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Thermal Hydraulics Verification Safety Studies to Support the Licensing of Neutron Source Accelerator Driven System

Keywords:
thermal-hydraylics,
safety studies,
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The State Scientific and Technical Center for Nuclear and Radiation Safety performs the scientific and technical support of Ukrainian regulatory authority (State Nuclear Regulatory Inspectorate of Ukraine) in licensing activities of the Neutron Source Based on a Subcritical Assembly Driven by a Linear Electron Accelerator (hereinafter designated as the Neutron Source). In the Neutron Source's licensing, the independent models of key nuclear facility elements were developed and calculations for verification safety studies using non-identical analytical tools were performed.
Aspects related to the development of independent (regulatory) thermal hydraulics models of the core elements of the Neutron Source are discussed in the paper. The process of modeling neutron-generating target and the fuel assembly of Neutron Source are described in the article. Moreover, the results of the verification safety studies are also presented.

Introduction

An experimental nuclear research facility driven by accelerator, called the Neutron Source is under commissioning in the National Scientific Center "Kharkov Institute of Physics and Technology" (NSC KIPT), Ukraine as an international collaborative project of NSC KIPT and Argonne National Laboratory (ANL), USA. Licensing of new research facility is impossible without use of various types of analytical tools for the verification safety analysis. Moreover, the International Atomic Energy Agency (IAEA) recommends to develop and use independent expert models for regulatory safety assessment of nuclear installations. The independent models of key nuclear facility elements have been developed. The verification safety studies using non-identical analytical tools have been performed. The models of key nuclear facility elements were developed in the directions of neutron-physics calculations, thermal hydraulics safety analysis and radiation protection calculations. Aspects related to thermal hydraulics models and safety studies are discussed in the paper.

Model aims and objectives

According to Ukrainian regulation "General Safety Provisions for Nuclear Subcritical Assemblies" [1], 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 substances spreading to the environment, and system of technical means and organizational measures on protection of physical barriers and maintaining their efficiency. The main purpose of 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. Based on Neutron Source design [2] the system of physical barriers on the way of radioactive substances and radiation spreading from the Neutron Source includes:

fuel cladding — confinement of fuel fission products, elimination of fuel contact with the coolant;

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;

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 fuel fission products at fuel cladding damage;

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 integrity criteria for the mentioned physical barriers were defined according to Preliminary Safety Analysis Report (PSAR) [3] and presented below:

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.

As it is shown by the integrity criteria, the following components of the Neutron Source are the key components affecting safety: neutron generating target (NGT) and the fuel assembly (FA) with claddings of aluminum alloy with the lowest melting temperature in the system. In order to perform independent verification calculations SSTC NRS on behalf of State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) has developed their own, independent models of key elements of the Neuron Source using other analytical tools and has been performed verification safety studies to support the licensing process.

Modeling approaches and tools

The thermal hydraulics verification safety studies are focused on verification of the Neutron Source safety criteria that are relevant to the integrity of the physical barriers. In view of integrity criterion for the mentioned physical barriers independent (regulatory) models for NGT and the FA with claddings of aluminum alloy with the lowest melting temperature in the system were developed with further performing verification safety studies. The development of independent thermohydraulic models of key Neutron Source elements is performed using Computational Fluid Dynamics (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.

The verification safety studies were performed in a few stages (Fig. 1): (1) steady state calculation aimed on models verification and validation and (2) studies of transient safety cases and finally results comparison.

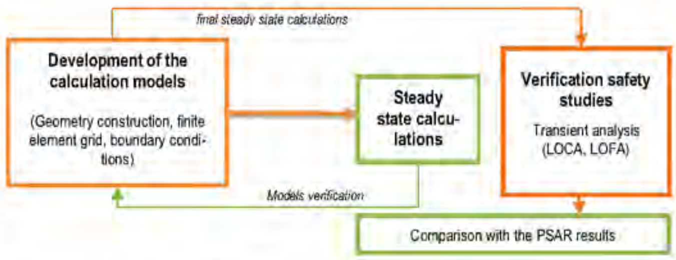

Fig. 1. Modeling of Neutron Source key elements

Development of the calculation models. The first stage of CFD calculation is pre-processing: geometry computer-aided design (CAD), meshing and boundary conditions. In case of thermal-hydraulics safety studies are interested exclusively in hydrodynamic characteristics of the NGT and FA, the temperature characteristics of the surface of the NGT plates and elements of FA.

