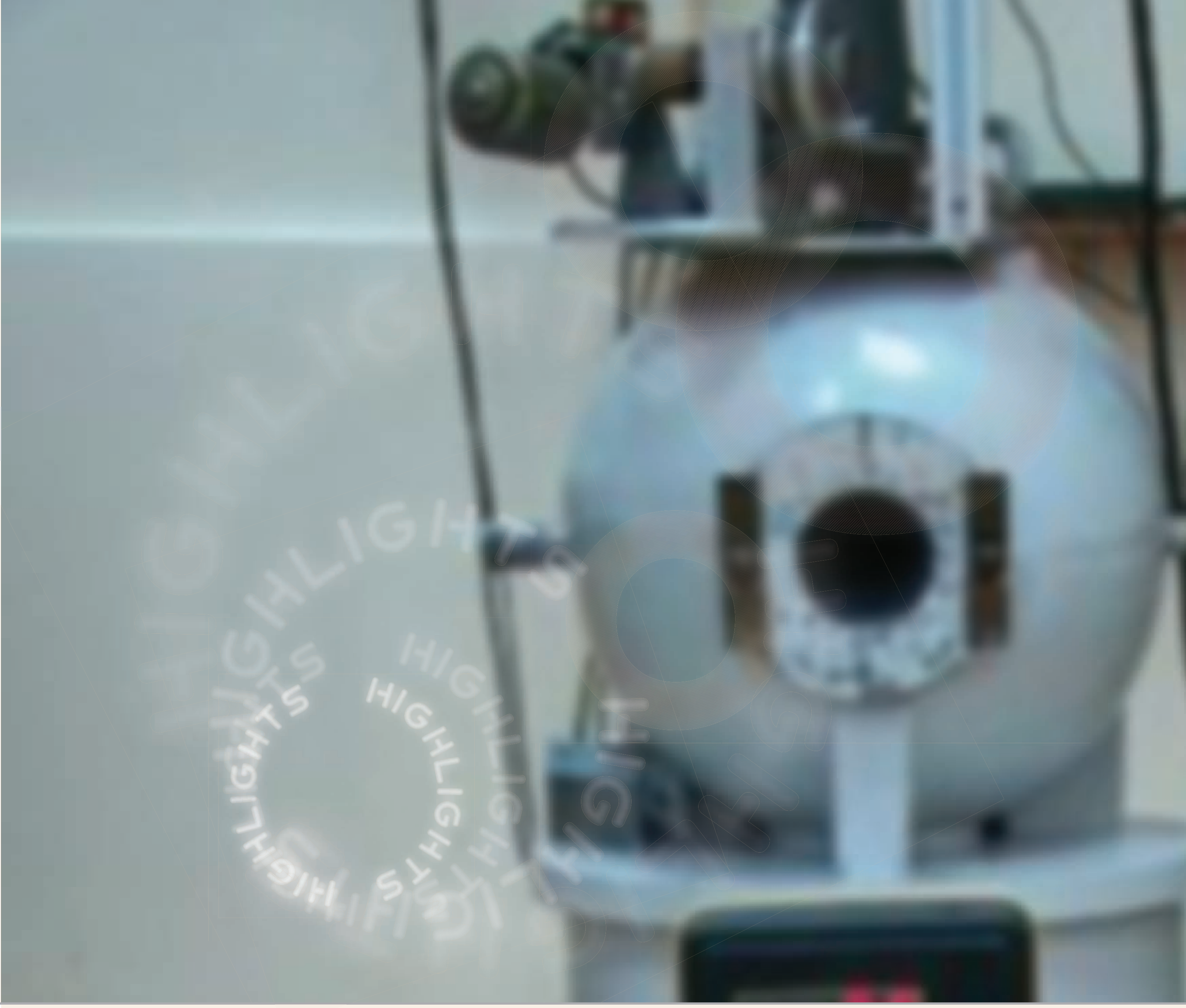

DEPARTMENT OF NUCLEAR ENERGY

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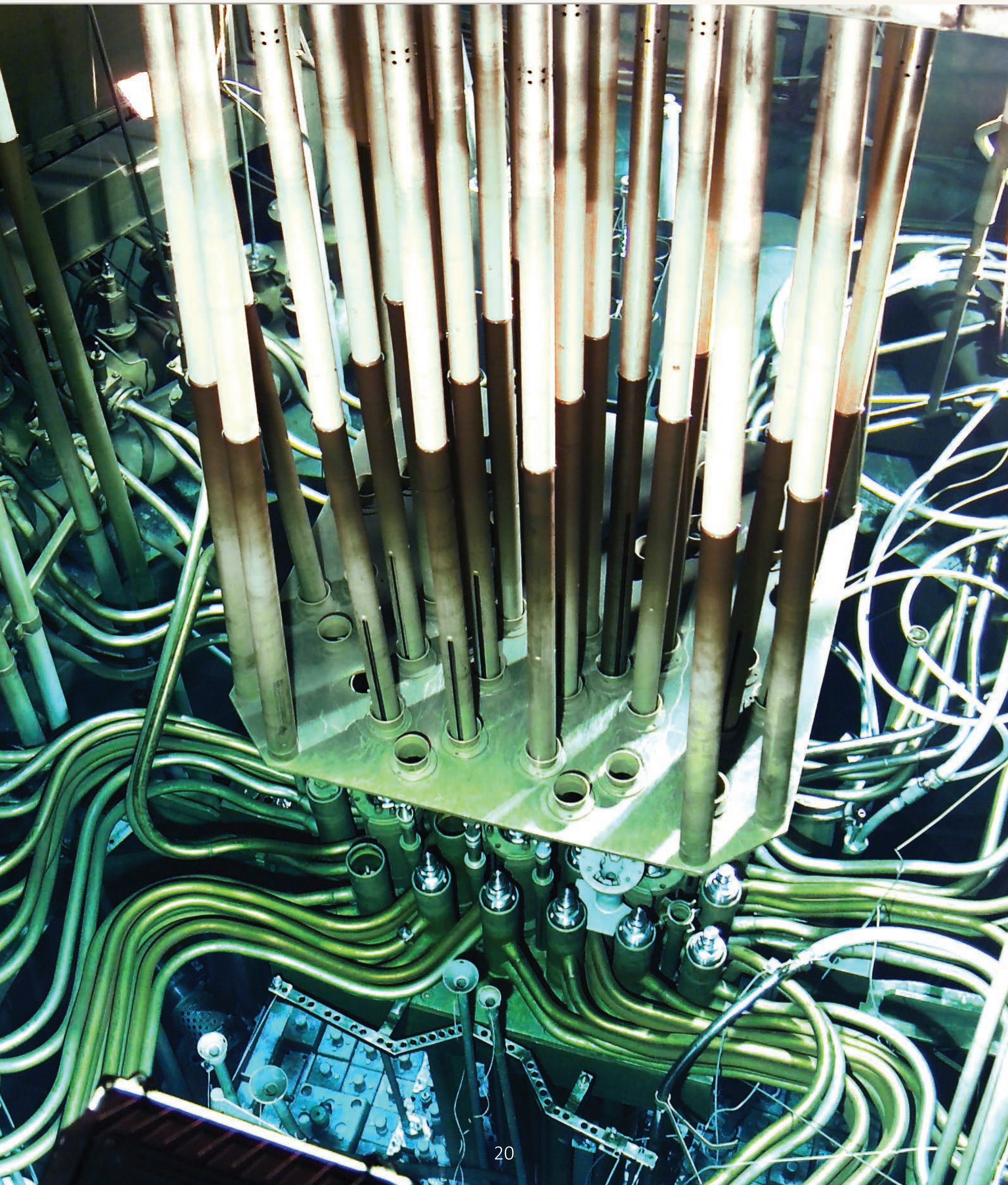

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The MARIA research reactor operated for 4282 hours in 2014 at power ranging from 18 to 24 MW.

The reactor was mainly used for irradiation of materials used in radioisotope production for RC (Radioisotopes Centre) "Polatom" and Mallinckrodt Pharmaceuticals company and for performing physical research at the outlet of the reactor horizontal beam ports. The supply of Mo-99 for nuclear medicine was limited due to unplanned downtime of the Processing Plant in Petten, Netherlands in the January – April period.

The reactor conversion was completed in August when the last high enriched burnt fuel element was replaced with a low enriched element.

Trial irradiation of MR-6/485 type low enriched fuel made in accordance with Russian technology was completed, whereas the study of spent fuel elements was conducted in October. The positive outcome of the Russian type fuel tests allowed us to acquire a potential second supplier of fuel for the MARIA reactor.

Within the framework of the Global Threat Reduction Initiative program, another batch (44 pcs) of high enriched spent MR type fuel was returned to the Russian Federation. In September the potential of the reactor was increased after installing a thermal neutron to 14 MeV fast neutron converter.

Within the framework of cooperation between NCNR and CEA-JHR heat generation measurements and calibration of in-core measuring devices in research reactors were made.

The Laboratory for Dosimetry Measurements conducted the following measurements at Świerk:
- contamination of the environment,
- threat to individual employes,
- background of gamma radiation.

The most important achievements of the laboratory for Mixed Radiation are as follows:

1. Construction and tests of a multisignal ionization chamber demonstrator for neutron dosimetry over a wide range of the energy spectrum;

2. Developing the concept of dose planning and restoration methods using PET and SPECT, computer simulations and Monte Carlo calculations based on a phantom with microspheres for SIRT therapy examination.

In the Nuclear Power Engineering division a number of projects associated with safety analyses for the MARIA reactor, power reactors and the interaction of nuclear facilities with the environment were carried out.

