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Contact Author: Temitope A. Taiwo
Argonne National Laboratory
9700 South Cass Avenue,
Argonne, Il 60439

Phone: (630) 252 1387
Fax: (630) 252 4500
E-mail Taiwo@anl.gov

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at the International Nuclear Safety Center

Authors:
L. N. Podlazov, O. Y. Kavun, A. I. Popykin, R. M. Shagaliev, A. V. Krayushkin
T. A. Taiwo, F. E. Dunn, J. E. Cahalan and H.S. Khalil

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COUPLED NEUTRONIC AND THERMAL-HYDRAULIC CODE BENCHMARK ACTIVITIES AT THE INTERNATIONAL NUCLEAR SAFETY CENTER

Lev N. Podlazov
Research and Development Institute for Power Engineering (RDIPE)
Moscow, Russia

Oleg Y. Kavun
AtomEnergoProekt (AEP)
Bakuninskaja Street 7
Moscow, Russia

Alexander. I. Popykin
Scientific Engineering Center on Nuclear and Radiation Safety
Moscow, Russia

Rashit M. Shagaliev
Russian Federal Nuclear Center (VNIIEF)
Sarov, Russia

A. V. Krayushkin
RRC Kurchatov Institute (KI)
Moscow, Russia

Temitope A. Taiwo
Argonne National Laboratory
Argonne, IL 60439, U.S.A.

Floyd E. Dunn
Argonne National Laboratory
Argonne, IL 60439, U.S.A

James E. Cahalan
Argonne National Laboratory
Argonne, IL 60439, U.S.A

Hussein S. Khalil
Argonne National Laboratory
Argonne, IL 60439, U.S.A

ABSTRACT

Two realistic benchmark problems are defined and used to assess the performance of coupled thermal-hydraulic and neutronic codes used in simulating dynamic processes in VVER-1000 and RBMK reactor systems. One of the problems simulates a design basis accident involving the ejection of three control and protection system rods from a VVER-1000 reactor. The other is based on a postulated rod withdrawal from an operating RBMK reactor.

Preliminary results calculated by various codes are compared. While these results show significant differences, the intercomparisons performed so far provide a basis for further evaluation of code limitations and modeling assumptions.

I. INTRODUCTION

The U.S. and Russian Federation International Nuclear Safety Centers (INSCs) were established in accordance with the Joint Declaration of the former Gore-Chernomyrdin Commission adopted during the Sixth Session in 1995. The centers provide avenues for cooperation in the areas of nuclear operational safety enhancement, nuclear safety assessments, and risk reduction measures. Various joint projects have been established by the two centers in order to achieve these goals. One of these joint projects, which is on "Coupled Neutronic and Thermal-Hydraulic Codes," establishes R&D collaborations between the two centers on the development, testing, and application of advanced computational capabilities for use in operational safety assessments and analysis of potential accident sequences in Russian and Western reactors. The first phase of this joint project was completed in 1997 and had as its main objectives the exchange of information on coupled code development and testing efforts in the Russian Federation and the U.S. and the development of specifications for benchmark problems that can be used in a systematic verification and validation activity as part of a second phase which is currently underway. The two Centers have recently developed a joint two-year plan that specifies activities in coupled code capability enhancement and efficiency improvement, code verification through joint execution of benchmark tests, and identification of relevant experimental measurements (and their specifications) that could be used for future work on code validation.

In this paper, we present the results of ongoing benchmarking activities in the second phase of the joint project. These benchmarks address dynamic phenomena in Soviet-designed VVER and RBMK reactors. Two benchmark cases have been selected for presentation in this paper. The first of these two problems defines a postulated unprotected rod ejection event in a VVER-1000 reactor at a low power level and beginning of life conditions. The second is a rod withdrawal event in an RBMK reactor operating at nominal conditions.