Structurally, the NGT consists of the following basic components: vacuum window, head, 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, target plates (7 (W) or 11 (U) pcs); helium chamber, fixing finger [3, 4]. The geometry of the calculated part of the tungsten NGT consists in modeling: flow part of the tungsten NGT; set of the NGT plates (7 pcs); simplified part of the separator. Considering the NGT's geometry (2 symmetric channels), decision on calculation of the half of the NGT was made, i. e. one channel with further assignment of boundary conditions for symmetry and distribution of results to the full-scale NGT. The geometry of the three parts of the computational model are presented in Fig. 2 [2].

The Neutron Source core consists of 35–38 fuel assemblies (depending on the target type) VVR-M2, and

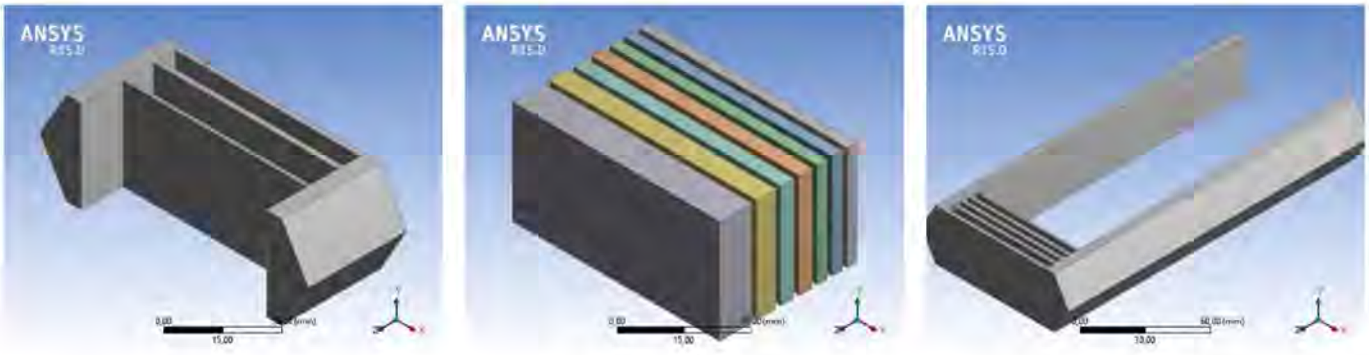

Fig. 2. The geometry of the computational domain NGT

is located in the subcritical assembly (SA) tank. 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. The FA model is a cell of the averaged core, which includes 120-degree sectors of three adjacent fuel assemblies in accordance with the triangular grid for placing (Fig. 3). Such a configuration in the general form of a hexagon multimaterial cell takes into account the entire spectrum of the possible leakage and coolant heat exchange mode not only within the limits of one cell, but also in the space between the fuel assemblies, which is especially important for the formation of circuit for cooling between the assemblies. Taking into account the similarity of fuel assembly geometrical profile and the similarity of thermal hydraulic processes in the azimuthal direction of the calculation cell (energy release of identical fuel rods, azimuthal repetition of cooling channels profile and geometrical and material configuration of fuel and claddings), it is enough to consider sectors of assemblies and not assemblies in general. For the circulation circuit (fluid volumes of the model) on the boundaries of the sectors forming the similarity conditions, the boundary symmetry conditions are set, which actually reflect the "mirror" behavior of the flow in the imaginary volume (that is not simulated in this task) opposite to the boundary of specified condition (the tensor of velocity flow components is zero). For a problem with natural convection of a fluid, this assumption is also acceptable, since the formation of the circuit of natural circulation in the channel of constant section is related only to the change in the axial component of the velocity tensor (in full simulation of the entire area, flows at the boundary of similarity would integrally go to zero) [4, 5].

ANSYS ICEM CFD package was used for construction of the computational finite element grid for the NGT and FA model. The selection of the calculation grid is determined by the conditions of fluid flow in the near-boundary layer and possible computer resources (the principle of correct sufficiency of spatial discretization in relation

