^2 K. It can be seen from Fig. 6 that the loss flow does not exceed 7% of the total reactor power.

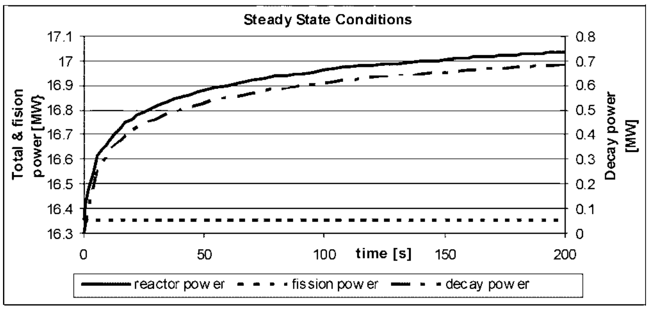

Fig. 2. Total power, fission power and decay power of the core.

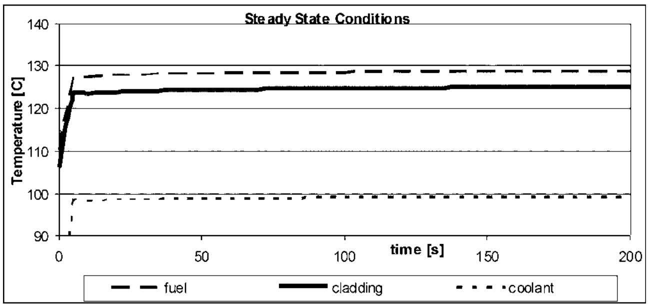

Fig. 3. Peak temperatures of fuel, cladding and coolant.

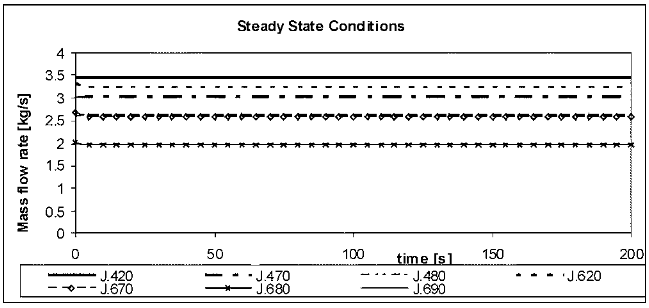

Fig. 4. Flow rate distribution through the peak FA channels.

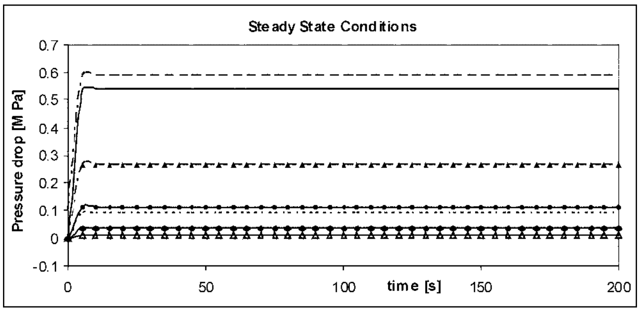

Fig. 5. Local pressure drops within fuel assembly.

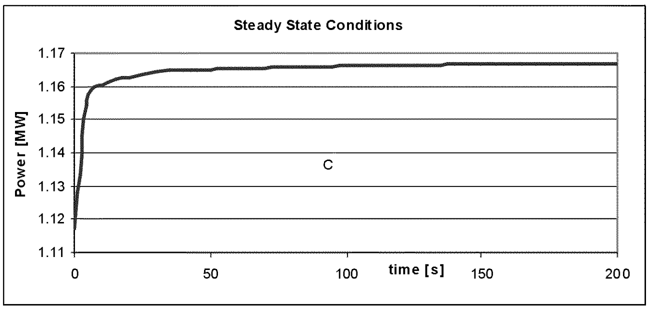

Fig. 6. Heat loss to the reactor pool.

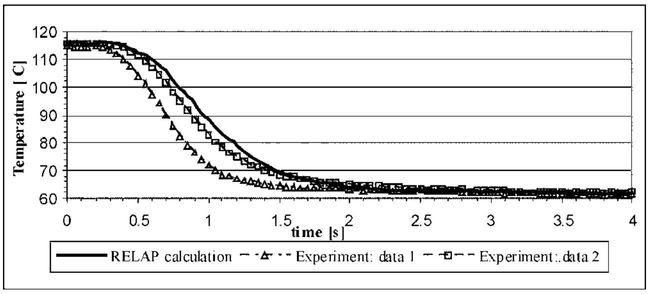

Fig. 7. Coolant temperature at the fuel assembly outlet.

