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DEVELOPMENT OF THERMOHYDRAULIC MODELS OF CORE ELEMENTS OF "NEUTRON SOURCE"

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This article focuses on the development of independent thermohydraulic models of the core elements of the research nuclear subcritical facility. Its design and construction is currently being implemented by the National Science Center "Kharkov Institute of Physics and Technology". The article describes the process of modeling neutron-generating target and the fuel assembly of "Neutron Source". Moreover, the results of the verification calculations of the developed models are also clarified. These models are unique, because they were made for the first time for the key elements of subcritical facility using CFD code ANSYS CFX.

INTRODUCTION

The National Science Center "Kharkov Institute of Physics and Technology" (NSC KIPT), with support of the Argonne National Laboratory (USA), is constructing the nuclear subcritical facility "Neutron Source Based on the Subcritical Assembly Driven by the Linear Electron Accelerator" (Neutron Source).

According to Article 1 of the Law of Ukraine "On Nuclear Energy Use and Radiation Safety" [1], the Neutron Source is a nuclear facility that requires all measures on its safety assessment and licensing.The licensing activities and commissioning are carried out by SNRIU with the technical support of the SSTC NRS. It is being constructed on the KIPT site.

Conducting activities on licensing of new nuclear installations is impossible without the use of various types of analytical tools for the safety analysis. IAEA is recommended to develop and use of independent expert models for regulatory safety assessment of nuclear installations. These models must be develop with using computational codes differing from those applied by Operator on PSAR and Design stage.

DESIGN FEATURESOF NEUTRON SOURCE

The Neutron Source is an entirely new type of nuclear facilities where the rate of ^{235}U isotope fission in the core is driven by an electron accelerator. The IAEA classifies such facilities as Accelerator Driven Systems (ADS) [2].

The neutron source consists of the following components (Fig. 1) [3, 4]:

- subcritical assembly (SA) on thermal neutrons with shielding;
- neutron-generating target (NGT) to produce primary (external) electrons, located inside the subcritical assembly core;
- linear electron accelerator with a channel for beam transport;
- cold neutron source (CNS);
- facility control panel;
- general engineering systems;
- test neutron channels for nuclear and physical surveys;
- engineering systems for the facility.

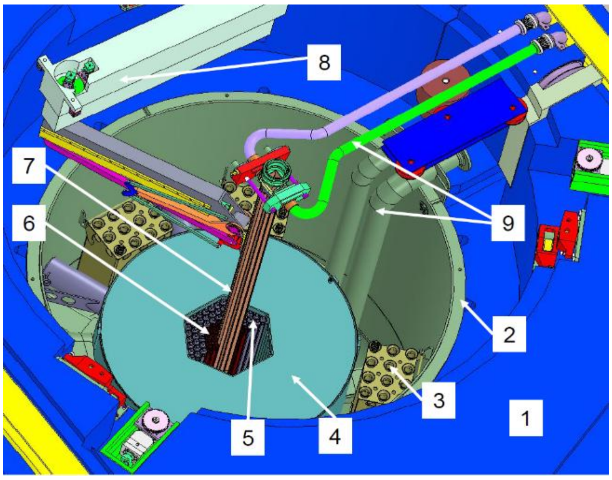


Fig. 1. The layout of "Neutron Source":
1 – biological shielding; 2 – SA tank;
3 – fuel containers; 4 – graphite reflector;
5 – beryllium reflector; 6 – core; 7 – target;
8 – refueling machine; 9 – cooling system

Neutrons are generated through the multiplication of primary neutrons from an external source in the environment of heavy elements (tungsten or natural uranium). The neutron source uses low enriched uranium containing 19.7% of isotope ^{235}U. The composition and geometry of the core ensure that the neutron effective multiplication factor is not higher than the regulated value (0.98). Hence, self-sustained chain fission of ^{235}U cannot occur in the neutron source core. [2].

For removal of heat generated in the process equipment during the nuclear facility operation, several two-loop systems are used for cooling: subcritical assembly (250 kW), target (100 kW), transportation channel (50 kW), and LEA (985 kW). The LEA cooling system includes three subsystems of water cooling. Ventilation cooling towers of secondary cooling system ensure heat dissipation in the environment [3, 4]. Detailed design specifications of Neutron Source presented in [2].

The Neutron Source is designed for research of subcritical assemblies, generation ofneutrons and their use in applied and fundamental research, and for training of experts in the sphere of nuclear energy.