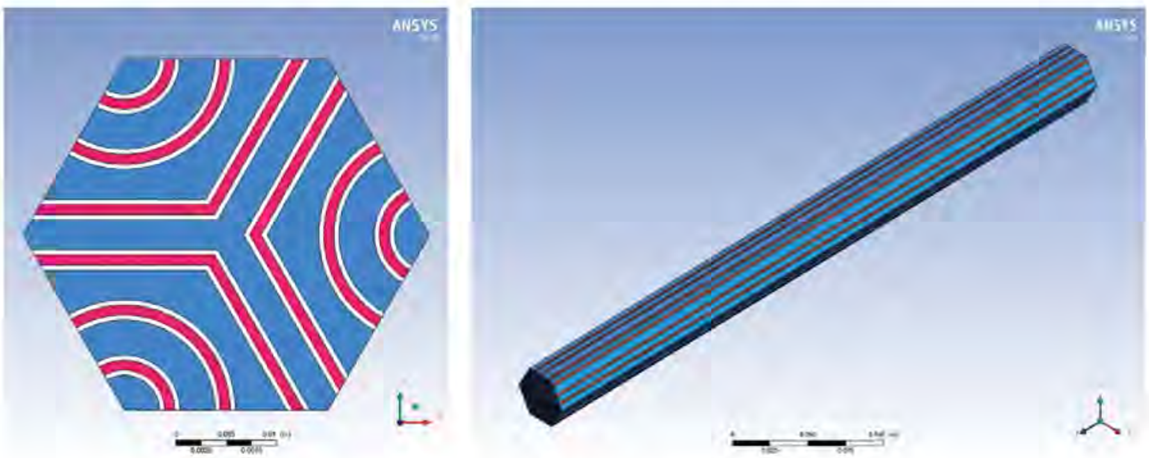

Fig. 3. The geometry of the computational domain FA

to the set task). In general, the calculation grid was developed for the conditions of natural circulation (task for the transient analysis). Therefore, for the stationary state with water coolant, the same grid is used, which makes it possible to correctly transfer initial conditions for the solution of the task related to the emergency transient with the instantaneous flow of water coolant (though for the stationary task, the grid with lower degree of fluid volume discretization is permissible). The calculation grid takes into account the formation of the profile of the fluid velocity from the wall of cladding (viscous sublayer) to the flow middle; therefore a gradual increase in the radial thickness of elements along the normal from the wall is observed, which correlates with the nature of velocity gradient change for the both laminar and turbulent modes of fluid flow. The selection of the number and thickness of the grid elements in the near-boundary layer, in general, is controlled by the so-called Y+ parameter, which shows the change in the fluid velocity gradient along the normal to the wall in relation to the element thickness in the same direction. In the ideal situation, such a parameter shall be 1.0, which is almost impossible to achieve for non-trivial tasks. In the engineering practice for solving applied tasks, the more so with the use of one-parameter or two-parameter models of turbulence, it is quite enough to be guided by the maximum value Y+ less than 50 (or less than 100 in the local spatial-temporal zone of the calculation task). For stationary task, Y+ max is within 2.8 ÷ 9.0. Parameter Y+ can be estimated analytically at the stage of calculation grid design, but such a task is quite voluminous. At the stage of forming preliminary models, one can perform calculation experiments with own boundary conditions and check the possible maximum value Y+ parameter, which is the resultant value for CFD packages.

For preliminary checking of the models, boundary conditions were assigned that are compliant with NGT/FA operation under normal conditions and the steady-state was calculated. Volumetric energy release of NGT and fuel are defined based on neutron calculation performed within PSAR. The average volumetric energy release from tungsten target is 1.5E09 W/m^3. Distribution of specific power is performed along the height of all fuel elements of maximum loaded FA and presented below: SM — stationary mode (steady state analysis), RH — residual heat release at 60 s (transient analysis).

Safe operation limits of the Neutron Source were used as acceptance criteria for thermal hydraulic parameters of the coolant, the NGT and FA. For the stationary state of fluid flow, it is sufficient to use RNG k-ε turbulence model, which rather accurately and stably simulates the turbulent regime of forced fluid flow in a simple geometry channel (classic k-ε turbulence model is badly suited for the regimes with a high Reynolds number, although it is somewhat less demanding to the grid).

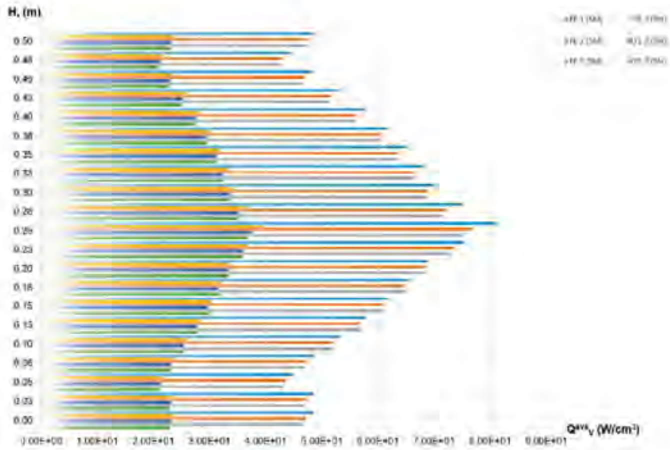

Fig. 4. FA power distribution (max-loaded FA)

Results of steady state calculation. As a result of test calculations, correction of the model and Solver setting, the working model of the Neutron Source NGT was obtained. The calculations resulted in the following:

temperature of secondary line outflow is 28.4 °C (PSAR — 29.8 °C);

average temperature at outlet is 29.1 °C (PSAR — 30.4 °C);

maximum temperature of the target plates reaches 160 °C that is significantly lower than melting temperature of the plate cladding and target casing [4].