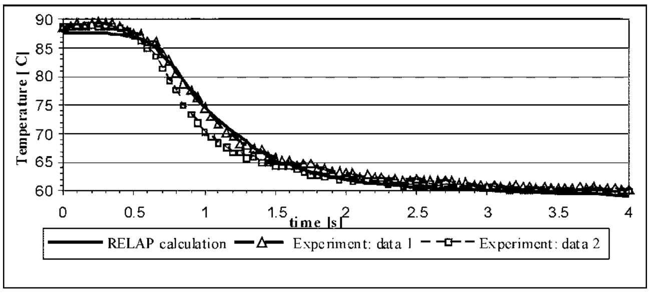

Fig. 8. Coolant temperature in lower plenum of the fuel assembly.

Qualification against experimental data

The experiment with reactor scram and without pump costdown [4] 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).

All experimental data related to the initial and boundary conditions were defined through the peak FA power *vs*. time as well as evolution of the flow rate and the coolant temperature at the peak FA inlet.

The kinetic module in the basic input deck has been deactivated because the power evolution was known explicitly before and after scram. Other changes to the basic input deck were limited to the redefinition of the inlet flow conditions within modules V.100 and J.110 (see Fig. 1) as well as to the small modifications related to time step control, trip data and printed output data.

Comparisons in Figs. 7 and 8 showed that the agreement with the experimental data is quite good.

Transient calculations with RELAP5/MOD3 for the MARIA reactor

In this chapter, the capability of the developed RELAP5 input model is demonstrated through the RELAP5/MOD3 calculations of the MARIA reactor under loss of flow conditions and reactivity insertion transients.

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 consequence, this leads to the complete loss of forced flow through the fuel channels.

Initial conditions at the initiating event onset were determined by pre-transient steady state calculations reported in the previous chapter. The main system parameters of the system model at time of transient initiation were summarized in Table 1. The reactivity feedback coefficients and power distribution characteristics were applied according to Ref. [1].

In the analyses, 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 s was applied. Assuming the constant acceleration of the safety rods (2.7 m/s^2) [1], the following reactivity *vs*. time was inserted into the core during scram:

Time after scram signal (s)	Inserted reactivity ($)
0.37	–0.0241
0.48	–0.1740
0.57	–0.5527
0.64	–1.1860
0.76	–3.2120
0.87	–5.8130
0.91	–6.8050
1.00	–7.4880

The following experimental correlation was selected [4] for defining the boundary condition related to the coolant mass flow rate *vs*. time after pump trip:

$M = M_0/(1 + 0.28*t)$; where M_0 is the initial value of mass flow rate.

The development of the selected parameters is given in Figs. 9 through 13.

Reactor scram signal appears at 1.3 s into the transient. The neutronic power trend (Figs. 9 and 10) can be explained considering two time periods:

– Up to 1.4 s: the fission power essentially depends upon the neutronic-thermal hydraulic feedback. It continues

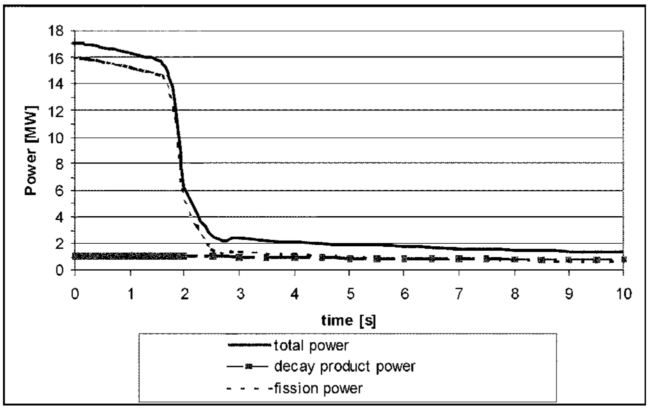

Fig. 9. Total core power, fission power and decay product power.

