MONSTR/RELAP5: CODE SYSTEM FOR COMPREHENSIVE MODELING OF TRANSIENTS AND ACCIDENTS IN PRESSURIZED WATER-COOLED NUCLEAR REACTORS

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ABSTRACT

The structure, methods and assumptions of the MONSTR/RELAP5 code system are described in the paper. The system was developed for calculation of 3D LWR core neutron kinetics, thermal hydraulics and thermal mechanics of fuel rods with account for reactivity feedbacks due to change in temperature and nuclear densities of core materials as well as feedbacks between fuel rod deformation and cladding-to-coolant heat transfer. In particular, reactivity-initiated accidents and loss-of-coolant accidents in VVER reactors are supposed to be modeled with the MONSTR/RELAP5 system. As a step of verification procedure, the system was used for calculation of AER-4 benchmark problem, which is a 3D dynamic benchmark for the VVER-440 reactor, simulating a cooler deborated water slug entry in the core. The results of the developed system are compared in the paper with the results obtained by the codes, widely used for LWR analysis.

1. INTRODUCTION

The structure, methods and assumptions of the MONSTR code system (Modeling Of Nominal Steady-state and TRansient conditions) are described in the paper. The MONSTR system includes JAR-IQS[1], GETERA[2], PANORAMA[3], FRED[4] codes, developed in RRC "Kurchatov Institute" and Moscow Engineering Physics Institute. The MONSTR system was coupled with the RELAP5/MOD3[5] code (MONSTR/RELAP5) for joint simulation of 3D core neutron kinetics, thermal hydraulics and thermal mechanics of fuel rods with account for reactivity feedbacks due to change in temperature and nuclear densities of core materials as well as feedbacks between fuel rod deformation and cladding-to-coolant heat transfer. The objective of the MONSTR/RELAP5 code system is to simulate, in particular, reactivity-initiated accidents (RIA) and loss-of-coolant accidents (LOCA) in light-water reactors (LWR) of VVER types.

The codes, included in the MONSTR/RELAP5 system, were extensively verified against experimental and/or analytical data [1-5]. As a step of verification procedure for the whole system, the MONSTR/RELAP5 was used for calculation of AER-4 benchmark problem[6], which is a 3D dynamic benchmark for the VVER-440 reactor, simulating the entry of a cooler deborated water slug in the core. The aim of the benchmark was comparison of calculational results obtained with different 3D core dynamics codes, emphasising on 1) simulating the coupled neutron kinetics and thermal hydraulics; 2) testing the tracking of boron and temperature fronts in the core; 3) using the own nuclear data libraries. The MONSTR/RELAP5 results are compared in the paper with results obtained by the HEXTRAN, KIKO3D, BIPR8, DYN3D codes, widely used for LWR analysis[7].

2. MONSTR/RELAP5 SYSTEM

The core calculational scheme consists of control volumes, which are square or hexagonal prisms, simulating axial layer of fuel assembly. Each control volume is divided in elementary cells or polycells, grouped in several types. As a rule, each fuel assembly is described by a thermal-hydraulic channel, heated by fuel rods, grouped in one or several types. Axial nodalization of the channel and fuel rods is the same as in neutronics calculations.

The structure of the MONSTR/RELAP5 system is presented on Figure 1. Main blocks of the system include: 1) GETERA code for elementary cell or polycell neutronics calculation, 2) PANORAMA code for 2D fuel assembly neutronics calculations, 3) JAR-IQS code for 3D reactor neutron kinetics calculation, 4) FRED code for fuel rod thermal mechanics calculations and estimation of fuel rod failure probability, and 5) RELAP5/MOD3 code for core and primary system thermal hydraulics calculations. All these codes are linked in one executable module.