The main calculation results are graphically presented in Fig. 6.

As a result of test calculations, correction of the model and Solver setting, the working model of the Neutron Source FA was obtained. The calculations resulted in the following:

at a coolant temperature of 25.0 °C at inlet to the fuel assembly with maximum fuel temperature, the coolant temperature at outlet from the fuel assembly is 29.5 °C (PSAR — 30.32 °C) which ensures large margin to coolant boiling;

maximum fuel temperature is 45.11 °C, which shows large margin to safety criterion 660 °C.

The main calculation results are graphically presented in Fig. 7, 8.

The calculation results show compliance within safety criteria. Therefore, the developed model can be used for local (for the fuel assembly) analysis of emer-

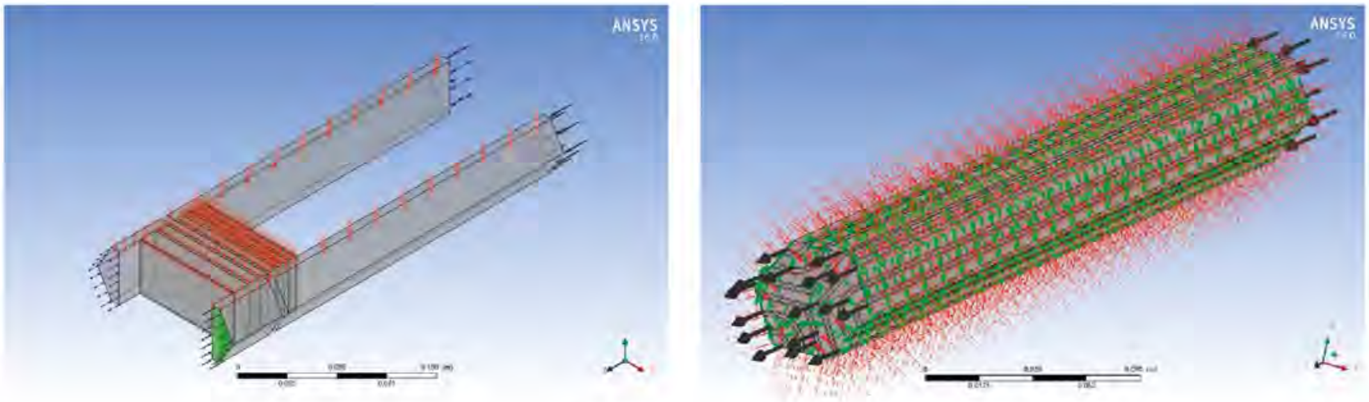

Fig. 5. NGT and FA models

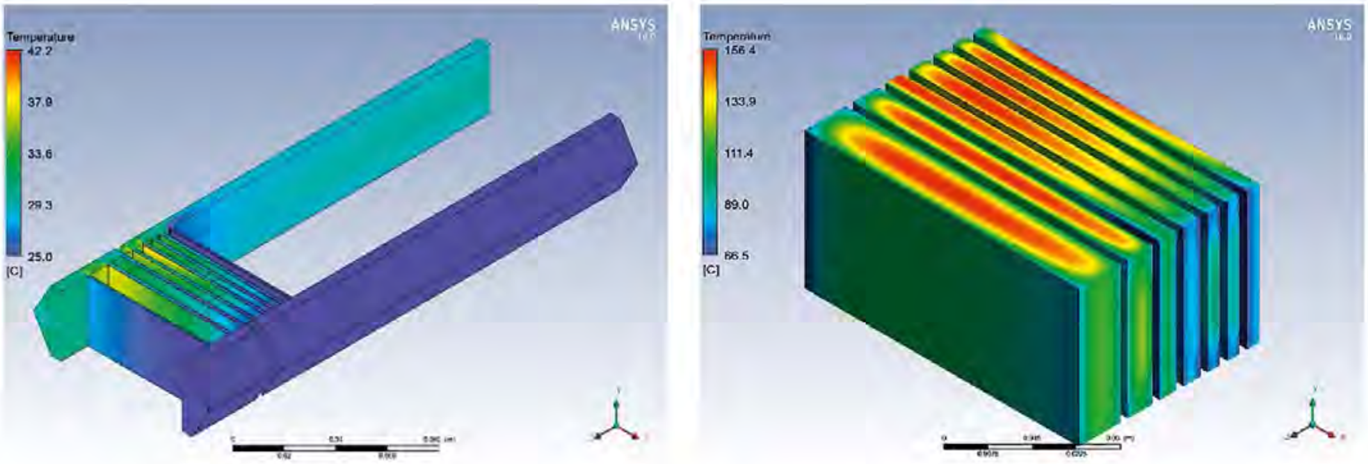

Fig. 6. NGT's coolant temperature distribution (left) and the temperature of the NGT's plates

gency processes and transients of the Neutron Source. Insignificant difference in results may be caused by assumptions made during calculations and peculiarities of mathematic models of the Solver and showed more conservative cases in PSAR studies [5].

Verification safety studies — transient analysis for FA

Scenario with coolant loss is one of the most interesting for analyses in the aspect of thermal hydraulic analyses. Task considers subcritical assembly coolant loss and assumes that the circulation circuit for the cooling of NGT is operable. After