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Safety analysis of reactivity insertion accidents in a heavy water nuclear research reactor core using coupled 3D neutron kinetics thermal-hydraulic system code technique

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ABSTRACT

Nuclear power plant Safety analysis using coupled 3D neutron kinetics/thermal-hydraulic codes technique is increasingly used nowadays. Actually, the use of this technique allows getting less conservatism and more realistic simulations of the physical phenomena. The challenge today is oriented toward the application of this technique to the operating conditions of nuclear research reactors. In the current study, a three-Dimensional Neutron Kinetics and best estimate Thermal-Hydraulic model based upon the coupled PARCS/RELAP5 codes has been developed and applied for a heavy water research reactor. The objective is to perform safety analysis related to design accidents of this reactor types. In the current study two positive reactivity insertion transients are considered, SCRAM protected and self-limiting power excursion cases. The results of the steady state calculations were compared with results obtained from conventional diffusion codes, while transient calculations were assessed using the point kinetic model of the RELAP5 code. Through this study, the applicability and the suitability of using the coupled code technique with respect to the classical models are emphasized and discussed.

1. Introduction

According to the IAEA recommendations (IAEA RCP J7.10.13, 2009) a tendency to perform accident analysis for nuclear research reactors by using well known and validated Best Estimate (BE) codes is stated. The idea is to get benefit from advanced and validated computer codes that were developed for the nuclear power plants (NPP) reactors and getting more qualified analysis in comparison with those obtained generally using in-house or conservative computational tools. Several attempts have been performed to apply Thermal Hydraulic System Codes (THSC), to simulate phenomena occurring in the core and the coolant loop by adopting a level of detail that corresponds to thousands of nodes, roughly. However, the afore-cited programs could perform only pseudo-BE calculations since the neutronic models embedded into them are generally limited to the point kinetic or fixed power distribution models (Hamidouche et al., 2004; Bousbia-Salah and Hamidouche, 2005; Hedayat et al., 2007; Bokhari et al., 2007). On the other hand, BE Neutron Kinetics (BE-NK) codes use generally simple thermal-hydraulic models (Meftah et al., 2006; Khattab et al., 2006; Waqar et al., 2008). This lacking could be overcome by coupling advances TH and NK codes. However, applications of coupled code method to research reactor safety analyses are up to now very limited (Rosenkrantz et al., 2014; Hamidouche et al., 2009).

Therefore the pursued aim of the present study is to extend the applicability range of the BE coupled codes technique in performing safety analysis of a heavy water research reactor. The full core was modeled using the 3D Neutron Kinetic (3D NK) diffusion code PARCS (Downar et al., 2004a). The neutron kinetic diffusion equations are solved using the nodal method, with two neutron energy groups. On the other hand, the Thermal Hydraulic (TH) modeling was carried using (THSC) RELAP5/3.3 (Ransom et al., 1990).

Actually, two typical Design Basis Reactivity Insertion Accidents(DBA-RIA) are considered. The first one concerns a rapid power excursion transient in which the power runaway is stopped

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by the SCRAM signal. The second transient concerns a self-power limiting transient in which the power excursion is stopped by the inherent feedback mechanisms. In absence of experimental data, the coupled code results were compared against the best estimate standalone RELAP5 calculations based upon conservative value of the point kinetics feedback coefficients. On the whole, the coupled codes technique has been successfully applied to simulate the reactor behavior, under steady state and RIA conditions.

2. Coupling approach and computational tools

The use of coupled codes technique is suitable for transients involving strong feedback between the core kinetic and the coolant loop as well as in situations where the core power excursion is important and its distribution changes during the transient (Bousbia-Salah and D'Auria, 2007). Under such situations, the accuracy of the analysis is improved significantly by modeling the interaction of the neutron kinetics and the fluid dynamics. This is particularly true for the simulation of almost all RIA transient cases. For this purpose, macroscopic cross section library have to be calculated and then incorporated in the coupling. The coupling is carried out through the parallel virtual machine (PVM) processing approach(Geist et al., 1994). It allows the PARCS and RELAP5 codes to be run separately by exchanging data during the calculation(Bousbia-Salah and D'Auria, 2007; Bousbia-Salah, 2004). As shown in Fig. 1, the data exchange between the PARCS and RELAP5 code is carried-out via an intermediate file that allows the PARCS code to get all the necessary thermal-hydraulic data from RELAP5. The PARCS codes will therefore evaluates the feedback using the fuel and coolant temperatures calculated by RELAP5 and calculates the neutron flux and turns back to the RELAP5 the instantaneous core power distribution.