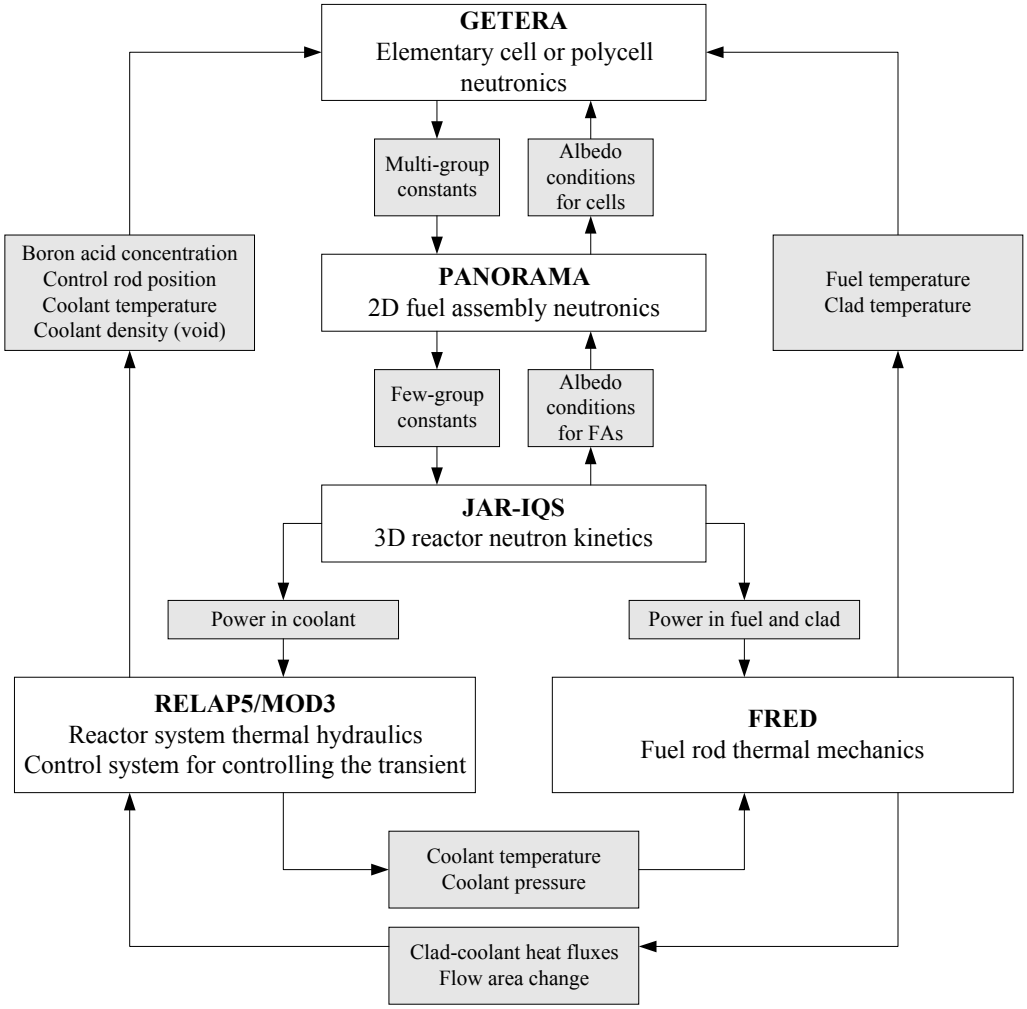


Figure 1. Flowchart of MONSTR/RELAP5 code system

Main approaches and assumptions of these codes (blocks) as well as information flows between the blocks are described below.

2.1 GETERA: CELL OR POLYCELL NEUTRONICS

The GETERA code[2] uses first-collision-probability method in cylindrical, spherical or slab geometry for calculation of spatial-energy neutron distribution in an elementary cell or polycell of different types (fuel rod/water, absorber rod/water, tube/water, shroud/water, etc). The GETERA library includes thermal and epithermal neutron data, generated from ENDF/B-4 and ABBN files, respectively. 100 and 22 energy groups are used in thermal and epithermal regions, respectively. The neutron spectrum in thermal region is calculated by differential models with specially generated functions, characterizing neutron scattering law for different moderators (water, graphite, etc.) The self-shielding factors for cross sections of resonance isotopes in epithermal region are calculated with account for the specified cell geometry.