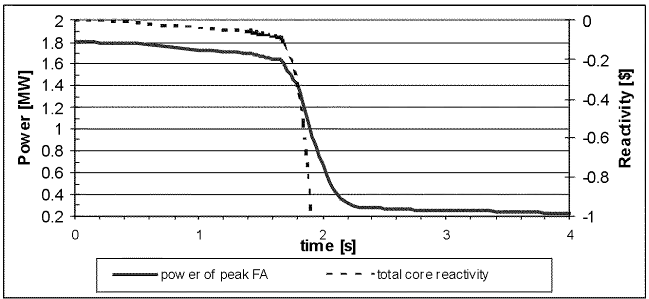

Fig. 10. Power of the peak FA and total core reactivity.

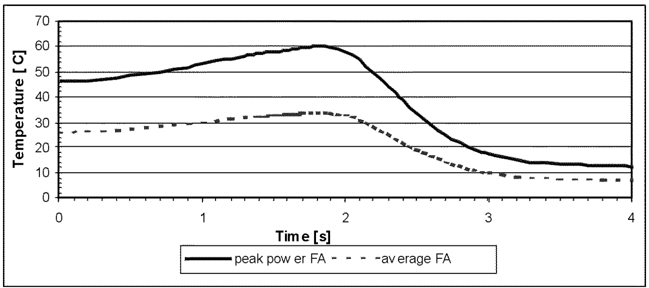

Fig. 11. Increase of coolant temperature at average and peak FA.

to decrease and at 1.4 s becomes about 94% of the nominal power value owing to negative reactivity feedback effects with increase of both fuel and coolant temperature (Figs. 11 and 13). The beryllium temperature feedback coefficient, which is positive for the MARIA reactor, was not considered in the RELAP code model.

– After 1.4 s the scram forces calculated power to decay values.

The maximum cladding temperature is the most important parameter to be calculated by the code, as far as safety aspect is concerned. Maximum temperatures of 141°C and of 145°C were predicted for the cladding and the fuel, respectively. Maximum cladding temperature was reached on the inside surface of the fuel tube heat structure 670-07 (Fig. 1), situated at 0.425 m above the lower edge of fuel elements.

The close relation between fuel/clad temperature trends and heat power balance within fuel elements can be seen easily from Figs. 12 and 13. The temperatures continue to increase from the beginning of the transient up to 1.69 s, and then the heat generation rate within fuel element exceeds the heat power transferred to the coolant. Exactly at 1.69 s, 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.

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.

Figure 14 shows that overestimation of the maximum cladding temperature due to neglecting of reactivity feedback contribution is about 5°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; therefore, subcooled boiling regime is not expected in any spot along the coolant channels during the simulated transient.

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

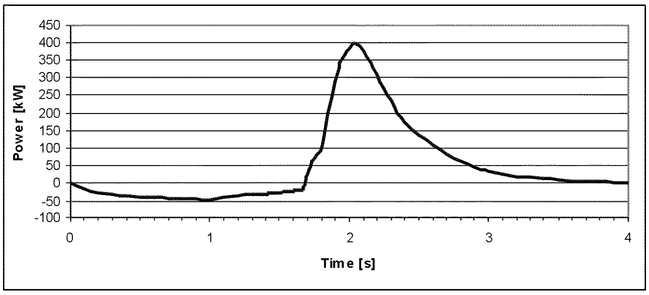

Fig. 12. Increase of heat transfer power above source power at the peak FA.

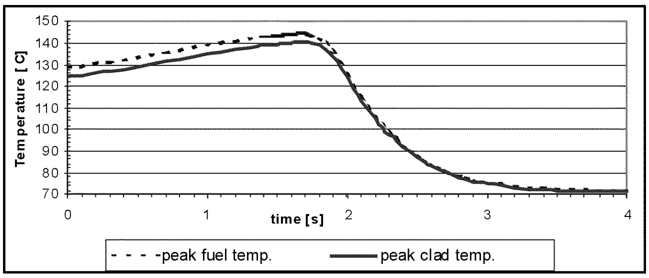

Fig. 13. Cladding and fuel maximum temperature.

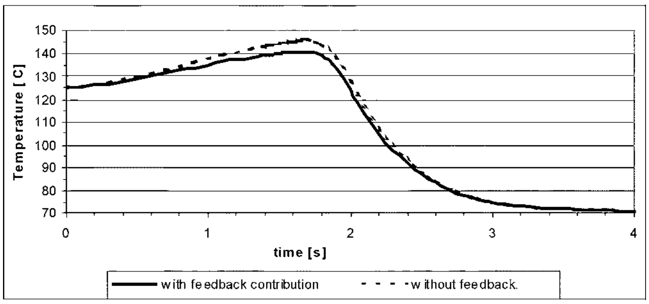

Fig. 14. Comparison of peak cladding temperature.

margin limit for scram activation. Two cases with and without reactivity feedback contributions were recalculated. The results are shown in Figs. 15 through 19.