3. Modeling issues

The modeling is achieved by the development of various issues related first to the generation of the macroscopic cross sections of the fuel elements. The later will be used by the reactor core kinetics model which will be run simultaneously with the developed model for the core and associated cooling circuit. These topics are addressed in details hereafter.

3.1. Cross section modeling

In the PARCS code the macroscopic nodal cross sections are function of square root of the effective fuel temperature, moderator temperature and density, and the boron concentration (Bousbia-Salah et al., 2006). Under the research reactor operating conditions, only the cross sections related to the fuel temperature and moderator densities are significant (Bousbia-Salah and Hamidouche, 2005). The latter are given according to the following perturbation formula (Downar et al., 2004b):

Table 1
Energy boundaries.

Energy group	Energy range	Remarks
1	10 MeV to 0.625 eV	Fast
2	0.625 eV to 0.0 eV	Thermal

$$\Sigma\left(T_f,\rho_m\right) = \Sigma_{Ref} + a_1\left(\sqrt{T_f} - \sqrt{T_f^{Ref}}\right) + a_2\left(\rho_m - \rho_m^{Ref}\right) \qquad (1)$$

where:

$$a_1 = \left(\frac{\partial\Sigma}{\partial\sqrt{T_f}}\right)_{Ref} \quad ; \quad a_2 = \left(\frac{\partial\Sigma}{\partial\rho_m}\right)_{Ref}$$

Subscript “Ref” denotes values calculated at reference fuel temperature and moderator density (T_f, ρ_m).

To solve the neutron kinetics equations, the macroscopic cross section library for various materials in the core are evaluated using the WIMS-D4 lattice code (Askrew et al., 1966) which is part of the MTR_PC package (Villarino, 1995). First flight collision probability option was used to generate group macroscopic constants for various fueled and non-fueled core regions. The latter were modeled using the cluster geometry option of the code. The calculations were done by considering 69 library group structure with 24 groups being thermal hence a cut off energy of 0.625 eV was applied.

Reactor dependent two-group homogenized cross section data, including scattering, absorption, fission and diffusion coefficients were then obtained, using the POST_WIMS routine (Villarino and Lecot, 1995a) by collapsing the 69 multi-group data into 2 energy groups. The energy boundaries are listed in Table 1 below.

In the current study, six (06) fuel temperatures and seven (07) heavy water densities were chosen to generate the rodded (with the control rods absorber element) and unrodded (without the control rods absorber element) macroscopic cross section tables in order to cover a large set of core conditions under normal and transient conditions.

3.2. Neutron kinetics modeling

The PARCS code is used to evaluate, in a three-dimensional geometry, the space-time distribution of the core power flux. For this purpose it uses a non-linear nodal method to solve two-energy group diffusion equations (Downar et al., 2004b). In this framework, the core is defined in XYZ geometry. Several core regions have been identified according to the material variation. Radially, the core is divided into 17X17 cells containing fuel assembly or reflector material. Each of the fuel assembly is numerically represented by one homogenized node region. Each node is

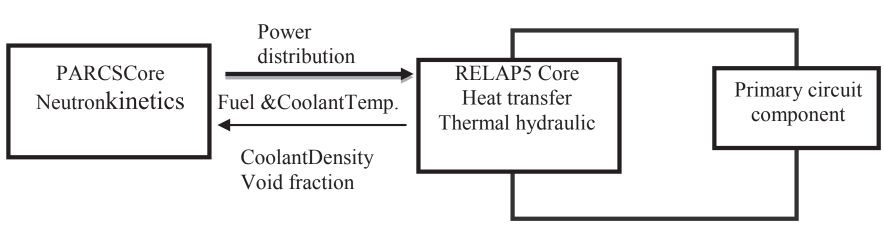

Fig. 1. Parallel coupling process.

characterized by its set of macroscopic cross section library. Axially, the core is divided into 23 layers, 21 layers in the fuel plus top and bottom reflector region. The control rods are grouped in 6 banks including the regulation and safety rods. The neutron kinetics model uses two prompt neutron energy groups and fifteen delayed neutron groups, while the boundary conditions for the neutron diffusion equation is reflective at the outer reflector surface.