As a part of the MONSTR/RELAP5 system, the GETERA uses boundary conditions of albedo type for each energy group obtained from PANORAMA block, which calculates the whole fuel assembly neutronics. Thus, neutron spectrum in the elementary cell is calculated with account for neighboring fuel and absorber rods. Neutron macroscopic constants in the MONSTR/RELAP5 system are re-calculated during a transient, when one or several parameters of reactor materials have significantly changed their values since previous re-calculation. These parameters include: boron acid concentration in coolant, control rod position, coolant temperature and coolant density (void), all calculated in the RELAP5/MOD3 block of core thermal hydraulics, as well as fuel and clad temperatures, calculated in the FRED block of fuel rod thermal mechanics. The re-calculated multi-group constants are directed to the PANORAMA block for fuel assembly calculation.

2.2 PANORAMA: FUEL ASSEMBLY NEUTRONICS

The PANORAMA code uses 2D balance method with discrete angular dependence of neutron flux along the elementary cell boundary (PS_n method)[3] for calculation of spatial-energy neutron distribution in fuel assembly. The PS_n method is a combination of the first-collision-probability method and discrete ordinate method. Similar to the discrete ordinate method, angular dependence of neutron flux function along the calculational cell boundary is described in discrete form. Neutron flux function is assumed constant within each of specified angular directions. On the other hand, c oefficients of equation, describing neutron transport from one boundary to another and from t he volume to the boundary within each angular direction, are calculated from the balance corr elations of first-collision-probability method.

For each fuel assembly the GETERA block in the MONSTR/RELAP5 system supplies multi-group constants for each elementary cell or polycell of this fuel assembly to the PANORAMA block, which uses boundary conditions of albedo type for each energy group obtained from JAR-IQS block, calculating the whole reactor neutronics. Thus, neutron spectrum in the fuel assembly is calculated with account for neighboring fuel assemblies. The PANORAMA block reduces multi-group constants for each fuel assembly with the weight of neutron flux density to few-group constants and supplies them to the JAR-IQS block for the whole-reactor calculation.

2.3 JAR-IQS: 3D REACTOR NEUTRON KINETICS

The JAR-IQS code uses multi-group diffusion approximation and improved quasistatic methods[1] for calculation of transient 3D spatial-energy neutron distribution in reactor. A nodal scheme[8] is used for solution of diffusion equation. The use of the nodal scheme allows to provide boundary conditions of albedo type to the PANORAMA block, calculating fuel assembly neutronics.

The JAR-IQS block receives few-group constants from the PANORAMA block and provides 3D power distribution in fuel and cladding to the FRED block of fuel rod thermal mechanics and, if necessary, 3D power distribution in coolant to the RELAP5/MOD3 block of core thermal hydraulics.

A number of features was specially developed for the JAR-IQS code to depress effect of "numerical diffusion" during some transients. In particular, if during a transient some control volume contains a moving boundary, separating two zones with different thermal or neutronics properties (front of deborated or cold water, control rod tip, etc.), than macroscopic constants are averaged for this volume with account for neutron flux axial form and the moving boundary position in the control volume.

2.4 FRED: FUEL ROD THERMAL MECHANICS

The FRED code[4] calculates in r-z geometry interrelated change of temperatures, heat fluxes, stresses, strains in fuel rods as well as estimates fuel rod failure probability. To correctly account for interrelation between channel thermal hydraulics and fuel rod thermal mechanics and to estimate fuel rod failure probability in accidents, the following phenomena are taken into account:

- dependence of thermal and mechanical material properties of fuel pellet on temperature and burnup;
- dependence of thermal and mechanical material properties of cladding on temperature, fluence, strain rates and heating rates;
- change of temperature fields in fuel rod with account for radial and axial heat fluxes;
- fuel-clad gap conductance dependent on gap width, inner gas pressure and composition, possible pellet-clad contact, fuel and clad surface temperature, etc.;
- stress-strain condition of a fuel rod cladding with account for thermal expansion, elastic-plastic deformation and creep in open and closed fuel-clad gap regimes;
- fuel pellet deformation due to thermal expansion, swelling and cracking;
- change in internal gas pressure and composition with account for gas plenum temperature, fuel and clad temperature distribution, fission gas release, fuel rod deformation, etc.;
- probability of cladding failure;
- fission gas release from fuel and fuel swelling;
- cladding oxidation and metal-water reaction;
- change in channel cross section due to cladding deformation.