loss of coolant, according to the operation of I&C system of protection and interlocks, the linear electron accelerator is turned off and the Neutron Source is transferred to the shutdown state. Heat release from fuel is at the level of residual heat. The calculation scenario for defining the impact of loss-of-coolant accident (LOCA) on fuel temperature envisages that the water in the cooling system is pumped out instantaneously and replaced by air with simultaneous tripping of the accelerator. The analysis of the LOCA transient with the partial loss-of-coolant (coolant level (water) at the midpoint of the core or below) is the subject of future analysis.

One also envisages natural circulating air loop that passes up the fuel assembly to the upper part of subcritical assembly volume, where it is cooled. Then, the air moves down along the reflector rods, as presented in Fig. 9.

The process of cooling during the LOCA is based on the presence of a column of cold air along the edge of the core and the ability of the air to remove released heat.

The task with transient related to the formation of natural air circulation circuit ideally used k-w turbulence model that exactly simulates the entire spectrum of processes in a thin transverse layer, which is especially typical for the natural circulation regime, whose driving force is fluid

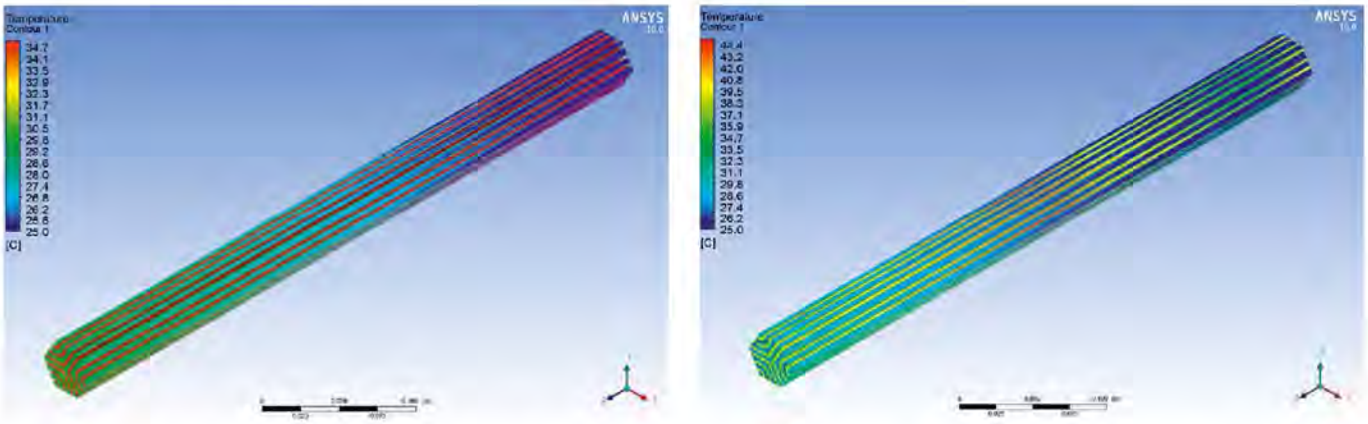

Fig. 7. Coolant temperature along longitudinal section of the fuel assembly (left) and temperatures along longitudinal section of the fuel assembly (right)