MODELING OBJECTIVES

According to NP 306.2.183-2012 "General Safety Provisions for Nuclear Subcritical Assemblies" [5], the nuclear subcritical assembly safety is ensured by consistent implementation of defense-in-depth strategy based on the use of physical barrier system on the way of radiation and radioactive substance spreading to the environment, and system of technical means and organizational measures on protection of physical barriers and maintaining of their efficiency.

The main purpose for the defense-in-depth strategy implementation is timely detection and elimination of factors leading to abnormal operation, emergencies, and prevention of their progression into accidents, limitation or mitigation of accident consequences. Considering this, it is necessary to define physical barriers, establish and justify safe operation limits.

According to PSAR [4] for the Neutron Source, the system of physical barriers on the way of radioactive substances and radiation spreading from the Neutron Source includes:

1) fuel cladding – confinement of ^{235}U fission products, elimination of fuel contact with the coolant;

2) cladding of the neutron generating target – confinement of products resulting from activation of target plates, elimination of contact between the target plates and the coolant;

3) equipment of primary cooling circuits of the subcritical assembly and the neutron generating target – confinement of the coolant activation products, corrosion of equipment of primary cooling circuits of the subcritical assembly and the neutron generating target and ^{235}U fission products at fuel cladding damage;

4) confining system of leaktight compartments – eliminate release of radioactive substances in the form of aerosols and inert radioactive gases to the experimental hall of the Neutron Source and the environment.

The following values of process parameters are considered the integrity criterion for the mentioned physical barriers according to PSAR [4]:

- maximum temperature of the fuel cladding and the neutron generating target cladding – 485 °C when melting temperature is 660 °C;
- maximum temperature of the coolant in the primary cooling circuit – not higher than 32 °C;
- maximum pressure of the coolant in the primary cooling circuit – not higher than 0.5 MPa.

Non-exceeding of the established criteria should be justified for all operating modes of the Neutron Source, emergencies and accidents. Accident management guides and their justifying documents should be developed based on purposes not to reach or exceed criteria of physical barriers integrity.

As it is shown by the integrity criteria, the following components of the Neutron Source are the key components affecting safety: neutron generating target and the fuel assembly with claddings of aluminum alloy with the lowest melting temperature in the system.

Safety analysis of a subcritical assembly in accordance with the criteria are made for all states: normal operation, violations of normal operation and design basis accidents within the Neutron Source PSAR [4].

Thermal-hydraulic analyzes within "Neutron source" PSAR [4] have been performed by Argonne National Laboratory (USA) using computational fluid dynamics (CFD) code STAR-CD. The results of these analysis show that the equipment corresponds to the design criteria of safety. In order to perform independent testing calculations regulator's technical support organization are working to develop their own, independent models of key "Neurons Source" using other analytical tools.

MODELING OF NGT AND FA

For a long period of time for computational researches in nuclear industry have been used so-called "system codes". As an alternative for "system codes" the international community, led by IAEA are being considered codes of CFD. CFD codes designed to analyze complex dynamic processes in nuclear and thermal plants. The codes usage enables for accurate modeling of hydraulic processes in two- and three-dimensional approximations by solving Reynolds-averaged Navier-Stokes equations. These processes cannot be researched using conventional systemic dimensional codes.

The most use software systems and codes for the analysis of processes in nuclear and thermal plants are: ANSYS (CFX, FLUENT, FLOTRAN), CD-adapco (STAR-CD, STAR-CCM +), Trio_U, Saturne and others. The development of independent thermohydraulic models of neutron generating target (NGT) and fuel assembly (FA) "Neutron source" are being performed using CFD code ANSYS CFX, which is a professional analytical software package, and is designed for a wide range of computational fluid dynamics and gas issues. Scope CFX – modeling of multiphase flows, chemical kinetics, combustion, radiation heat transfer, liquid-structure interaction. The software package can solve the tasks with a movable grid can apply adaptive net condensation.

Modeling of thermal hydraulic models of the neutron generating target and fuel assembly is accepted to be performed in several stages. The modeling process diagram is presented in Fig. 2 and it consists in consistent performance of five stages.

Stage 1 is a preparatory stage to analyze the Neutron Source materials, accumulate data on the neutron generating target and fuel assemblies of the Neutron Source. The stage data sources include the following: Neutron Source Design, Preliminary Safety Analysis Report (PSAR) for the Neutron Source.