The most important of them are:

1. Burnup calculations of three types of MARIA fuel elements and poisoning in beryllium blocks using the APOLLO and MCNP codes;

2. Implementation of the SCALE code package for neutron-physics analysis;

3. Modelling of HTR core with various distributions of fuel and dummy elements.

THERMAL HYDRAULIC SAFETY MARGIN DETERMINATION FOR THE IN-CORE NEUTRON STREAM CONVERTER DESIGNED FOR THE MARIA RESEARCH REACTOR - FUEL ASSEMBLY STUDY

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The BNCT converter is a unique type of densely packed fuel assembly (triangular lattice with a pitch-to-diameter p/d=1.2) in a tight housing with a relatively small flow cross-section. According to a conservative-based study [1], multiplying the power of all rods by little more than 2.5, the surface temperature of the cladding should not exceed 85˚C (358 K) in a steady state and 136˚C (409 K) in a transient when a short-term peak of power in the core would affect the power of the BNCT converter. This analytical approach used numerous simplifications and further computations using other means, i.e. CFD were still required. The whole process of modelling was conducted using ANSYS Workbench 15.0. Geometry and grid discreti-zation are depicted in Fig. 1.

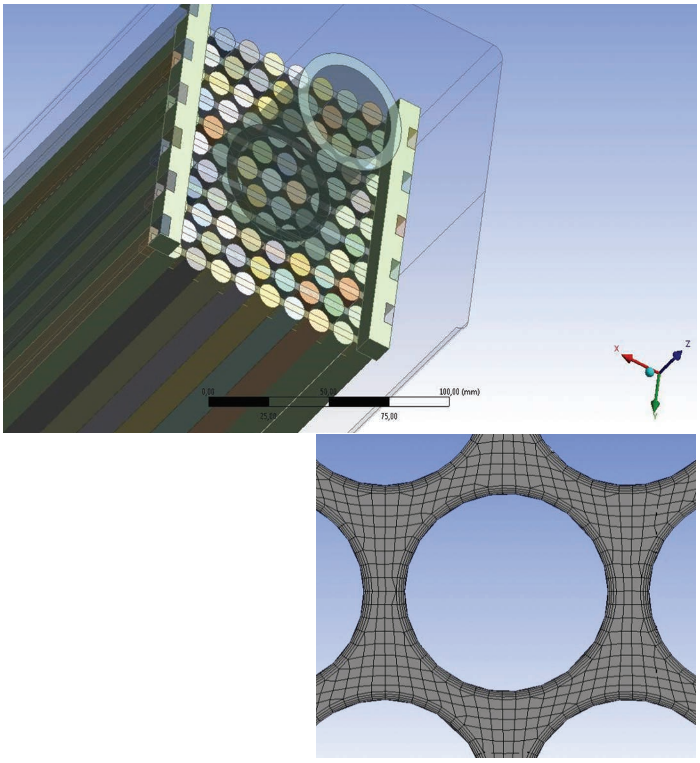

Fig. 1. Geometry and discretization grid.

During the simulation unexpected strong temperature oscillations occurred (Fig. 2.) in the outer part of the converter as depicted in Fig. 3. The first impression was that this was due to numerical error. The timestep was decreased but because of the mesh resolution could not be dropped below 0.2 ms. This significantly reduced the oscillations (see Fig. 2.), but they were still persistent. Hence, further mesh refinement is recommended. As this effect did not appear in the inner part of the assembly, one may assume that the oscillations had physical origins and that due to insufficient mesh resolution they became amplified. It was found that the difference in velocities between the inner-core (between the rods) and the outer-core space was almost a factor of 2. This gradient, depicted in Fig. 3., shows a strong relation between the velocity and temperature oscillations. Taking this phenomenon into account as an additional negative factor in any further considerations makes the analysis more conservative, is good practice where safety is concerned.

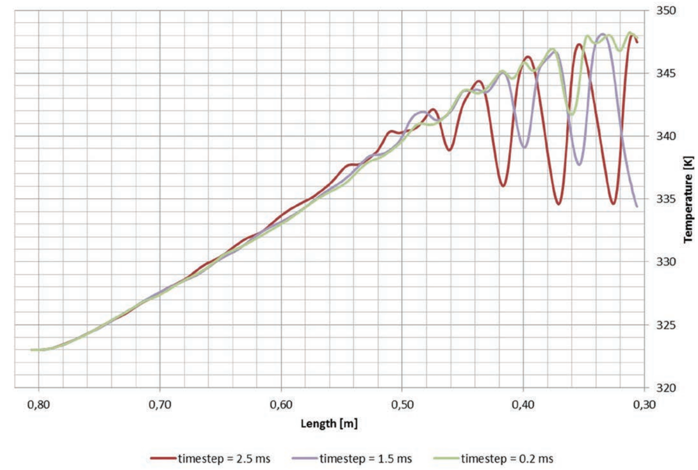

Fig. 2. Coolant temperature oscillations measured along the flow (along the line depicted as a red dot in Fig. 3.).