The descriptions of the benchmark problems are provided in Section II. Section III contains brief descriptions of the coupled neutronic and thermal-hydraulic codes used by benchmark participants for analyzing the problems. The preliminary results currently available are presented in Section IV. The paper ends with a discussion on future plans for this benchmarking activity.

II. BENCHMARK PROBLEMS

A. VVER-1000 Problem Description

The VVER-1000 problem defines a hypothetical ejection of three control and protection system (CPS) rods without scram from the core of a VVER-1000 reactor operating at 10% nominal power and at beginning of life. The benchmark problem is based on a scenario similar to the design-basis accident case recommended for consideration during the substantiation of the safety of VVER-reactor plants. The problem, which is specified by the participant from AtomEnergoProekt (AEP), is intended for verification and validation of the codes used for coupled, full-scale dynamic simulation of VVER-1000 reactors. Figure 1 displays the core layout and CPS rod positions. The rods in the three shaded locations are assumed ejected. This ejected rod configuration is selected in order to produce a non-symmetric three-dimensional flux field in the core during the transient.

In addition to the core layout data of Fig. 1, the benchmark problem also specifies other data pertaining to the geometric, neutronic and thermal-hydraulic characteristics of a VVER-1000 reactor. These include:

- Dimensions of the core, fuel assembly, and fuel and control rods.
- Two-neutron group cross sections and their dependence on burnup, critical boron concentration (CBC), fuel temperature, coolant temperature and density, and control rod state.
- Six delayed-neutron group and 24 decay-heat group data representative of the VVER-1000 core state.
- Albedo. boundary conditions that accurately reflect the neutron leakages radially and axially out of the core.
- Core inlet and outlet temperatures and system pressures and coolant flow rate, which are boundary conditions for the thermal-hydraulic calculations.
- Variation of fuel and cladding thermal conductivity and heat capacity with temperature.

The calculation results requested for this benchmark problem include:

- critical boron concentration,
- power distribution for the initial steady state,
- time evolutions of the total power and power distribution,
- coolant temperature and density distributions,
- and average and maximum fuel temperatures.

B. RBMK Problem Description

The RBMK problem is based on a postulated rod withdrawal event starting from an operating state of unit 3 of the Smolensk nuclear power plant; the control and protection systems are assumed to function as designed. The nominal power of this unit is 2877 MW. The benchmark problem was specified by participants from the Research and Development Institute for Power Engineering (RDIPE) and RRC Kurchatov Institute (KI). Being a full-scale neutronic and thermal-hydraulic model of an RBMK, this problem is computationally challenging because of the large number of channels. (The problem contains 2488 fueled and non-fueled channels. This is a factor of about 10 or more greater than the number of assemblies in typical PWR and VVER reactor cores. In addition, the RBMK core height is about twice or more of these other cores.) Thus, this problem additionally provides a good test case for demonstrating the capability of advanced parallelized coupled codes designed for plant simulations.

The problem specification includes the core loading, the core burnup distribution, detailed descriptions and core locations of the control and protection system control rods and detectors, coolant channel flow rates, fuel and absorber channel dimensions, and material densities and properties. The pressure at the steam drum and the inlet coolant temperature are prescribed as the boundary conditions for the thermal hydraulic calculations. Other thermal-hydraulic data include regional cross-section areas, equivalent diameters, roughness factors and distributed coefficients of local resistance. Group cross section data are also provided for the non-fuel and fuel channels. The cross section dependencies specified for the fueled channels are burnup state, fuel temperature and coolant density. The non-fuel channel cross sections have no feedback dependencies. Delayed neutron data and group velocities are also provided.

This benchmark problem was designed for either the separate or combined testing of the neutronic and thermal-hydraulic modules of coupled codes employed for analyzing RBMK reactors. For this reason, the problem is divided into three stages. The first stage involves a neutronic calculation using specified cross section sets (i.e. no thermal-hydraulic feedback). This problem stage provides a reference state for comparing the accuracy of the neutronic codes. The second stage establishes the initial stationary state for the transient. This stage requires coupled thermal-hydraulic and neutronic iterations for the convergence of the steady state neutron, temperature and density fields. The final stage involves the coupled neutronic/thermal-hydraulic calculation of the rod withdrawal transient. Here it is prescribed that the rod is withdrawn at the speed of 0.40 m/s with the protection systems functional. Very elaborate descriptions of the core power monitoring detectors and CPS rod operations are provided for modeling this stage of the benchmark problem.