The FRED block uses temperature and pressure fields in adjacent coolant, calculated in the RELAP5/MOD3 block, as well as power distribution in fuel and clad, calculated in the JAR-IQS block.

2.5 RELAP5/MOD3: CORE THERMAL HYDRAULICS

The RELAP5/MOD3 code[5] is used for modeling of LWR core thermal hydraulics. The LWR core is simulated as a system of parallel channels. The control system of the RELAP5/MOD3 code is used for tracking boron fronts or control rod tip in the core. The coolant temperature front is not tracked.

As a part of the MONSTR/RELAP5 system, the RELAP5/MOD3 block uses boundary conditions for clad-coolant heat fluxes and flow area change due to fuel rod deformation, calculated in the FRED block of fuel rod thermal mechanics.

3. AER-4 BENCHMARK PROBLEM

This 3D dynamic benchmark in hexagonal core geometry simulates a boron dilution event in a VVER-440 reactor core. The core is initially in a hot subcritical state at the beginning of fuel cycle with all control rods inserted. A slug of boron-diluted and cooler water is injected into the core by forced flow, caused by restart of one reactor coolant pump. The core experiences a prompt supercritical power excursion and becomes subcritical again as the slug leaves the core.

The objective of the benchmark is comparison of calculational results obtained with different 3D core dynamics codes, emphasising on

- simulating the coupled neutron kinetics and thermal hydraulics;
- testing the tracking of boron and temperature fronts in the core;
- using the own nuclear data libraries.

Participants are requested to use their own the best-estimate nuclear cross section data, tuning them to reference reactivity parameters, obtained with the HEXTRAN code:

- the boron concentration in the initial state is close to the critical boron concentration of the reference state (the hot zero power state with no control rods inserted);
- the neutronics data for control rods should be adjusted to obtain the subcriticality of the initial state with all control rods inserted close to -10283 pcm;
- the boron concentration in the slug should be adjusted to obtain the static overcriticality of the whole slug close to 2270 pcm.

Some key reactivity parameters in AER-4 benchmark calculated by different codes before tuning and after tuning are shown on Figure 2.