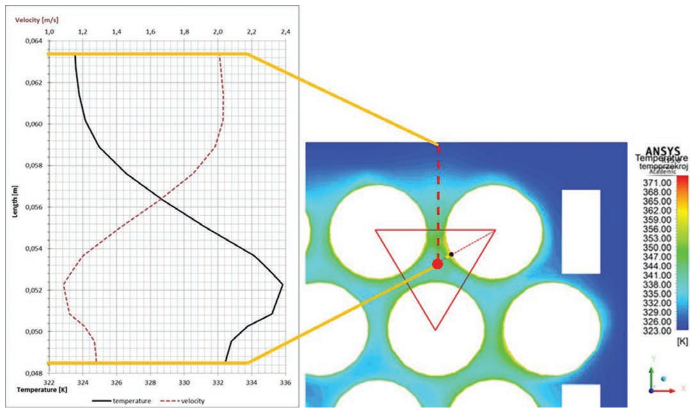

Fig. 3. Triangle of the hottest rods – cross-section at the level of highest coolant temperature.

Numerous cases were modelled and simulated. During a Loss Of Flow Transient (LOFT) simulation, the coolant flow through the converter channel was safely reduced to 60% of its nominal value. As a result the maximum temperature of the water did not exceed a saturation level of 110˚C even if the power of the converter where 2.5 times higher than in the normal regime (conservative presumption). This allows as to conclude that safe operation of the BNCT converter could be assured despite the oscillations, but these should be avoided, so some geometrical improvements would be advised.

Reference

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ANALYSES OF TURBULENT FLOW IN A TIGHT LATTICE BARE ROD-BUNDLE

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Accurate prediction of the flow distribution inside fuel rod bundles is required for both design and safe operation of innovative as well as conventional nuclear systems. Numerical modelling, in the form of Computational Fluid Dynamics (CFD) represents an essential tool which can allow the limitations of traditional applied subchannel analysis codes to be overcome.

Axial coolant flow inside bare fuel rod bundles presents complex behaviour. Hydraulic experiments [1] and a thermal-hydraulics experiment [2] have revealed the existence of flow oscillations or unsteadiness in a tightly spaced rodbundle with pitch-to-diameter ratio (P/D) of 1.1 and 1.06 respectively. The flow oscillations can cause flow-induced vibration in a rod-bundle. It has been revealed by the thermal hydraulics experiment that these flow oscillations result in temperature oscillations. These temperature oscillations, in turn, can induce thermal shock or thermal fatigue damage of the rod cladding.

In general, RANS (Reynolds Averaged Navier-Stokes) approaches are used to analyse the flow and mixing processes within fuel assemblies [3]. However, RANS modelling is associated with numerical errors derived from the averaged Navier-Stokes equations. Moreover, taking into account the lack of an experimental database (due to the fact that an experimental investigation is often impossible or too expensive to be performed) or additional reference data, e.g. based on fully detailed high fidelity DNS (Direct Numerical Simulations), the reliability of RANS approaches in modelling flow and heat transfer in fuel assemblies is often questionable and, if possible, strictly limited. Although DNS gives high fidelity results, since the modelling errors are kept to a minimum, its application is frequently limited by the high computational cost associated with the required spatial resolution. Thus, this limitation often restricts the applicability of this methodology to relatively simple flow geometries. It is important to underline the fact that in order to obtain the required resolution for a computationally affordable high fidelity DNS of sub-channel flow, the RANS approach has to be developed first. Subsequently the obtained results from supporting RANS analyses will form the basis for further numerical investigation where a detailed reference simulation (DNS) of the coolant fluid flow through a sub-channel configuration will be pursued.

To establish the RANS approach, first the geometry of a well-documented case was selected [4]. Next, a sensitivity study of the flow dynamics to the selected domain length was performed. A careful selection of the domain extension is highly desirable for a high fidelity DNS analysis. Afterwards, a set of unsteady RANS simulations were performed, in which the experimental mass flow rate was systematically scaled down. Then, the relevant turbulent scales, e.g. Kolmogorov scales, η, and Taylor micro-scales, λ_g, were estimated. All these steps finally allowed the estimation of a computationally affordable spatial resolution to perform a high fidelity DNS of Hooper's sub-channel flow.

In conclusion, it should be emphasized that the results of the macroscopic flow processes are well captured with URANS. These pulsations strongly correlate with the periodic flow pulsations experimentally observed in Hooper's experiments [4,5]. In order to give a qualitative impression of the development of the flow within the sub-channel geometry of Hooper's configuration, Fig. 1. shows the results of the velocity magnitude. Through the calibration procedure of the original mass flow rate, the spatial resolution for computationally affordable high fidelity DNS was finally estimated.

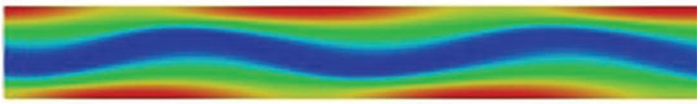

Fig.1. Flow pulsations in a sub-channel of a bare rod bundle. Contours are shown for the plane-xz at y=0. Flow di-rection is from left to right.