The requested benchmark results include:

- multiplication factor, k_{eff},
- radial and axial power density distributions and peaking factors for the initial stationary state,
- transient reactor power evolution,
- time variations of fuel and cladding temperature in certain channels,
- temporal evolution of the inlet flow rates in certain channels, and
- control rod axial positions in certain channels.

III. COUPLED CODES USED BY PARTICIPANTS

The codes employed for analyzing these benchmark problems are those currently in use by participants in the INSC joint project.

The SAS-DIF3DK[1] code was developed at ANL for analyzing a wide range of dynamic situations in thermal reactor systems. The code currently consists of a nodal neutron kinetics model coupled with a core channel thermal-hydraulics model which uses a five-equation treatment for two-phase coolant dynamics. This code, though still under development, has been shown to provide a robust capability for modeling different types of thermal reactor systems.

The TENAR[2] code is designed for simulating coupled reactor dynamics and safety problems. The code uses the KORAT3D finite-element neutronic model for the solution of the multigroup, multidimensional, neutron diffusion approximation and the RATEG module for the solution of the two-velocity, two-temperature, multi-channel thermal-hydraulic approximations.

The RAINBOW[3] code is a systems code designed for simulating dynamic processes in reactor plants of the VVER design. The core and primary and secondary circuits of a VVER reactor plant are modeled by the code. The neutronic module of the code uses the finite difference approximation with large nodes for solving the time-dependent, two-group neutron diffusion equation and the precursor concentration equations. The thermal-hydraulic module for the reactor core models coolant dynamics using the one-dimensional, homogeneous flow, incompressible liquid approximations.

The STEPAN-KOBRA[4] code solves the coupled neutronic and thermal-hydraulic equations for the RBMK reactor. The time-dependent, multidimensional, two-neutron group diffusion equations are solved in the STEPAN module, using the finite difference approximation. Typically, one radial node is used per each RBMK channel and between 16 to 32 axial nodes are used along the axial direction. The KOBRA thermal-hydraulic code uses the non-equilibrium, homogeneous model for determining the coolant condition.

The DINA-SERPENT[4] code system solves the time-dependent, one or two-group neutron diffusion equations using either the finite difference or nodal option in the DINA module. The SERPENT module is a general purpose code that is used for calculating the balance of plant of a water cooled nuclear plant. The code contains different order approximations for the solution of the coolant hydraulic problem.

IV. PRELIMINARY RESULTS

A. Results for VVER-1000 Benchmark Problem

Three different organizations (AEP, VNIIEF and ANL) participating in the INSC joint project contributed coupled-code solutions to this benchmark problem. The ANL participants contributed solutions generated by the SAS-DIF3DK code. The AEP group contributed solutions obtained using the RAINBOW code. The solutions generated using the TENAR code were contributed by the VNIIEF participants. For the purpose of comparing the results, the SAS-DIF3DK solution is selected as the reference.

The critical boron concentrations calculated by the three groups are very similar, with the maximum deviation being about 0.9%. This difference is attributed in part to the fact that the k_{eff} of the critical boron search was not identically unity is some of the calculations. The differences in the calculated assembly powers at the initial state are presented in Fig. 2. The maximum differences between the reference SAS-DIF3DK solution and the TENAR and RAINBOW codes are 6.5% and 26.4%, respectively. These maximum power differences occur in control rod locations. Similarly large differences were observed throughout the transient. These differences are attributed to the different spatial solution schemes used for solving the neutron diffusion equations. The SAS-DIF3DK code uses a nodal method with one radial node per assembly, while the TENAR code uses a finite element scheme with multiple grids per assembly. The RAINBOW code uses the finite difference scheme with one node per assembly, but with an auxiliary correction scheme to account for heterogeneity effects.