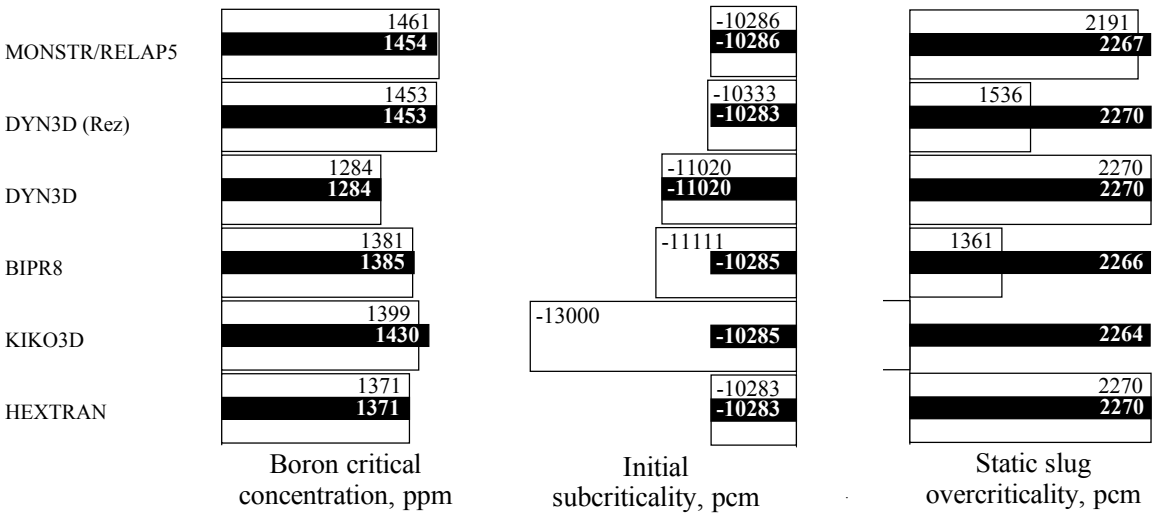


Figure 2. Some key reactivity parameters in AER-4 benchmark calculated by different codes before tuning (white bars) and after tuning (black bars)

Based on the data of [7], some of the MONSTR/RELAP5 results were compared with results obtained by the HEXTRAN, KIKO3D, BIPR8, DYN3D codes, widely used for LWR analysis. The MONSTR/RELAP5 results for time histories of reactor power and reactivity (Figures 3, 4), peak fuel and clad temperatures (Figures 6, 7) in the AER-4 benchmark are in a good agreement with results of other codes. The values and time moments of peak neutron power (Figure 5) as well as peak fuel and clad temperatures (Figures 8) are inside the interval, determined by the spread of the other codes.

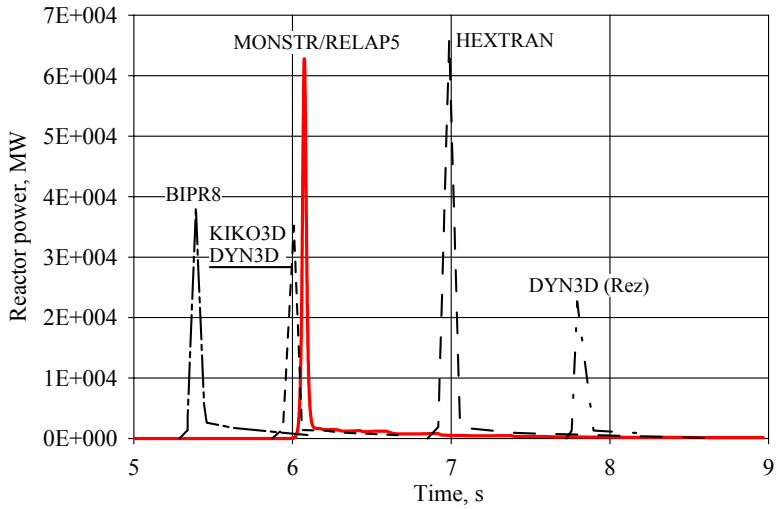


Figure 3. Time history of reactor neutron power in AER-4 benchmark predicted by different codes

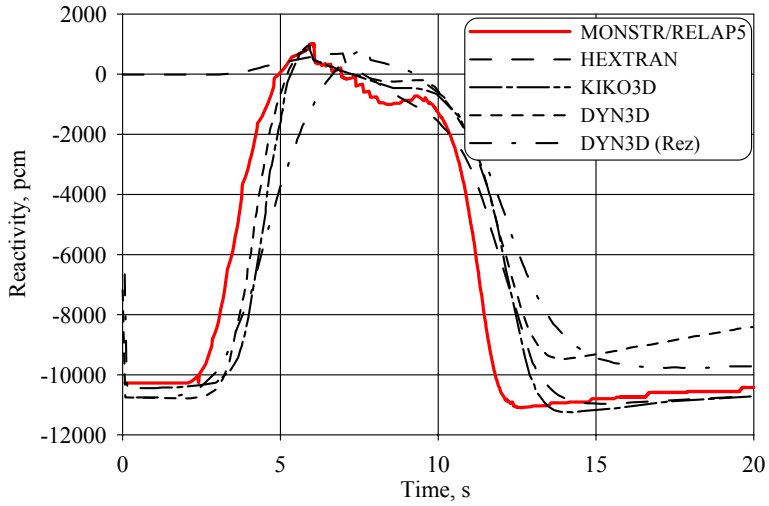


Figure 4. Time history of reactivity in AER-4 benchmark predicted by different codes