References

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CFD ANALYSIS OF FLOW THROUGH THE SECTION OF A HTR PEBBLE BED

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This project involves the execution of a thermal-hydraulic CFD analysis of a high temperature reactor (HTR) using specialized computational codes. Analysis is performed in order to identify and better understand the phenomena occurring inside the reactor core. For research purposes both commercial and non-commercial software were used. This approach will verify whether „open source" code will provide results on the same level of reliability and accuracy as professional tools. However, due to the very high demand for computing power of a CFD analysis, the model will include a section of the HTR pebble bed. The simulation is divided into three main stages: geometry modelling, high resolution mesh generation and calculation execution. All steps can be carried out using software either within a single package (free or paid) or software from several different packages.

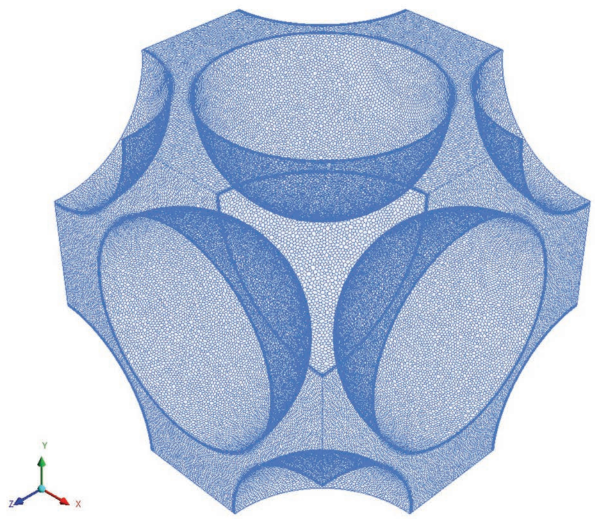

Fig. 1 Face-Centered Cubic mesh done in Star-CCM+ software (preliminary data).

Determination of the volume temperature distribution inside the fuel pebble of a high temperature reactor makes a significant contribution to this field of knowledge. Moreover, this project will provide new experience for computer modelling, because the case itself is extremely complicated from the numerical point of view. The importance of this analysis is even greater due to the fact that acquisition of experimental data is impossible. The results obtained from CFD simulations for the pebble bed will be used to determine the temperature of the graphite in various places and for further studies of the possibility of graphite layer cracks and leaks of fuel. Such knowledge may also be used for further research on strengthening the graphite used in high temperature reactors or in an environment with similarly difficult conditions. Additionally, these highly detailed data will improve our understanding of the physical phenomena occurring in the field of thermodynamics, strength of materials, heat transfer and fluid mechanics.

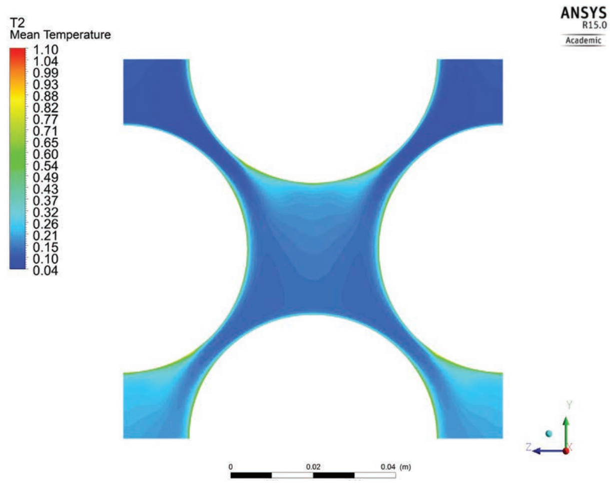

Fig. 2. Cross-section of cubic mesh with presented mean temperature distribution for helium (preliminary data).

The mesh used for these calculation was produced by Afaque Shams from the Nuclear Research and Consultancy Group (NRG). For a better understanding of the phenomena occurring inside the pebble bed, the flow rate of helium was measured and presented based on the shortest distance between two balls positioned in the middle of the opposite walls. The data obtained were compared with various turbulent models and the reference model with the q-DNS approach.

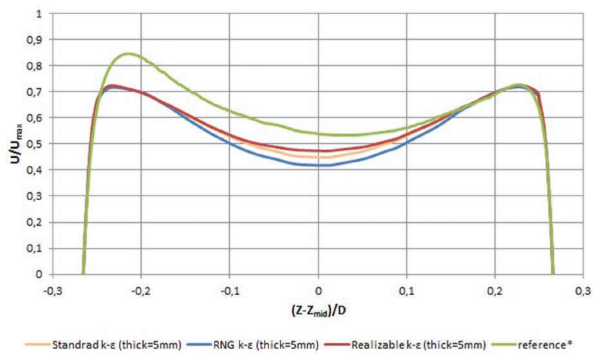

Fig. 3. Velocity profile for helium for different turbulence models inside the pebble bed (preliminary data).