Figure 3 shows the time evolution of the core power calculated by the three groups. The initial power rise, which resulted from the ejection of the three CPS rods, was limited and terminated by feedback effects of the heated fuel and coolant. The time evolutions of the maximum and average fuel temperatures are compared in Figs. 4 and 5. The observed differences in the computed transient quantities are due to the differences in the calculated ejected rod worth and reactivity feedbacks, which resulted from the different flux and power distributions calculated by the codes.

B. Results for RBMK Benchmark Problem

Preliminary results for this benchmark problem have been provided by RDIPE (using the DINA-SERPENT code), KI (using the STEPAN-KOBRA) code, VNIIEF (using the TENAR code), and ANL (using SAS-DIF3DK). As a result of the complexity of the benchmark problem, additional discussions, analysis and comparisons of results are required before the final solutions can be reported. The calculated k_{eff} and peaking factors for the first two stages of the benchmark problem are summarized in Tables 1 and 2. The largest deviation in the k_{eff} of the initial state (over 1.3% delta-k/k) was recorded by the TENAR code. This large difference is attributed to possible misinterpretation of the benchmark problem or input error. Additionally, while the STEPAN-KOBRA and DINA-SERPENT codes give very similar values for the axial peaking factors of stages one and two, the differences in the calculated radial peaking factors are about 30% and 13% for stages one and two, respectively. The causes of these differences are being investigated.

In stage three (transient) calculations, the maximum channel power difference between STEPAN-KOBRA and DINA-SERPENT is about 18% and the maximum fuel and cladding temperature differences are about 10% each. These differences further increase as the transient progresses. Figure 6 shows the time evolution of the power in channel 21-25 as calculated by the STEPAN-KOBRA and DINA-SERPENT codes. The reactor power initially increases as a result of the rod withdrawal. This power increase is terminated by the action of CPS rod motion and feedback. The two codes calculated different power versus time profiles for this channel. Similarly large differences are observed in the channel flow distributions and control rod positions. The maximum channel flow rate difference in the final state is about 20%. Figure 7 displays the differences in the time evolutions of some of the CPS (LAR) rods as calculated by the two codes.

These differences are probably due to the fact that the DINA and STEPAN neutronic modules use different spatial approximations for solving the two-group diffusion theory problem. The DINA code uses a nodal approximation, while the STEPAN code uses the finite-difference approach with one radial node per channel. Other differences also arise because of the different approximations used in the thermal-hydraulic modules. The breakdown of errors arising from the coupled-code modules could be studied at a future time by defining an additional problem that de-couples the thermal-hydraulic calculation.

V. FUTURE ACTIVITIES

Future coupled-code benchmark activities at the INSC will stress efforts to resolve the differences that were identified during the preliminary comparison of the benchmark results. It is anticipated that further assessment of the results will suggest needed corrections or improvements in the benchmark analysis (e.g. improved convergence), as well as potential refinements in the code systems themselves. Other pressurized water reactor and RBMK benchmark problems are also planned for joint U.S.-Russia analysis. Additionally, work is underway to provide the descriptions and pertinent data for these benchmark problems on the INSC website, so that other organizations can use the benchmarks to qualify their codes.

VI. SUMMARY AND CONCLUSIONS

Coupled neutronic and thermal-hydraulic benchmark problems have been defined as a basis for intercomparison of codes available at the U.S. and Russian Federation International Nuclear Safety Centers. Preliminary results for two of the benchmark problems defined by the Russian participants were presented. Further analyses of these benchmark problems with U.S. and Russian codes are ongoing, with final comparison of results and assessment of capabilities to be completed in the future.

It is anticipated that the results from the ongoing benchmarking activities will help demonstrate the capabilities and limitations of the existing codes, and guide further enhancement of code capabilities.