3.3. Thermal hydraulic modeling

The developed RELAP5 model for the heavy water reactor is shown in Fig. 2. It includes the main reactor components such as the core zone, the holdup tank, the main coolant pump, and the heat exchanger (Boulhaouchet et al., 2003). For the core zone, special attention is made since an adequate channel mapping should be performed for the coupling process with the kinetic core of the PARCS code.

In the present case, the core is represented by 2 channels representing individually the central and peripheral zones of the core. Such modeling reflects the real operating condition of the reactor. Actually, the central and peripheral zone are not identical from the design point of view, they have slightly different thermal-hydraulic parameters as the flow area, the channel number, the friction losses, the coolant flow rate, etc.. In fact, detailed thermal-hydraulic requires significant computational resources. The costs could be reduced when collapsing similar assemblies into a single thermal-hydraulic channel. Such approach is suitable and sufficient due to the small number of fuel assemblies of the research reactors cores with respect of PWR or BWR nuclear power reactors.

The fuel elements are connected to the same upper and lower plenum components using two multiple junctions components to specify the inner and upper boundary conditions. As the PARCS kinetic model, each active part of the fuel elements together with the coolant region are divided into 21 axial nodes.

4. Calculation results and discussions

4.1. Hot zero power (HZP) calculations

Firstly, qualitative assessments at steady state calculations level are carried out to check the validity of the developed kinetic model

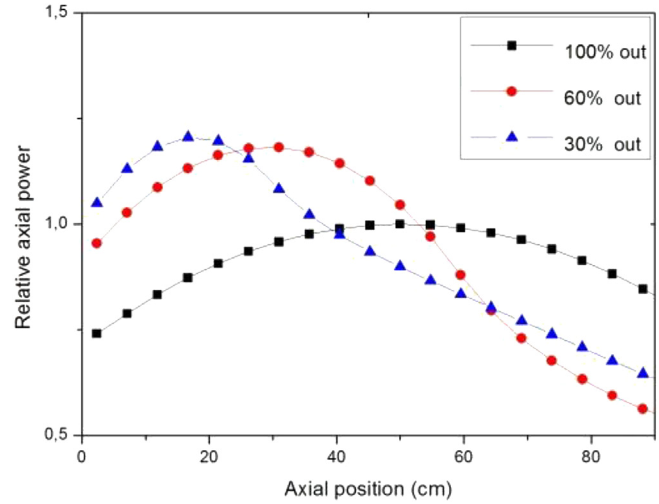

Fig. 3. Control rod position effect on axial power profile.

(core nodalization, control rod positions and efficiency, fuel assembly's compositions, burnup …). These calculations are carried out at the so-called hot-zero-power (HZP) conditions (Ivanov B. and Ivanov K., 2002) where different control rods insertion configurations; 100%, 60% and 30% out are evaluated. As shown in Fig. 3, the coupled PARCS/RELAP5 codes give, from the qualitative point of view, good axial power profile distributions. This is a good indication about the quality of the used cross sections tables, and more particularly the rodded and unrodded sets.

4.2. Criticality calculations

After the HZP assessments, the PARCS/RELAP5 model was run for criticality calculations. The objective is to evaluate the multiplication factor k_{eff} for the considered core configuration. For this purpose, the initial power level was set to 100% of its nominal value and all the control rods were 100% out of the core. The calculated eigenvalue is afterwards compared to the results obtained from other diffusion, namely the 3D diffusion code CITVAP (Villarino and Lecot, 1995b). It is found that the calculated k_{eff} is within the range