SEVERE ACCIDENT SEPARATE PHENOMENA STUDY DURING THE IN-VESSEL ACCIDENT PHASE IN LIGHT WATER REACTORS

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In the actions undertaken by the Deterministic Analysis Team, severe accident analysis is one of the research topics investigated. Severe accident analyzes are performed with the use of computational tools run on the CIS high power computing cluster. Analyses have been made for Light Water Reactors (LWRs) and new Generation IV reactors (gas cooled). These analyses will from the basis for the preparation of the Safety Analysis Report of the Polish nuclear power plant. The calculations performed for a typical water reactor involve scenarios leading to core damage and fission product release, e.g. Loss of Coolant Accident without Safety Injection (LOCAwSI) and Station BlackOut (SBO).

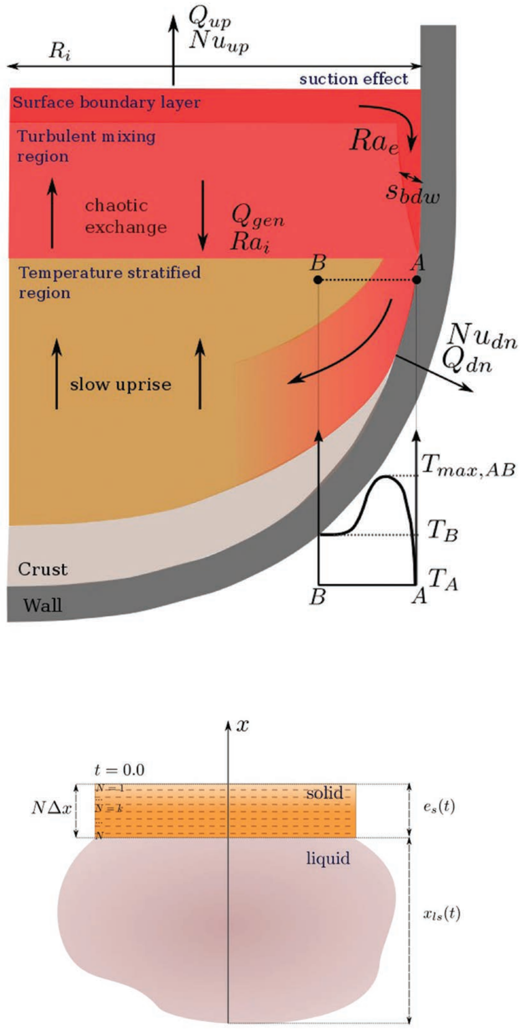

Fig. 1 Molten fuel pool, convection phenomena (upper) and melting problem illustration (lower).

Apart from a full analysis of the severe accident scenarios analyses involve research on specific topics present during core damage scenarios. One of these is modelling the 1D fusion/solidification plane fusion front two-phase problem, which is significant during molten stratified pool formation. Of prime interest for the transient upper steel layer fusion/solidification for the vessel failure risk is the focusing effect, which depends on the ability of the steel layer to conduct heat towards the lateral vessel walls. This issue becomes the first step towards more complicated cases connected to full plant transient calculations.

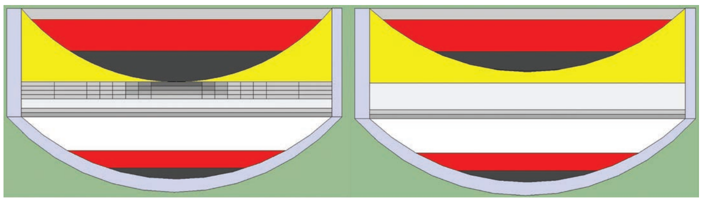

Fig. 2 The two possible scenario cases for the core support plate – bottom and top heat flux (left) and no contact case (right).

The second issue studied, which influences the course of the severe accident scenario, is core support plate failure due to thermal and mechanical loads. The melting of the steel core support plate is crucial; it can increase the molten metal layer thickness and decrease the focusing effect. Studies involved scenarios in which the plate was under the influence of the heat flux coming from the bottom, top or both molten fuel pools. Sensitivity studies were undertaken showing the need for future studies and creation of a better model representing the core support plate.

The analytical and numerical models will be incorporated in future analyses and calculations.

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INTRODUCTORY MEASUREMENTS OF PARTICULATE MATTER CONCENTRATION IN AMBIENT AIR IN THE VICINITY OF A POTENTIAL LOCATION OF A NUCLEAR POWER PLANT (KROKOWA COMMUNE)

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[2]Warsaw University of Technology, Warsaw, Poland

Air quality monitoring has been undertaken for dust pollutants better to characterise the baseline air quality in the vicinity of a potential location of the first Nuclear Power Plant (NPP) in Poland. The monitoring campaign was undertaken between 21st of May 2014 and 28th of May 2014, as part of a Central Laboratory for Radiological Protection (CLOR) research project and the process of testing new equipment. This paper presents the concept of the application of a mobile air monitoring station to support environmental survey in areas proposed as locations of nuclear power stations.