Two curves in Fig. 15 show a substantial difference concerning RELAP5 prediction of the cladding temperature excursion at hot spot of the peak fuel assembly of the MARIA reactor in these two cases. 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. It can be seen easily from Fig. 16 that at the moment of cladding temperature excursion the power is well below 50% of the initial power. Such a rate of power decrease is determined by negative reactivity feedback coefficients for the coolant temperature, coolant void, and fuel Doppler coefficient. As stated earlier, the positive beryllium reactivity feedback coefficients [6] were not considered in the RELAP input model.

Due to impairment of forced cooling by failure of the coolant pumps, thermohydraulic flow instabilities (flow excursion) arise in the narrow cooling channels (Fig. 19) owing to steam formation, which results in critical heat flux load being exceeded within a few seconds (Fig. 17). 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. The flow rate is finally stabilized at a very low value where the flow consists of pure steam, which is not sufficient to cool the channel. The drop in the flow rate is practically equivalent to a blockage of the channel and a failure of the cooling system of the associated fuel tubes, which are destroyed within a few seconds.

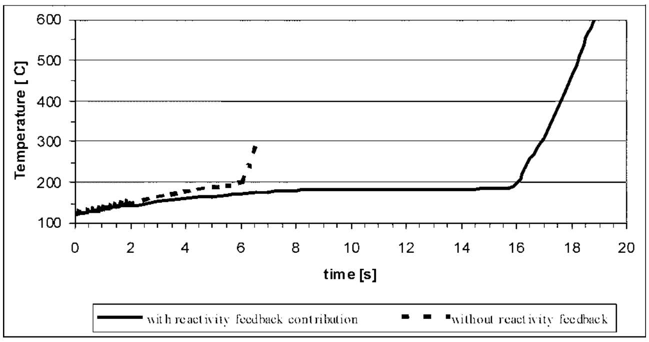

Fig. 15. Unprotected core: maximum cladding temperature.

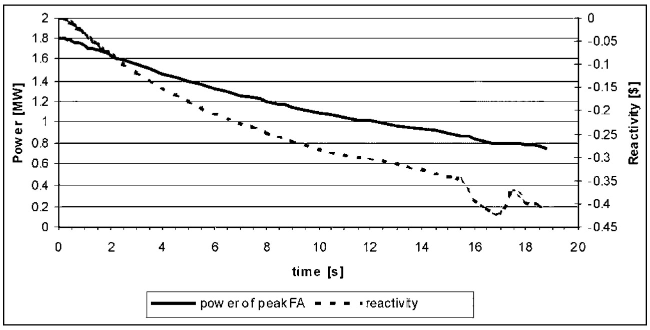

Fig. 16. Unprotected core: power of the peak FA and total core reactivity.

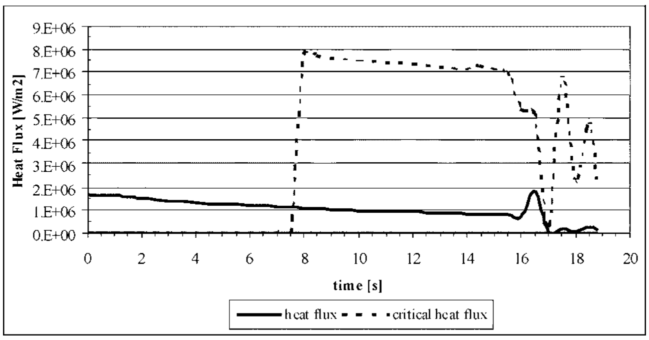

Fig. 17. Unprotected core: heat flux and critical heat flux at hot place.

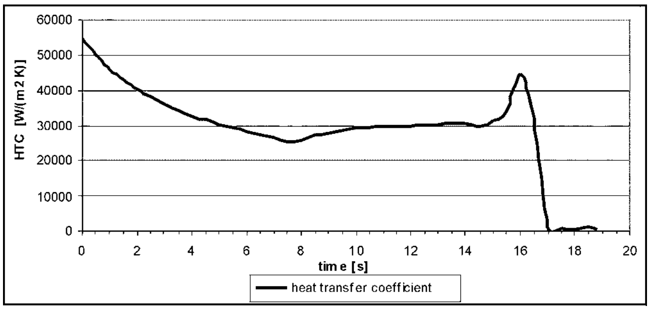

Fig. 18. Unprotected core: heat transfer coefficient at hot place.