Stage 2 covers the construction of geometry of the neutron generating target and fuel assembly cooling circuits with regard to the modeling purposes. Initial data: Stage 1 results.

Stage 3 covers the construction of the finite element grid under the developed geometry in order to perform thermal hydraulic calculations. One should select the needed type of the finite element grid and break down the geometry of the neutron generating target and fuel assemblies using ICEM CFD module.

Stage 4 should define and assign boundary conditions. The initial data for definition of boundary conditions are Stage 1 results. When boundary conditions are assigned, one should perform test calculation of the developed models and compare results with calculations presented in the Design and PSAR for the Neutron Source.

Stage 5 covers selection of scenarios to analyze the Neutron Source transients, modification of models and calculation regarding safety analyses presented in PSAR for the Neutron Source, and new calculations. *Collection of initial data on equipment*. The stage data sources include the following: Neutron Source Design [3], Preliminary Safety Analysis Report (PSAR) [4] for the Neutron Sourceand equipment documentation. The neutron-generating target is designed to receive primary neutrons controlling the subcritical assembly operation. The primary neutrons are generated at bombardment of a target from tungsten or natural uranium by electrons with energy of 100 MeV and average beam power of 100 kW. Reaction of the target bombardment by electrons is accompanied by release of large amount of heat in the target plates. Structurally, the neutron generating target consists of the following basic components: vacuum window for electron beam entrance; head designed for vacuum connection with electron line and process handling operations on reloading of the target, whose structure ensures reliable fixation of the transfer cask gripper; connector for supply and removal of the coolant designed for connection of the target to the primary circuit for target cooling; target socket designed for fixing of target plates and arrangement of the coolant motion; target plates designed for generation of primary neutrons in the subcritical assembly core (7 (W) or 11 (U) pcs); helium chamber; fixing finger designed for fixing and orientation of the target in the core [3, 4].

The subcritical assembly core consists of 37 fuel assemblies VVR-M2 in case of uranium target and 38 in case of tungsten target, and is located in the subcritical assembly tank.