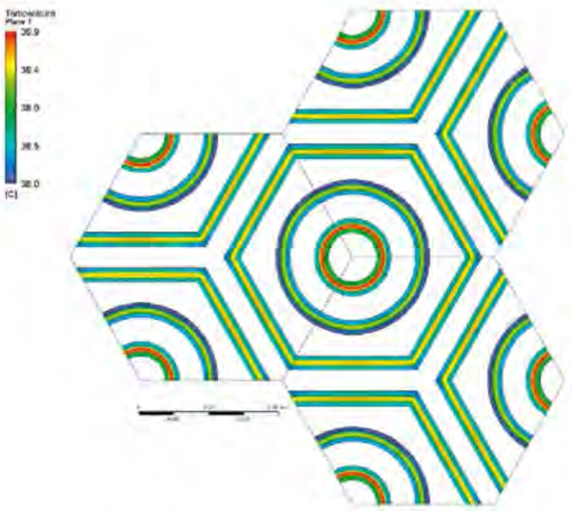

Fig. 8. Temperature of fuel and fuel cladding in section at fuel assembly outlet

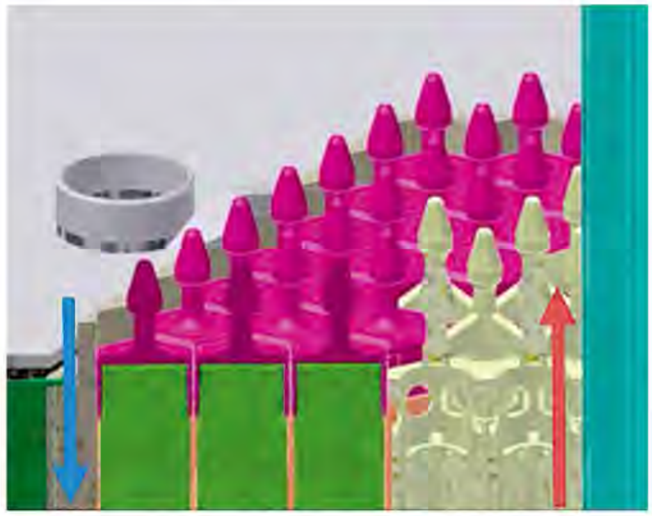

Fig. 9. Natural circulating air loop

density gradient in the near boundary layer. However, such a turbulence model requires rather a detailed discretization of near boundary layers of the grid, which significantly increases the required computer resources and time of calculation. At the stage of preliminary calculations it was established that turbulent stable mode of natural convection was formed, taking into account simple geometrical configuration of model vertical channels (without so-called separation points of flow fluctuations), which is possible with a sufficient degree of accuracy in using RNG k-ε turbulence model (renormalization of groups in RNG k-ε model in comparison with the classic k-ε model allows more accurate modelling of near-wall layers even on coarse grids, which requires somewhat more time for calculation). In general, use of k-ω turbulence model is critical only in case of very narrow channels, in which parallel near-boundary layers create impact on each other, complex geometry with the presence of stochastic separation points, low Reynolds flow modes (which do not apply to our task). Only temperature of stationary model materials is used as the initial conditions for the transient analysis. It is conservatively assumed that the air from the zero second is already warmed to the temperature of water before flow, and the air velocity at the beginning of the emergency process is zero, that is the natural circulation circuit only gradually begins to form (in maximum value of initial fuel energy release). The capacity of the assembly is established at the level of residual heat release of the maximum loaded fuel, calculated on the basis of data of PSAR (RH in Fig. 4). The used power levels are conservative, and need to be clarified.

As a result of the calculation, the following results for modelling transient with LOCA were received:

at the initial stage of transient, after loss of coolant, the natural air circulation is established gradually (Fig. 10), natural convection is not enough to remove residual heat of fuel, which leads to the heating of the entire fuel assembly and increase of the temperature gradient between fuel and coolant (air) (Fig. 10), at the same time, heating of the area intensifies natural circulation of air, in the future, taking into account the gradual decrease of fuel rods power to the level of residual heat;

there is stabilization of parameters and the transition to a quasi-stationary state of residual heat removal due to the natural convection of the coolant.