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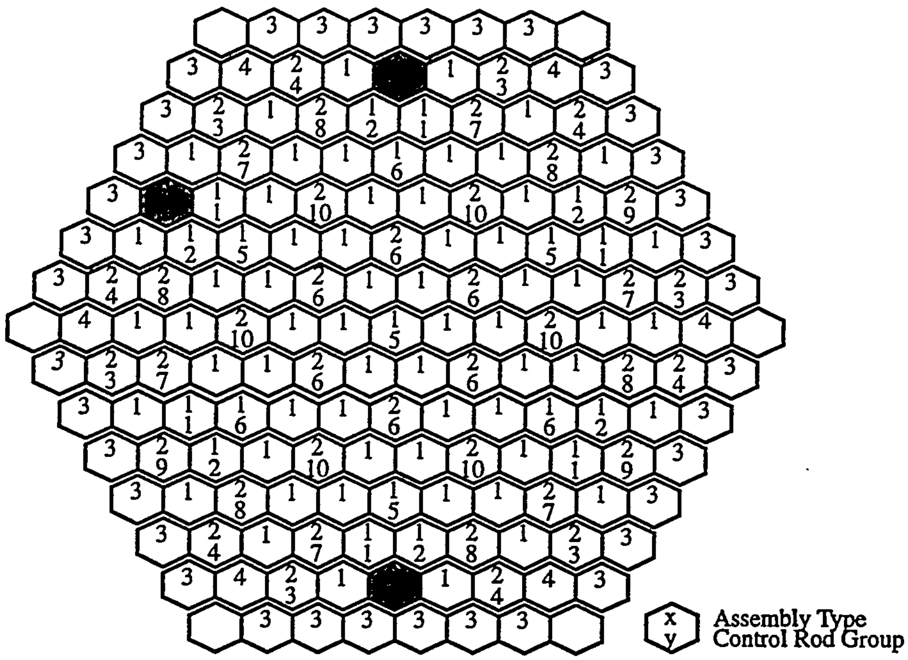
4. L. Podlazov, et al., Private Communications, June 1997.

Table 1: Stage 1 Results (Steady State Without Feedback)

	DINA-SERPENT	STEPAN-KOBRA	. TENAR
K_{eff}	0.9957	1.00004	1.0133
Radial peaking factor (location)	1.95 (35-65)	1.51 (24-55)	1.61 (35-65)
Axial Peaking factor	1.14	1.13	1.15

Table 2: Stage 2 Results (Steady State With Feedback)

	DINA-SERPENT	STEPAN-KOBRA	TENAR
K_{eff}	0.99445	0.99988	1.0163
Radial peaking factor (location)	1.68 (35-65)	1.49 (24-55)	1.37 (35-65)
Axial Peaking factor	1.15	1.12	1.9



Assembly Types
1. 2% enrichment
2. 3% enrichment
3. 3.3% enrichment
4. 3.3% enrichment with
   power profiling