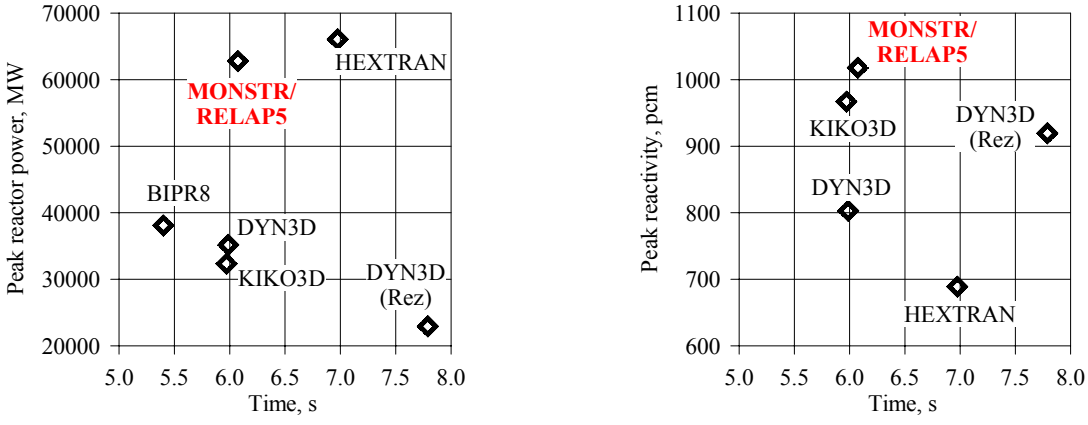


Figure 5. Values and time moments of peak reactor neutron power (a) and peak reactivity (b) in AER-4 benchmark predicted by different codes

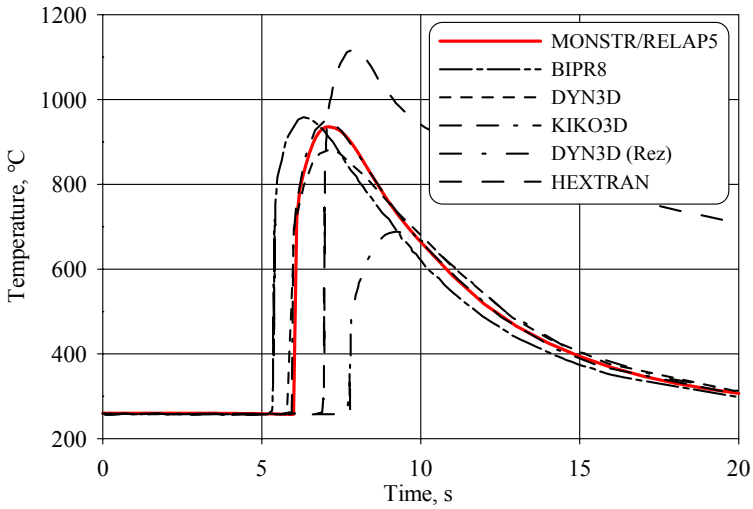


Figure 6. Time history of peak fuel temperature in AER-4 benchmark predicted by different codes