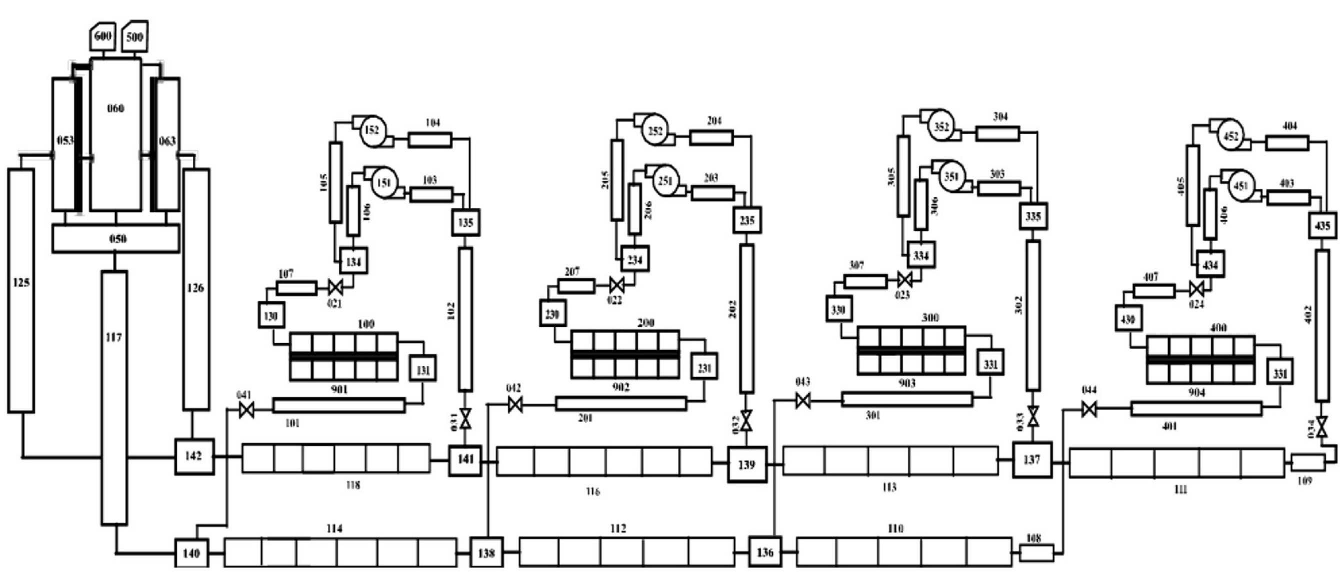

Fig. 2. Reactor nodalization scheme.

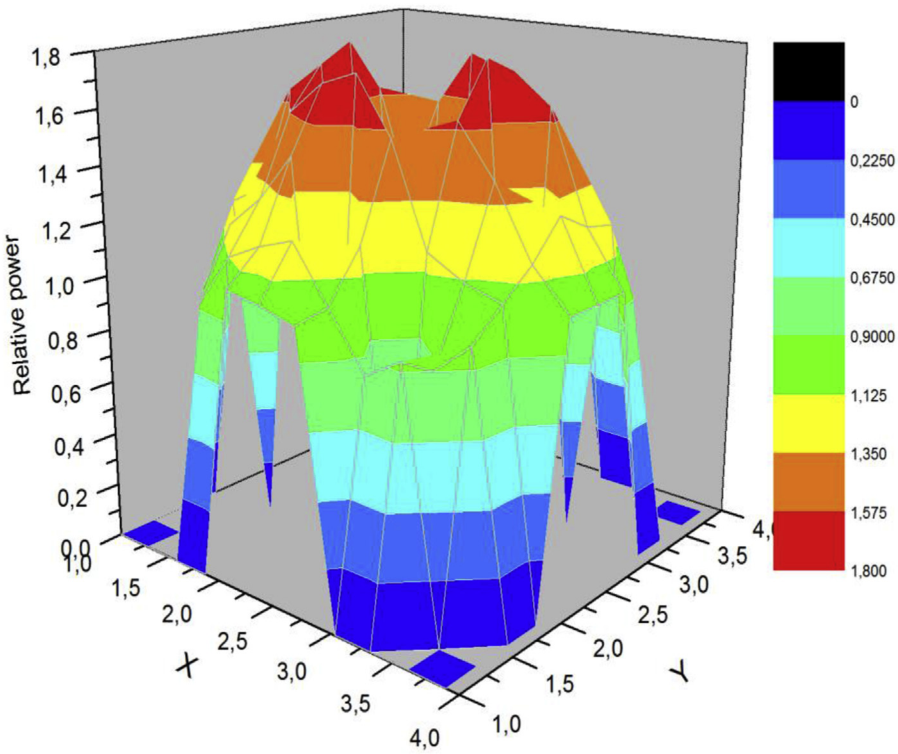

Fig. 4. Radial power distribution at steady state level.

of the one calculated by CITATION with a deviation of 0.30%. On the other hand, the calculated steady state core radial power profile distribution is sketched in Fig. 4.

4.3. Transient calculations

The nuclear power can rapidly increase as consequence of a positive reactivity addition into the core. There are many causes that lead to reactivity change. These are generally classified in two categories; the reactivity induced by control rod withdrawal and reactivity induced by the inherent feedback mechanisms.