Fig. 1. Loading Pattern and CPS Positions for the VVER-1000 Problem

```
                ------  -3.31  -2.59  -3.17  -3.17  -2.59  -3.31  ------
                ------  -0.60   0.41   2.43   2.43   0.41  -0.60  ------

             -3.31  -1.95  -0.70  -3.13   6.52  -3.25  -0.70  -1.95  -3.31
             -0.60   3.72   2.58  -6.51 -26.44  -6.63.  2.58   3.72  -0.60

          -2.59  -0.70  -1.87   1.69  -0.85  -0.85   1.69  -1.87  -0.70  -2.59 .
           0.41   2.58  -3.66   3.89  -4.59  -4.59   3.89  -3.66   2.58   0.41

       -3.17  -3.13   1.69  -0.44   0.19   1.18   0.19  -0.44   1.69  -3.25  -3.17
        2.43  -6.51   3.89  -2.63  -2.02  -2.90  -2.02  -2.63   3.89  -6.63   2.43

    -3.17   6.52  -0.85   0.19   3.64   0.72   0.72   3.64   0.19  -0.85   6.52  -3.17
     2.43 -26.44  -4.59  -2.02   4.29  -0.09  -0.09   4.29  -2.02  -4.59 -26.44   2.43

 -2.59  -3.25  -0.85   1.18   0.72   0.49   3.78   0.49   0.72   1.18  -0.85  -3.13  -2.59
  0.41  -6.63  -4.59  -2.90  -0.09   0.82   8.74   0.82  -0.09  -2.90  -4.59  -6.51   0.41

-3.31  -0.70   1.69   0.19   0.72   3.78   1.24   1.24   3.78   0.72   0.19   1.69  -0.70  -3.31
-0.60   2.58   3.89  -2.02  -0.09   8.74   1.91   1.91   8.74  -0.09  -2.02   3.89   2.58  -0.60

------  -1.95  -1.87  -0.44   3.64   0.49   1.24   2.30   1.24   0.49   3.64  -0.44  -1.87  -1.95  ------
------   3:72  -3.66  -2.63   4.29   0.82   1.91   0.89   1.91   0.82   4.29  -2.63  -3.66   3.72  ------

-3.31  -0.70   1.69   0.19   0.72   3.78   1.24   1.24   3.78   0.72   0.19   1.69  -0.70  -3.31
-0.60   2.58   3.89  -2.02  -0.09   8.74   1.91   1.91   8.74  -0.09  -2.02   3.89   2.58  -0.60

 -2.59  -3.25  -0.85   1.18   0.72   0.49   3.78   0.49   0.72   1.18  -0.85  -3.13  -2.59
  0.41  -6.63  -4.59  -2.90  -0.09   0.82   8.74   0.82  -0.09  -2.90  -4.59  -6.51   0.41

    -3.17   6.52  -0.85   0.19   3.64   0.72   0.72   3.64   0.19  -0.85   6.52  -3.17
     2.43 -26.44  -4.59  -2.02   4.29  -0.09  -0.09   4.29  -2.02  -4.59 -26.44   2.43

       -3.17  -3.13   1.69  -0.44   0.19   1.18   0.19  -0.44   1.69  -3.25  -3.17
        2.43  -6.51   3.89  -2.63  -2.02  -2.90  -2.02  -2.63   3.89  -6.63   2.43

          -2.59  -0.70  -1.87   1.69  -0.85  -0.85   1.69  -1.87  -0.70  -2.59
           0.41   2.58  -3.66   3.89  -4.59  -4.59   3.89  -3.66   2.58   0.41

             -3.31  -1.95  -0.70  -3.13   6.52  -3.25  -0.70  -1.95  -3.31
             -0.60   3.72   2.58  -6.51 -26.44  -6.63   2.58   3.72  -0.60

                ------  -3.31  -2.59  -3.17  -3.17  -2.59  -3.31  ------     % diff. TEMAR
                ------  -0.60   0.41   2.43   2.43   0.41  -0.60  ------     % diff. RAINBOW
```