Preliminary measurements of particulate matter concentration in ambient air were performed in Krokowa commune. This location was indicated, among 2 other areas, as a feasible site for the first nuclear power plant in Poland. A mobile container with a set of hi-tech analyzers was built within the framework of the "4Labs" Project, financed by the Regional Operational Programme for the Mazovieckie Voivodship for the years 2007-2013. It was set up in the village of Lubocino, approx. 4 km southeast of the location of the former Żarnowiec NPP.
Six analyzers were used for measuring the number and mass concentration of particulate matter in different size ranges (fig. 1). The one-week measurements covered monitoring of PM_{10} and $PM_{2.5}$ concentrations, number concentration of particles in the range 20 nm to 1 µm (fig. 3), number and mass concentration of particles in the range 0.3 to 10 µm. The mobile container was also equipped with a meteorological station, which continuously records wind, temperature, humidity and pressure conditions during the monitoring period.

Fig. 1. Mobile container and measuring devices.

In general, particulate matter concentrations during the course of the monitoring campaign were low. Dust concentrations decreased significantly after rainfall. The drop in data capture was due to a massive storm that occurred on the 24th of May.

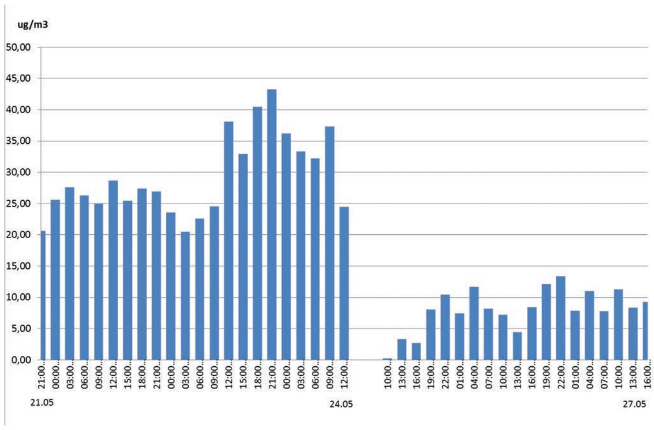

Fig. 2. PM10 concentration during monitoring period.

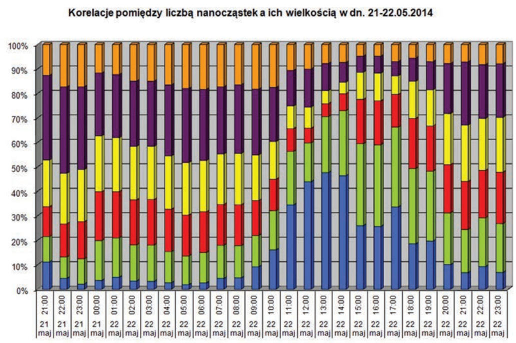

Fig. 3. Cumulative percentage plot of number of particles registered in different particle sizes as a function of time.

Preliminary measurements in Krokowa were the introduction to a more robust assessment of air quality at this site. Registering zero-point of the environment is an important element of preparation for the construction of a NPP, as it will become the benchmark for future analyses. The need for both radiological and air quality monitoring in terms of gaseous and dust pollutants before running the plant and during its operation should be emphasised, such they allow the subsequent identification of any anomalies in the operation of the facility at any time.

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THE THERMAL TO 14 MeV NEUTRON CONVERTER COMMISSIONING IN THE MARIA REACTOR

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The ITER & DEMO thermonuclear research facilities are to be operated using the deuterium – tritium nuclear fusion reaction. Fast neutrons of 14 MeV energy resulting from the fusion reaction are essential to carry away the released thermonuclear energy and to breed tritium.

The 14 MeV neutrons are also the main source of radiation damage of the construction materials. Therefore, constructional materials of thermonuclear devices need to be selected carefully. Radiation resistance of existing and new materials needs to be tested in a 14 MeV neutron field. Shortage of strong 14 MeV neutron sources can be supplemented by a fission reactor equipped with a thermal to 14 MeV neutron converter. The conversion is based on thermal neutron capture by ^{6}Li:

$$^{6}\mathrm{Li}+n \rightarrow \mathrm{T}+{}^{4}\mathrm{He} \quad (Q \cong 4.79\ \mathrm{MeV}) \qquad (1)$$

followed by reactions of the produced triton with deuterium:

$$\mathrm{T}+\mathrm{D} \rightarrow {}^{4}\mathrm{He}+n \quad (Q \cong 17.58\ \mathrm{MeV}) \qquad (2)$$

or ^{6}Li:

$$\mathrm{T}+{}^{6}\mathrm{Li} \rightarrow {}^{8}\mathrm{Be}+n \quad (Q \cong 16.02\ \mathrm{MeV}) \qquad (3)$$

The energy of the outgoing neutrons in reactions (2) and (3) is 14 MeV.