In this study, two design basis abnormal insertions of reactivity accidents (DBA) are selected: the uncontrolled withdrawal of regulating rod and a sudden decrease in coolant temperature.

4.3.1. Control rod withdrawal accident

At the beginning of the accident, two regulating rods are located at half height in the core. The one is on automatic regulating conditions, and the other is on standby condition. The accident starts when the regulating rod on automatic condition is uncontrolledly withdrawn. After the regulating rod reaches its top position, the second regulating rod state is changed from standby condition to automatic condition and again it is uncontrolledly withdrawn. Two regulating rods are thus at the top of the core in 2 s. During this time period the amount of inserted reactivity is about 1.5$. As a consequence, the reactor power exhibits an exponential rise, and when the power achieves 120% of its nominal value, the SCRAM signal is sent out. The SCRAM becomes effective with a delay time of 0.26 s. The safety rods start to drop and they are fully inserted within a period of 1 s.

The relative core power profile at the SCRAM time is shown. Fig. 5. As could be seen the power profile shows higher value in the zone surrounding the central trap and decreases in the vicinity of the control rods locations (blue color punctual zones). On the other hand, Fig. 6 shows the global power evolution during the transient in comparison with the point kinetics model. Indeed, the transient was also simulated by the REALP5 standalone, using the point kinetic module, in order to get a code-to-code comparison purposes.

As it could be noticed, the main difference between the coupled 3D NKTH and the stand alone point kinetic model appears during the excursion phase during which the 3D NKTH power shows a faster excursion. This difference is mainly due to different feedback modeling invoked during the transient and also to the power profile. Indeed as shown in Fig. 7, the axial power profile calculated by the 3D NKTH evolves in time while it is not the case for the point kinetics one. Furthermore, the point kinetic feedback coefficients are fixed in time; they represent averaged values which are generally artificially increased for conservatism purposes. On the other hand, the feedback predicted by the coupled 3D NKTH code are instantaneously evaluated in space and time. The evaluated feedback by the 3D NKTH exhibits lower values with respect to the used point kinetics values. Consequently a higher peak power is obtained.

As could be seen in Fig. 8, one can notice that in both cases the power excursion is stopped just after the insertion of all other control rods. The Doppler feedback effects, which start to become effective during the power excursion is not strong enough to slow-down the power excursion and the feedback due to the coolant temperature rise becomes effective after the SCRAM time.

4.3.2. Coolant temperature decrease accident

This accident is characterized by a positive reactivity insertion in the core following an inherent feedback mechanism due to the positive feedback effect of the coolant temperature change. As in the first case, the transient course is simulated numerically using the coupled PARCS/RELAP5 codes together with the RELAP5 standalone for comparison purposes. At the beginning of the transient the reactor is operating at 100% of its nominal power

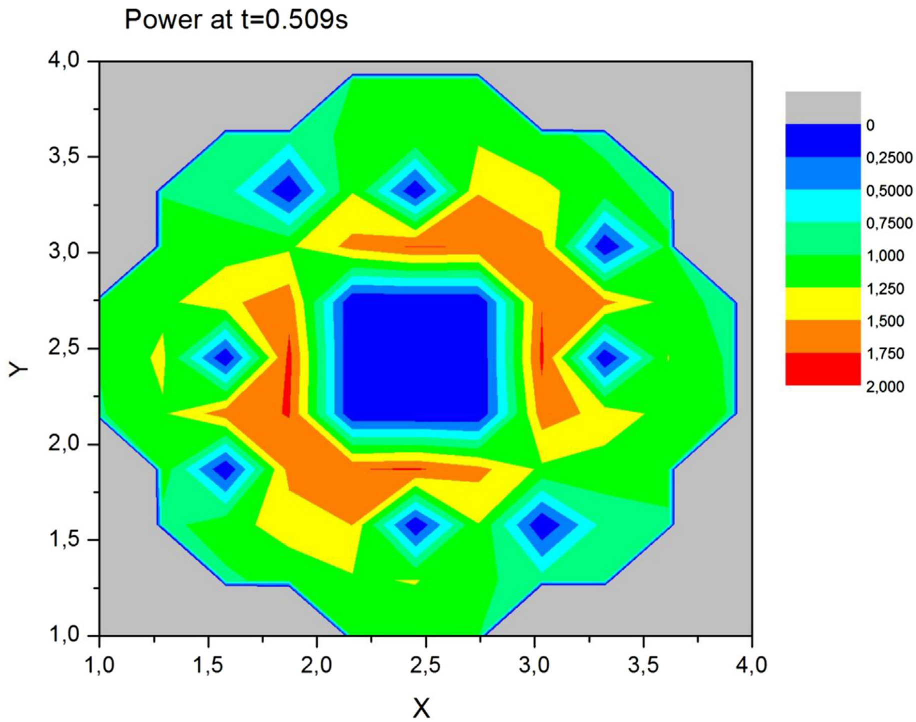

Fig. 5. Relative radial core power distribution.