The heating fuel assembly does not lead to fuel cladding integrity damage and does not cause the release of radioactive substances from the fuel matrix. The maximum temperature of the fuel reaches 278.6 °C, and the cladding — 277.7 °C, which does not exceed the safety criteria. On the other side, in uncovering core elements, there is an intensification of radioactive substances release due to the deflation of radioactive aerosols. Aerosols are removed from the vessel volume by means of air intake device of the special ventilation, release of radioactive substances beyond the limits of biological protection of the subcritical assembly does not occur. The presented results of the calculation are conservative, taking into account made assumptions. The LOCA analysis should be detailed taking into to account 1) the actual flow leakage of flow rate 2) justified level of residual heat release. The effect of reducing water level in the subcritical assembly vessel on nuclear safety is considered in frame of neutron-physic calculations. It is shown that the accident with drying of subcritical assembly vessel does not lead to the increase of multiplying properties of the medium and does not reduce nuclear safety of the Neutron Source.

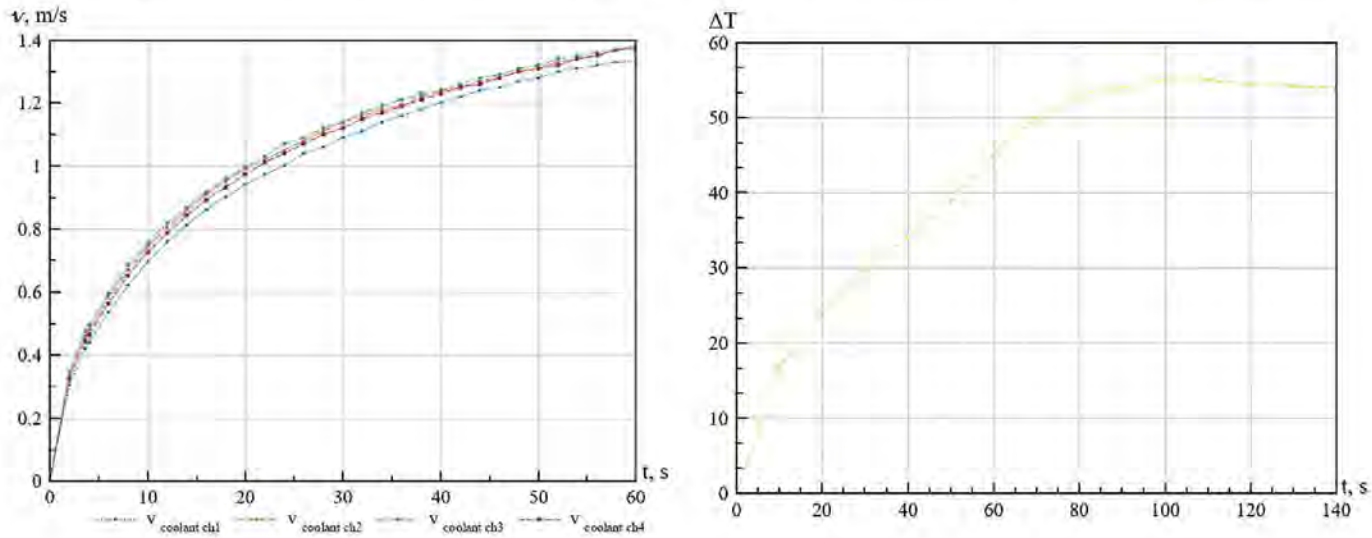

Fig.10. Coolant velocity in channels (transient) and coolant/fuel temperature gradient