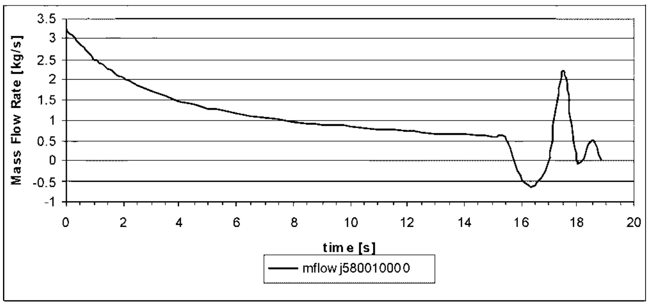

Fig. 19. Unprotected core: inlet mass flow rate to hot channel (V.620) of the peak FA.

Nucleate boiling arises from the fact that boiling is initiated at the cooled heating surfaces, which have to transfer high heat flows at low system pressures, although on the average the coolant has not yet reached the saturation temperature associated with the system pressure. A superheated metastable thermal boundary layer is formed close to the heating surface, in which existing bubble nuclei are activated at the heating surface. The bubble nuclei initially grow in this boundary layer until they reach a maximum size and are detached and enter into the subcooled flow core where they condense after a short residence time. Owing to their high population density, these short-lived bubbles can lead to high volumetric steam contents in the flow channel of up to 80%. Conditions in the channel under consideration are characterized by strong thermodynamic

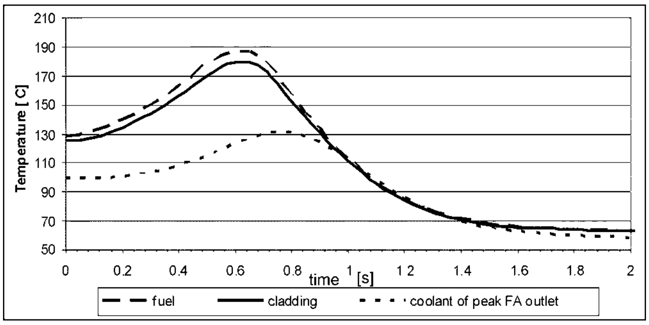

Fig. 20. Peak temperature of fuel, cladding and coolant.

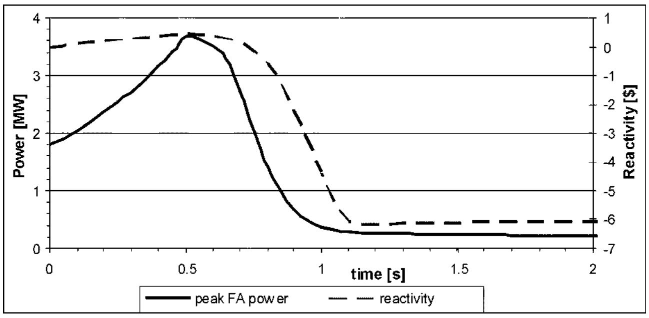

Fig. 21. Peak fuel assembly power and total core reactivity.

non-equilibrium, since the steam bubbles and subcooled liquid exist side by side at different temperatures. Moreover, mechanical non-equilibrium is also present owing to the rate of advance of the bubbles in comparison with the water phase.

Reactivity transient

Figures 20 and 21 show the response of the MARIA reactor to the reactivity insertion rate $ 1.1/s for 1.0 second. The transient starts from reference core steady state nominal power of 17 MW. The scram is initiated when the power exceeds 20% of the nominal reactor power. The time delay of the reactor protection system equal to 0.1 s was applied.

These plots show that the peak values for reactor power, fuel temperature, and cladding temperature are well within acceptable limits, even though the local coolant temperature at the hot place reaches the saturation temperature.

Conclusion

The paper presents the current status of the work related to application of the thermohydraulic system code RELAP5/MOD3 to the analysis of the MARIA research reactor.

The RELAP5/MOD3 input data model of the MARIA research reactor has been developed to provide the capability for the analysis of the reactor core under the 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. [1], 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 the 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 the reactivity feedback was found.

The presented input data model should be treated as the first step for developing of the model including the primary cooling circuit representation. Such a model should provide a capability for the RELAP5/MOD3 analysis of the MARIA reactor under larger spectrum of transients including LOCA accidents.

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