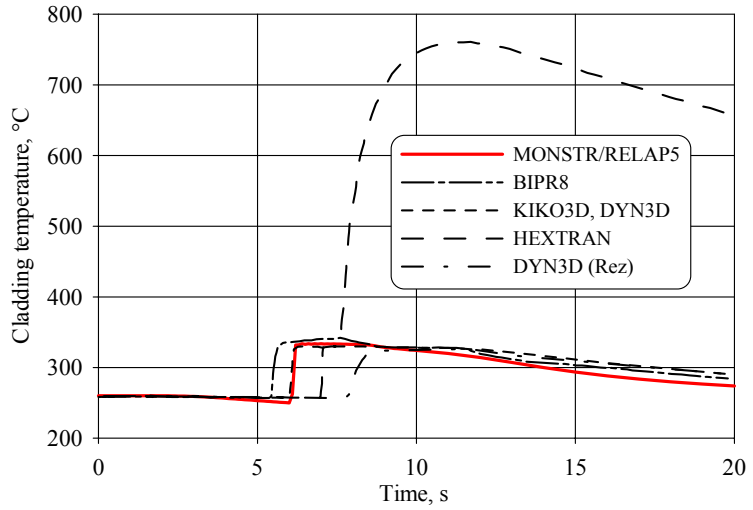
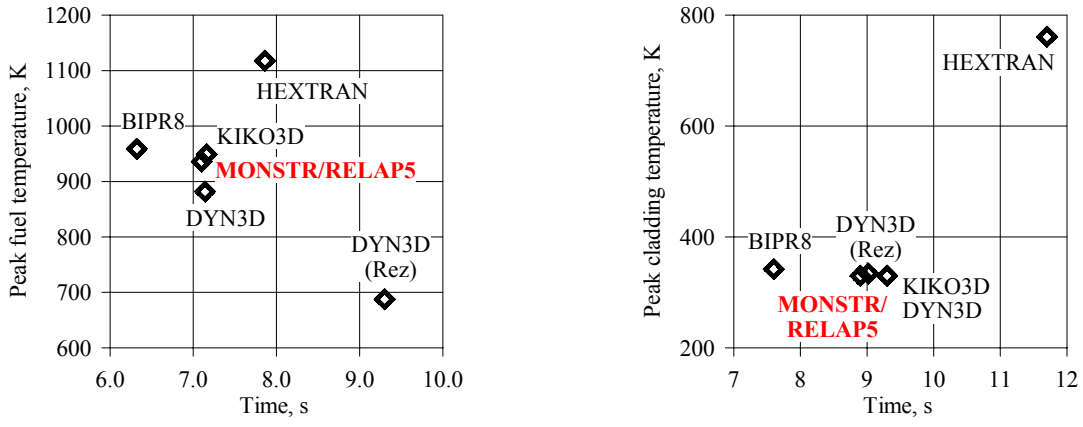


Figure 7. Time history of peak clad temperature in AER-4 benchmark predicted by different codes



(a) (b)

Figure 8. Values and time moments of peak fuel temperature (a) and peak clad temperature (b) in AER-4 benchmark predicted by different codes

CONCLUSIONS

The MONSTR/RELAP5 code system was developed for calculation of 3D LWR core neutron kinetics, thermal hydraulics and thermal mechanics of fuel rods with account for reactivity feedbacks due to change in temperature and nuclear densities of core materials as well as feedbacks between fuel rod deformation and cladding-to-coolant heat transfer. In particular, the system is supposed to be used for modeling of RIA and LOCA in VVER reactors. Main stages of the transient calculation include calculation of macroscopic constants, reactor neutron kinetics calculation, fuel rod thermal mechanics calculation and estimation of fuel rod failure probability, core and primary system thermal hydraulics calculation. When during a transient one or several parameters of reactor materials have significantly changed their values since previous re-calculation, the neutron macroscopic constants are re-evaluated by two-step procedure (elementary cell/fuel assembly calculations).

The codes, included in the MONSTR/RELAP5 system, were extensively verified against experimental and/or analytical data. As a step of verification procedure for the whole system, the MONSTR/RELAP5 was used for calculation of AER-4 benchmark problem, which is a 3D dynamic benchmark for the VVER-440 reactor, simulating a cooler deborated water slug entry in the core. The results of the developed system were compared in the paper with the results obtained by the HEXTRAN, KIKO3D, BIPR8 and DYN3D codes, widely used for LWR analysis. Not large amount of tuning was necessary for the MONSTR/RELAP5 system to obtain some static key reactivity parameters, including initial boron critical concentration, initial subcriticality with control rods inserted, static overcriticality of deborated and cooler water slug. A good agreement was obtained between results of the other codes and the MONSTR/RELAP5 results for time histories of reactor power and reactivity, peak fuel and clad temperatures.

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