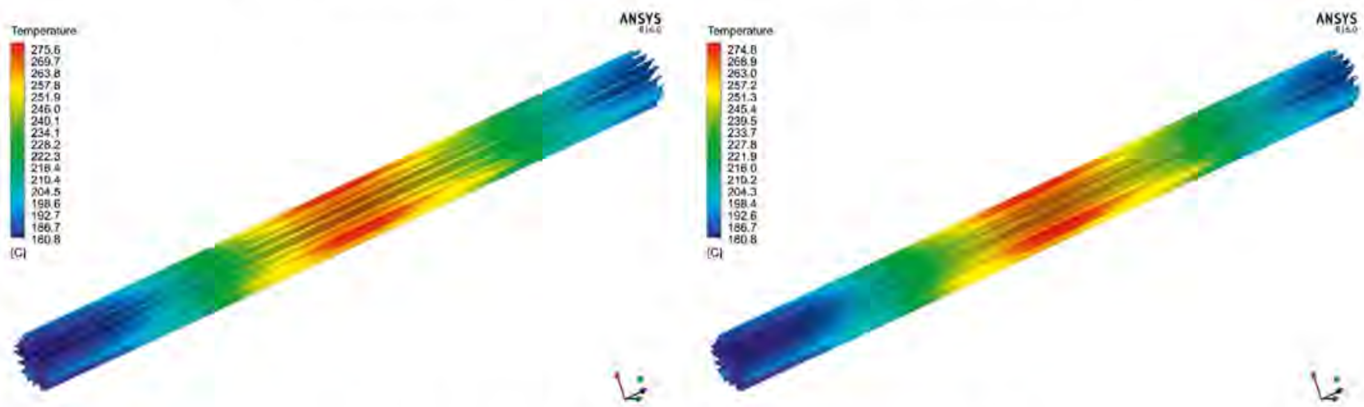

Fig. 11. The maximum temperature of the fuel and cladding

Conclusions

Licensing of the new research facility is impossible without using various types of analytical tools for verification safety studies. The independent thermo-hydraulic models of nuclear facility key elements developed and performed verification safety studies using non-identical analytical tools. Performed test and validation calculations for the developed models of the Neutron Source key elements showed good convergence, and their compliance with identic calculation performed within PSAR. Also, the scenario with coolant loss (LOCA) is one of the most interesting for analyses in the aspect of thermal hydraulic analyses. LOCA was analyzed in these verification safety studies. The results show that the heating of the fuel assembly does not lead to fuel cladding integrity damage and does not cause the release of radioactive substances from the fuel matrix, in case of residual heat. Taking into account result of safety studies the duplication (by an operator) of automatic accelerator shutdown actions should be predicted on Accident Management Guidelines for LOCA scenarios.

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Теплогідравлічні перевірочні розрахунки для підтримки ліцензування ядерної підкритичної установки «Джерело нейтронів»

Дослідницька ядерна установка «Джерело нейтронів, яка базується на підкритичній збірці, що керується лінійним прискорювачем електронів» (ЯПУ «Джерело нейтронів») наразі перебуває на стадії введення в експлуатацію в Національному науковому центрі «Харківський фізико-технічний інститут» (ННЦ ХФТІ) як міжнародний проект ННЦ ХФТІ і Аргонської національної лабораторії (ANL), США. Відповідно до Міжнародної класифікації Міжнародного агентства з атомної енергії (МАГАТЕ) такі установки належать до Accelerator Driven Systems — систем, що керуються прискорювачем. Державний науково-технічний центр з ядерної та радіаційної безпеки здійснює наукову та технічну підтримку українського регулюючого органу (Державна інспекція ядерного регулювання України) при ліцензуванні ЯПУ «Джерело нейтронів». Здійснення діяльності з ліцензування нової ядерної установки неможливо без використання різного роду аналітичних інструментів для виконання перевірочних розрахунків аналізу безпеки установки в цілому і її ключових елементів. Тому при ліцензуванні «Джерела нейтронів» були розроблені незалежні теплогідравлічні моделі основних елементів ядерної установки та проведені перевірочні розрахунки та дослідження безпеки з використанням неідентичних аналітичних інструментів. Перевірочні теплогідравлічні дослідження орієнтовані на перевірку неперевищення критеріїв безпеки «Джерела нейтронів». У статті розглянуто аспекти, пов'язані з розробкою незалежних теплогідравлічних моделей ключових елементів установки — нейтрон-утворюючої мішені та паливної збірки. Розробка незалежних теплогідравлічних моделей виконувалась з використанням коду розрахункової гідродинаміки (CFD) ANSYS CFX. Описано процес розробки моделей нейтрон-утворюючої мішені та паливної збірки, представлено результати тестових розрахунків для стаціонарного стану, а також виконано аналіз перехідного процесу — аварійного сценарію з течею теплоносія бака підкритичної збірки (LOCA).

Ключові слова: теплогідравлічний, дослідження безпеки, ліцензійний процес, джерело нейтронів, прискорювач.

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Теплогидравлические проверочные расчеты для поддержки лицензирования ядерной подкритической установки «Источник нейтронов»

Государственный научно-технический центр по ядерной и радиационной безопасности осуществляет научную и техническую поддержку украинского регулирующего органа (Государственная инспекция ядерного регулирования Украины) при лицензировании ядерной подкритической установки «Источник нейтронов, основанный на подкритической сборке, управляемой линейным ускорителем электронов» (далее — «Источник нейтронов»). При лицензировании «Источника нейтронов» были разработаны независимые теплогидравлические модели основных элементов ядерной установки и проведены проверочные расчеты и исследования безопасности с использованием неидентичных аналитических инструментов. В этой статье рассмотрены аспекты, связанные с разработкой независимых теплогидравлических моделей основных элементов, описан процесс моделирования нейтрон-образующей мишени и топливной сборки и представлены результаты проверочных расчетов.

Ключевые слова: теплогидравлический, исследования безопасности, лицензионный процесс, источник нейтронов, ускоритель.

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