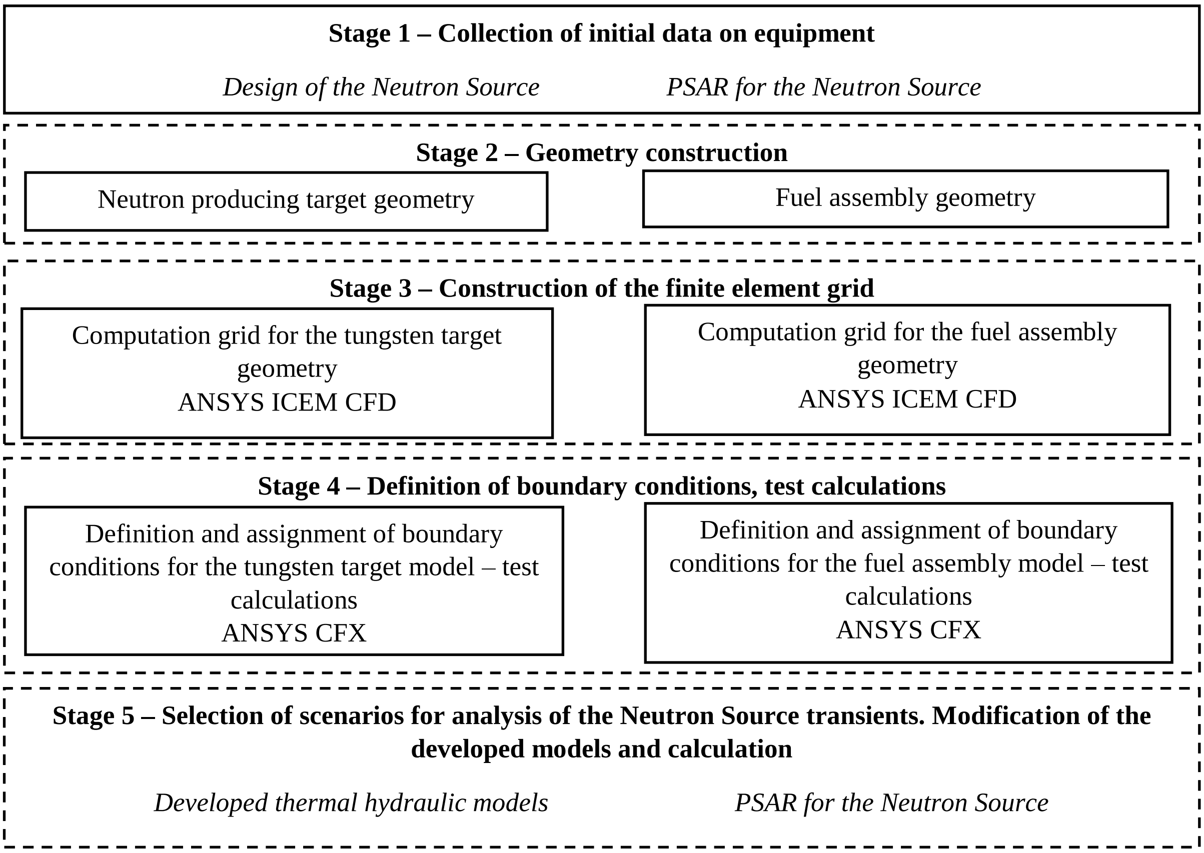


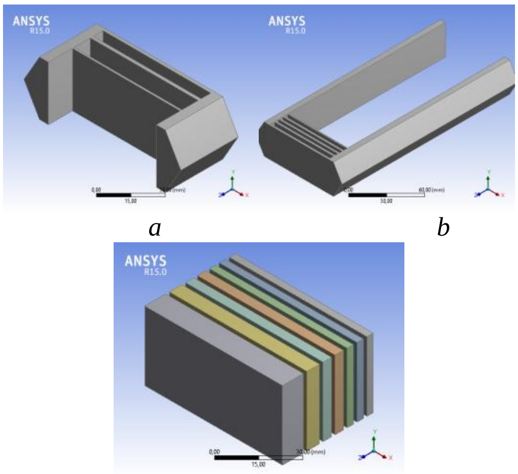
Fig. 2. Modeling of "Neutron Source" key elements

The subcritical assembly core uses nuclear fuel in fuel assemblies of VVR-M2 type. The fuel assembly consists of three elements (fuel elements) of a tubular form: two coaxial tubes of a cylindrical form and one outer tube that is hexagonal. Nuclear fissile material in the form of UO_2 dioxide (^{235}U enrichment – 19.7%) is dispersedly distributed in the aluminum matrix of 1 mm thick. In order to prevent ingress of uranium fission products to the coolant, surfaces of fuel elements are covered by protective layer of aluminum of 0.5 mm thick [3, 4].

Geometry construction. Since during modeling we are interested exclusively in hydrodynamic characteristics of the neutron generating target and FA also the temperature characteristics of surface of the plates and elements of FA. The modeling of the calculated part of the tungsten neutron generating target consists in modeling: flow part of the tungsten neutron generating target; set of the neutron generating target plates (7 pcs); simplified part of the separator.

Considering the neutron generating target geometry (2 symmetric channels), we take decision on calculation of the half of the neutron generating target, i. e. one channel with further assignment of boundary conditions for symmetry and distribution of results for the full-scale neutron generating target. Taking into account peculiarities of the neutron generating target, separator that is a device aimed at separation of flow for cooling fluid is modeled in a simplified way at the stage of assigning boundary conditions, with the help of adiabatic wall functions.

The geometry of the three parts of the computational model are presented in Fig. 3.

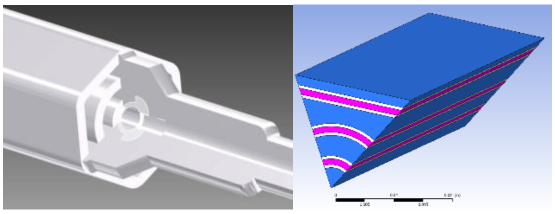


a *b*

c

Fig. 3. The geometry of the computational domain NGT (a, b – flow part of NGT; c – target plates (7 pcs))

The modeling of FA is performed regarding: 3 fuel assemblies (two coaxial tubes of a cylindrical form and one external tube of a hexagonal form); fuel element cladding with thickness compliant with the design; cooling channels of fuel assemblies; half the thickness of the coolant around the fuel assembly, i. e. cell, because otherwise we will obtain incorrect heat load on the external fuel element.The developed computer model is a 1/6 part of the fuel assembly. The simplification was performed based on the symmetry of the fuel assembly in order to reduce calculation time. Fig. 4 presents a sectional view of the Neutron Source fuel assembly (a) and 1/6 part of the developed model (b).