Fig. 2. Differences (%) in Calculated Assembly Powers

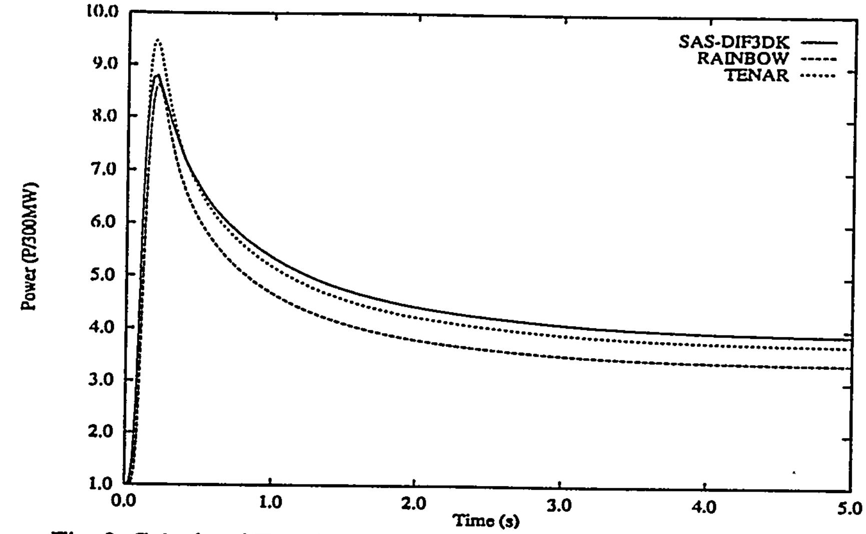


Fig. 3. Calculated Transient Power for the VVER-1000 Benchmark Problem

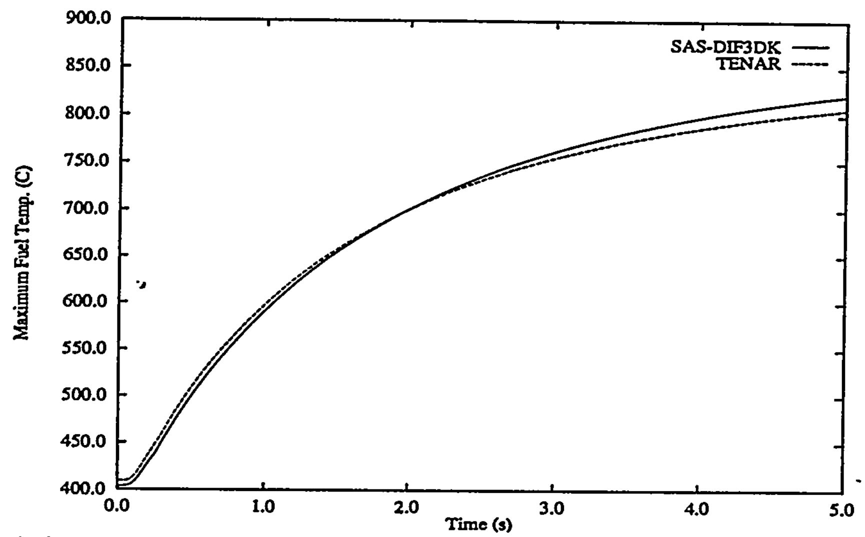


Fig. 4. Calculated Maximum Fuel Temperature for the VVER-1000 Benchmark Problem

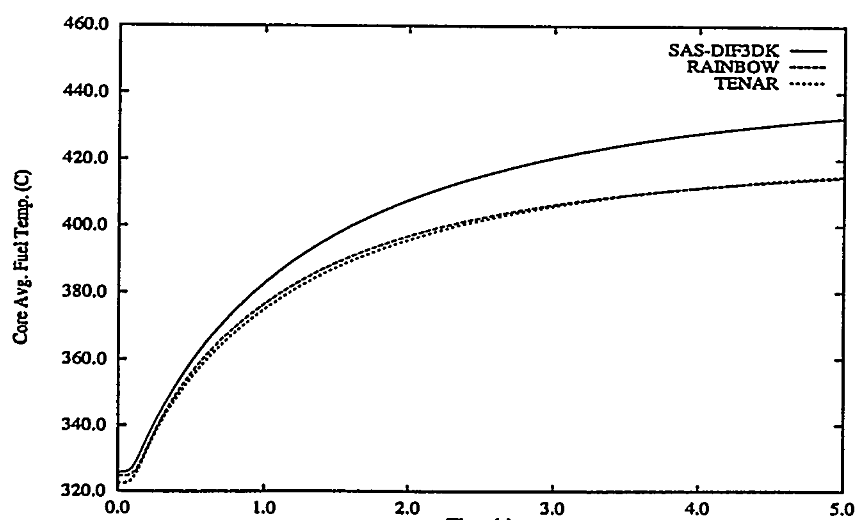
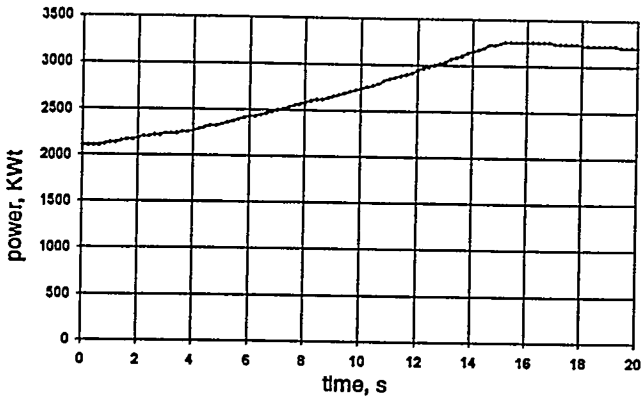