value, and all the heavy water pumps are in operation. The transient starts due to a malfunction of the heat exchanger leading to a coolant temperature decreased by 10 °C in 0.2 s. Consequently, due to the negative feedback effect of the coolant, a positive reactivity is added into the core as soon as the cold slug reaches the core zone. Consequently, the core power starts to rise. However due to the fact that the value of the inserted reactivity is low with respect to the first case, the power, as could be seen in Fig. 9, exhibits a self-limiting behavior (Bousbia-Salah and Berkani, 2001). The power rise is stopped by the feedback mechanisms due to the combined effects of the Doppler and coolant feedback (see Fig. 10). The latter are able to compensate the added reactivity and consequently limit the power rise without triggering the SCRAM signal.

As it was observed in first case, stronger feedback are predicted by the RELAP5 standalone (see Fig. 9) since it uses conservative feedback coefficients that allows getting higher amount of inserted positive reactivity in the core. The power rise is faster and the late power decrease is stronger. On the other hand, the feedback, as evaluated by the macroscopic cross section variation of the PARCS/RELAP code, are weaker and subsequently, slower power rise is observed and a higher value of the power at the end of the transient is reached.

5. Conclusion

The current study constitutes an attempt for a contribution of extending the use of coupled code method to the operating ranges of nuclear research reactors. Through this study the applicability of the coupling of advanced thermal-hydraulic and 3D neutron

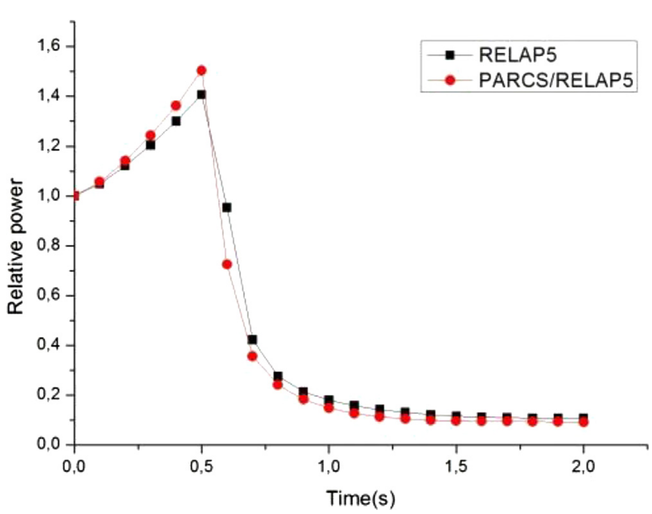

Fig. 6. Core power evolution during control rod withdrawal accident.

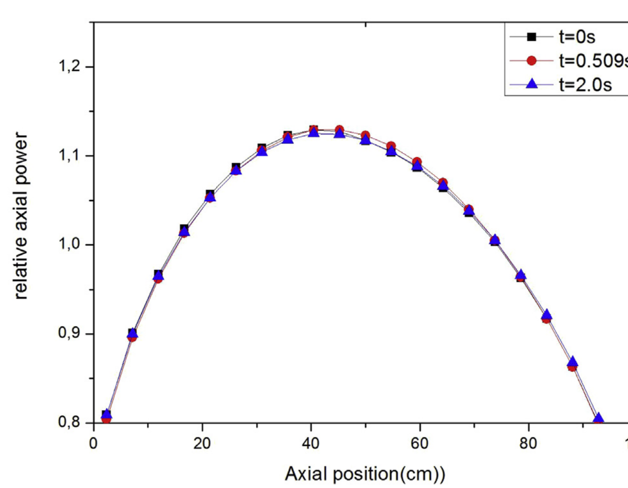

Fig. 7. Axial power distribution during RIA.