a *b*

Fig. 4. The geometry of the computational domain of FA

The geometry of calculated part is modeled by means of integrated software module ANSYS Design Modeler. The stage of assigning the calculation region geometry is followed by creation of named selections to facilitate further set of boundary conditions at next stages. The logics of named selections creation is as follows: after creation of the required geometry, it is necessary to understand, which boundary conditions are needed for calculation. Then, according to these boundary conditions, the named selections should be created. These named selections are assigned in ANSYS Design Modeler for the calculated region of the target plates: volumetric section of each of seven target plates, surfaces for assignment of interfaces plate-liquid. For the calculated region of fluid, named selections were assigned at the stage of computation grid breakdown.

Construction of the finite element grid. ANSYS ICEM CFD package is used for construction of the computational grid for the neutron generating target model. The decision on the choice structured hexahedral mesh for NGT and FA models were made. The solution of using hexahedral grids can improve the quality of items to reduce the dimension of the computational models and to improve the convergence of tasks. The process of constructing a structured hexahedral mesh finite element is an iterative process in order to achieve a high quality mesh. The process of meshing model in ANSYS ICEM CFD package is presented as a Fig. 5. Mesh for NGT and FA model, isolated for each region, the decision was made define boundary condition with respect to each section.

As a results of the above stages, the gridded computation model of the neutron generating target and FA were constructed with the grid quality of not lower than 0.5 for neutron generating target model and 0.9 for FA model.

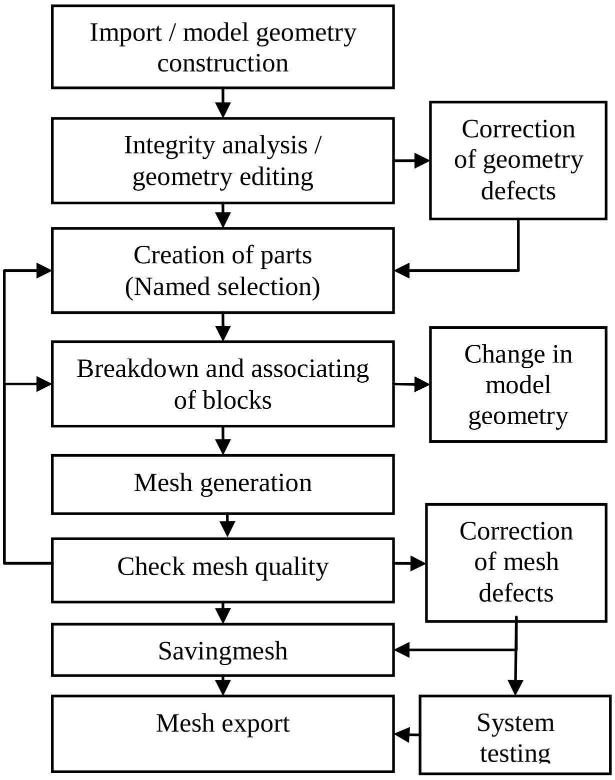


Fig. 5. The process of meshing

Definition of boundary conditions. The preliminary testing of the model required assignment of boundary conditions that meet the NGT and FA operation under normal conditions, so as to calculate the steady state. Safe operation limits of the coolant and elements were used as acceptance criteria for thermal hydraulic calculations.

Boundary conditions for simulation were adopted on the basis of "Neutron Source" Design and PSAR for normal operation. As the value of the energy of tungsten plates (NGT) made average volumetric energy release tungsten target on the basis "Neutron Source" Design and PSAR-1.5E09 W/m^3. The calculation NGT model for steady state is shown in Fig. 6.

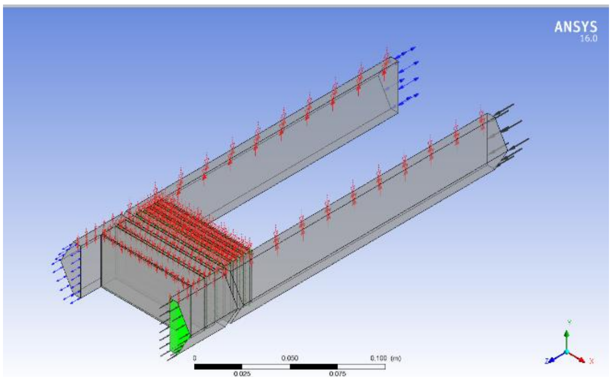


Fig. 6. Model of the tungsten NGT of the Neutron Source with assigned boundary conditions