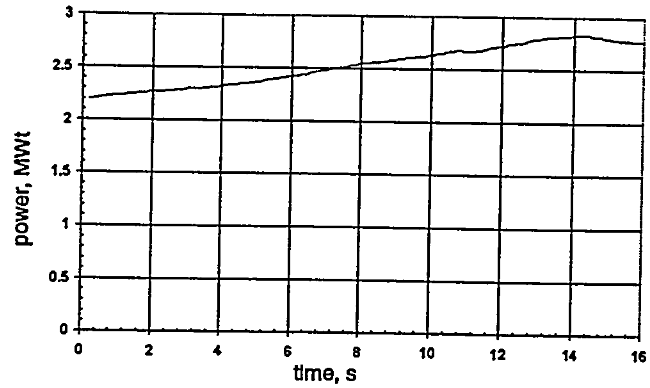


Fig. 5. Calculated Average Fuel Temperature for the VVER-1000 Benchmark Problem

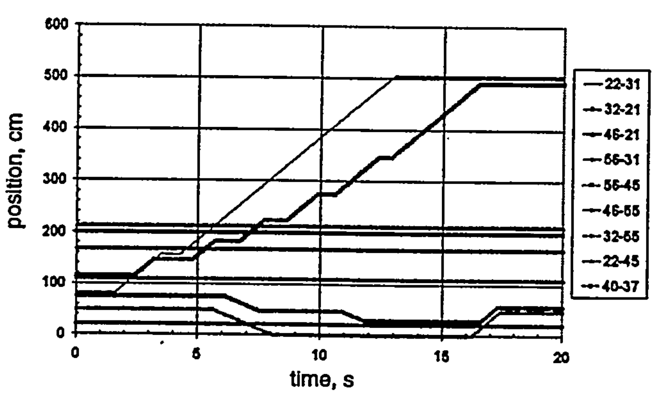


DINA-SERPENT



STEPAN-KOBRA

Fig. 6. Transient Power in Channel 21-25



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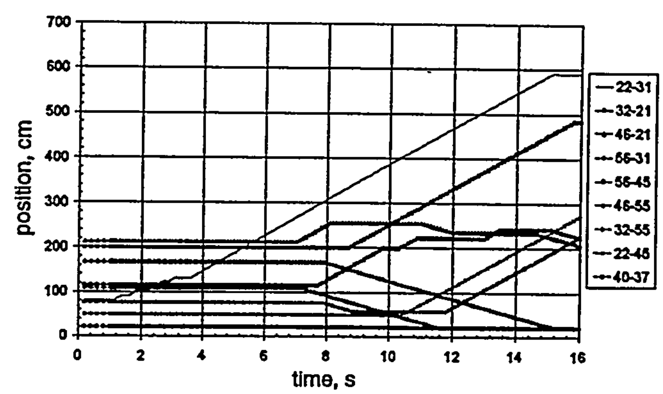

STEPAN-KOBRA

Fig. 7. LAR Control Rod Positions as a Function of Time