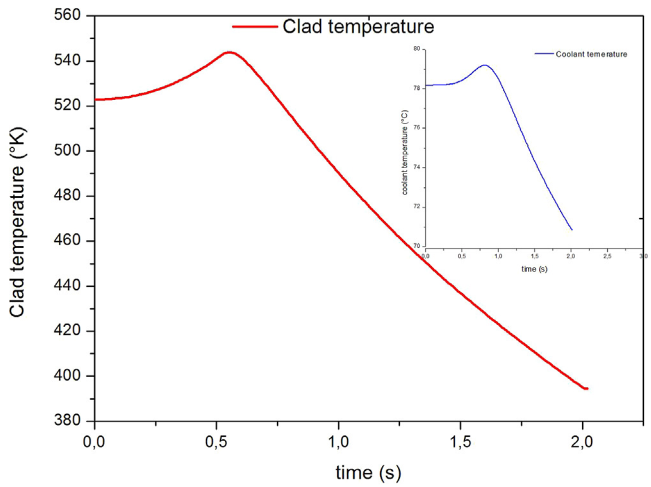

Fig. 8. Fuel clad and coolant temperature during the transient.

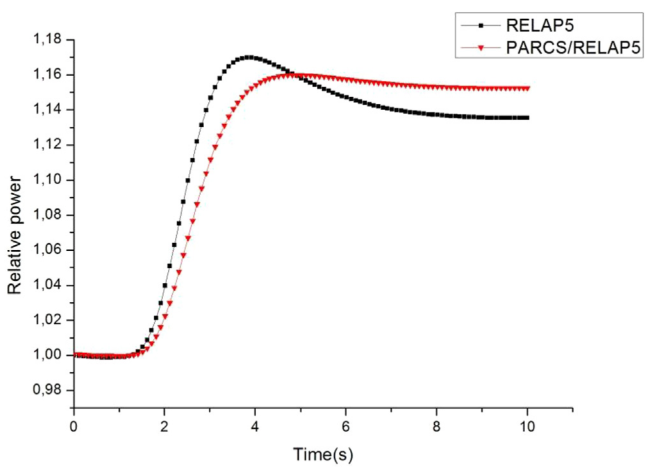

Fig. 9. Core power evolution during coolant temperature decrease accident.

kinetics codes to heavy water research reactors is achieved by using the coupled code PARCS/RELAP5 under steady state and transient conditions. In absence of experimental data, the coupled code

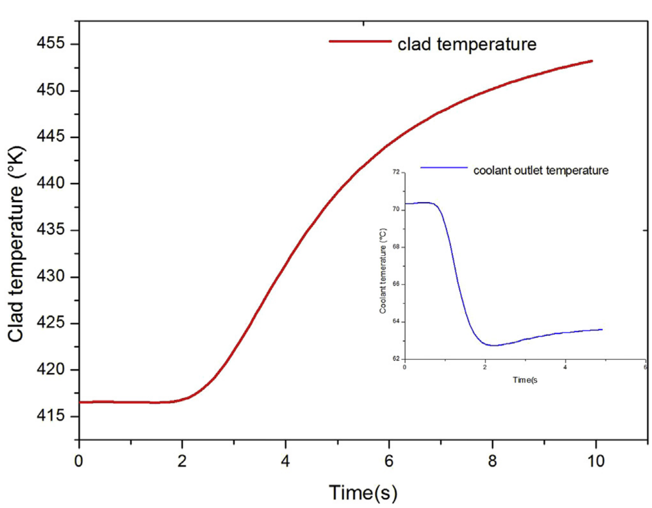

Fig. 10. Clad and coolant temperature during the transient.

results were compared against the best estimate standalone RELAP5 calculations based upon conservative value of the point kinetics feedback coefficients. The calculations show that the conservatism related to the use of BE code using point kinetics model could provide non-conservative results. Indeed, the use of coupled BE tools may lead to more conservatism in comparison with less detailed evaluation tools. Therefore additional work should be performed to validate the coupled code approach through the validation matrix for RR that will be published soon by the IAEA, and to revisit the conservatism obtained through the use of less advanced computational tools.

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Acronyms

BE best-estimate
CR control rod
CSC cross section code
DBA design basis accident
FA fuel assembly
HZP hot zero power
IAEA International Atomic Energy Agency
NK neutron kinetic
NKC neutron kinetic code
NPP nuclear power plant
PVM parallel virtual machine
RIA reactivity initiated (or induced) accident
RR research reactor
TH thermal-hydraulic
THSC thermal-hydraulic system code
3D or 3-D three-dimensional

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