For FA model, isolated Fluid-Volumes are modeled by separate Fluid-Domains, as in case of modeling by one Fluid-Domain with one inlet and outlet at the unbound grids one can receive incorrect results. For each Fluid-Domain, its mass flow rate is assigned, thus the total is equal to flow rate through 1/6 part of the fuel assembly. The relative ratio between flow rates for different Domains was obtained by additional calculation with one common input (in terms of continuity of the geometry and grid), which is a standard practice. In order to assign specific power, maximally loaded fuel assembly is used. Volumetric energy release is defined based on neutron calculation performed within PSAR. Distribution of specific power is performed along the height of all fuel elements.

RESULTS OF TEST CALCULATIONS

Comparative analysis of the results for the NGT model with PSAR results present on Tabl. 1. Maximum temperature of the target plates reaches 160 °C that is significantly lower than melting temperature of the plate cladding and target casing.

Table 1

Comparative analysis of the results for the NGT model

Parameter	PSAR model	The developed model
Temperature of secondary line outflow, °C	29.8	28.4
Average temperature at outlet, °C	30.4	29.1

Figs. 7 and 8 shows the results of calculating the NGT model for steady state.

The calculation results show compliance of the model developed under this task with the PSAR model. Insignificant difference in results can be caused by the made assumptions during calculations and peculiarities of the solver mathematical models.

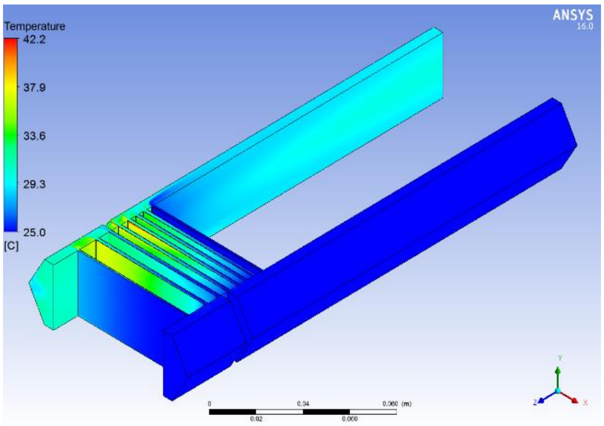


Fig. 7. Results of calculation of the neutron generating target model – coolant temperature distribution

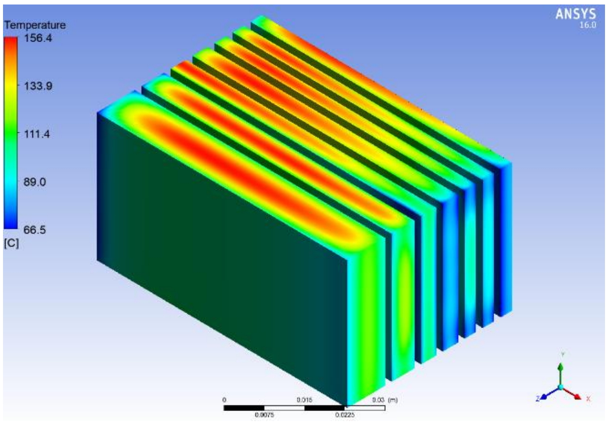


Fig. 8. Results of calculation of the neutron generating target model (steady state) – the temperature of the target plates

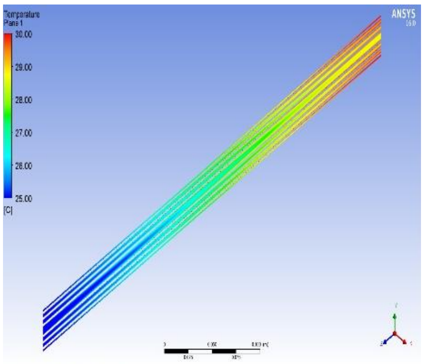
As a result of test calculations, correction of the model and Solver setting, the working model of the "Neutron Source" FA was obtained. Comparative analysis of the results for the FA model with PSAR results present on Tabl. 2.

Table 2

Comparative analysis of the results for the FA model

Parameter	PSAR model	The developed model
A coolant temperature at inlet, °C	25	25
The coolant temperature at outlet from the fuel assembly, °C	29.5	30.5
Maximum fuel temperature, °C	45.0	41.0

Fig. 9 shows the results of calculating the FA model for steady state.



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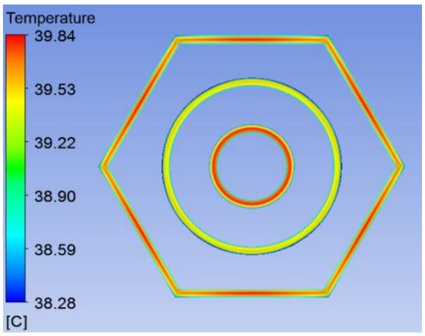

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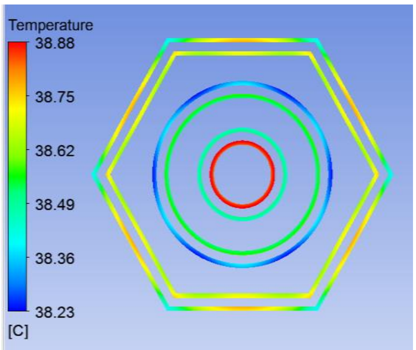

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Fig. 9. Temperature of fuel and fuel cladding in section at fuel assembly outlet

The calculation results show compliance of the model developed within this task and the PSAR model. Insignificant difference in results can be caused by assumptions made during calculations and peculiarities of mathematic models of the Solver.

CONCLUSIONS

At licensing of new nuclear facilities regulators and technical support organizations need to develop independent modes of key nuclear facility elements and perform calculations for calibration safety studies using non-identical analytical tools.

Currently SSTC NRS are developed local thermal hydraulic models of safety significant components of the Neutron Source using CFD of ANSYS code:

- thermal hydraulic model of the tungsten neutron generating target of the Neutron Source;
- thermal hydraulic model of the fuel assembly VVR-M2 of the Neutron Source.

Performed test and validation calculations for the developed models of the neutron generating target and fuel assembly of the Neutron Source showed good convergence, and their compliance with identic calculation performed within PSAR.

The developed thermohydraulic models of the neutron generating target and fuel assembly of the Neutron Source are models independent of operating organization, can be modified and used during state review of nuclear and radiation safety of documents on the Neutron Source in order to perform check calculations.

Despite the positive outcomes, there is scope for improvement models in terms of: more accurate accounting of the structural features of the elements, as well as the working conditions of subcritical assembly. In addition, the developed model can be modified to analyze the behavior of elements in transients mode and accidents.

Work on the modification and use of models is already under way, the respective materials will be published in the following publications.

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РАЗРАБОТКА ТЕПЛОГИДРАВЛИЧЕСКИХ МОДЕЛЕЙ ЭЛЕМЕНТОВ АКТИВНОЙ ЗОНЫ ЯДЕРНОЙ ПОДКРИТИЧЕСКОЙ УСТАНОВКИ «ИСТОЧНИК НЕЙТРОНОВ»

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Описана разработка независимых теплогидравлических моделей элементов активной зоны исследовательской ядерной подкритической установки (ЯПУ). Ее проектирование и сооружение в настоящее время осуществляется Национальным научным центром «Харьковский физико-технический институт». В работе представлено описание процесса моделирования нейтронообразующей мишени и тепловыделяющей сборки ЯПУ «Источник нейтронов». Освещены также результаты тестово-верификационных расчетов разработанных моделей, которые являются уникальными, так как выполнены впервые для критических элементов подкритической установки с использованием CFD-кода ANSYS CFX.

РОЗРОБКА ТЕПЛОГІДРАВЛІЧНИХ МОДЕЛЕЙ ЕЛЕМЕНТІВ АКТИВНОЇ ЗОНИ ЯДЕРНОЇ ПІДКРИТИЧНОЇ УСТАНОВКИ «ДЖЕРЕЛО НЕЙТРОНІВ»

О.В. Кухоцький, А.В. Носовський, О.М. Дибач

Описано розробка незалежних теплогідравлічних моделей елементів активної зони дослідницької ядерної підкритичної установки (ЯПУ). Її проектування і спорудження в даний час здійснюється Національним науковим центром «Харківський фізико-технічний інститут». У роботі представлено опис процессу моделювання нейтроноутворюючої мішені і тепловиділяючої збірки ЯПУ «Джерело нейтронів». Висвітлено також результати тестово-верифікаційних розрахунків розроблених моделей, які є унікальними, так як виконані вперше для критичних елементів підкритичної установки з використанням CFD-коду ANSYS CFX.