Such a device has been designed and constructed in NCBJ under the National Centre for Research and Development Strategic Programme for Technologies Supported Development of Safe Nuclear Power Engineering.

Lithium deuteride (LiD) and lithium deuteroxide hydrated by heavy water (LiOD·D2O) have been used as the conversion materials in the MARIA reactor. Both enriched with ^{6}Li isotope to 96%. Due to the corrosivity of the above mentioned compounds to aluminium, stainless steel has been selected as the converter construction material. Due to the MARIA reactor core characteristics a cylindrical shape has been chosen for the converter. The converter construction consists of a set of concentric tubes, located inside a vertical channel in the reactor beryllium moderator. A cylindrical converting layer surrounds a container with the irradiated samples.

Neutron and triton transport calculations have been carried-out to estimate the thermal to 14 MeV neutron conversion efficiency and optimize the converter construction. The theoretical conversion efficiency has been estimated as ca. $2\cdot10^{-4}$ [1].

Due to the extremely high reaction (1) cross-section for thermal neutrons and the high energy release (4.79 MeV), a significant amount of heat is generated in the converting layer when placed in the reactor. Proper converter cooling is, therefore, a crucial issue. The heat transport has been assesed to ensure proper device cooling conditions. A set of thermocouples has been installed in the converter to monitor its temperature distribution on-line. The influence of the converter on reactor excess reactivity has been studied. Safety analyses of steady states and transients, including radiological hazards, have been done. The calculations performed and analyses allow the converter to be designed and its operation limits and conditions to be formulated [2].

According to current National Nuclear Energy Agency permission, it is allowed to operate the converter located in a K-VIII vertical channel in the MARIA reactor core. In this location the excess reactivity perturbation caused by the converter is low (-0.06$) and a relatively low amount of heat is generated inside the converter due to the lower thermal neutron flux density.

The first tested operation of the converter in the MARIA reactor was launched on 18th September 2014 under the supervision of the National Nuclear Safety Department representatives. Testing operation took 135 h. The neutron energy spectrum inside the converter depends on its location in the reactor core. In the chosen location, during testing operations the 14 MeV neutron flux density was estimated to be over 10^9 cm^{-2} s^{-1}, whereas fast fission neutrons inside the converter achieved 10^{12} cm^{-2} s^{-1}, and thermal neutrons were reduced down to 10^9 cm^{-2} s^{-1}. The neutron flux densities were measured by means of the activation method with a set of various activation foils.

Among many things, a set of ITER construction steels have been irradiated in the above mentioned neutron field. They are currently under investigation.

Taking into account the feasibility of almost incessant converter operation for a number of months, it has emerged as one of the most powerful (in terms of fluence), currently available 14 MeV neutron sources.

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ACCREDITATION OF A METHOD FOR DETERMINATION OF PLUTONIUM ISOTOPES IN URINE SAMPLES

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The Radiation Protection Measurements Laboratory (LPD) aspires to extend the capabilities of the internal monitoring of workers of the nuclear facility at Otwock based on in vitro measurements of alpha emitters. Work under the strategic project of The National Centre for Research and Development: "Technologies supporting development of safe nuclear power engineering" (Task 6, Stage 12) allowed the implementation of a method for determination of plutonium isotopes in urine samples (operating procedure W-9) to the LPD management system documentation and the accreditation of this procedure.
Accreditation of the W-9 procedure required carrying out validation of the method, which according to EN ISO/IEC 17025:2005 is the confirmation that the method is appropriate for the particular purpose. An analysis of the main parameters which characterize the validated method (selectivity, accuracy and linearity) was conducted.

Apparatus:

An Alpha Analyst Model 7200 spectrometer (Canberra, USA) equipped with passivated silicon planar implanted detectors was used for measurement of alpha spectra.

Procedure:

- plutonium isotopes co-precipitation with calcium phosphate,
- mineralization of precipitate with 65% HNO_3,
- chromatographic purification (Bio-Rad, AG 1-X2, 50-100 mesh, chloride form),
- electrodeposition,
- alpha spectrometry measurement.

In order to validate the method for determination of plutonium isotopes the following urine samples were analyzed:

- blank sample,
- urine sample with a ^{236}Pu tracer,
- urine samples spiked with ^{238}Pu and ^{236}Pu tracer (at three different levels of ^{238}Pu activity),
- urine samples containing plutonium isotopes ^{238}Pu, $^{239+240}$Pu (intercomparison exercise).

Selected criteria used to evaluate the validated method are shown in Table 1.

Table 1. Selected acceptance criteria

Parameter	Criterion
Selectivity	Possibility of plutonium isotope identification in the alpha spectra (lack of interfering isotopes, appropriate peak resolution)
Accuracy (trueness and precision)	Trueness: $\left[C_{lab} - C_{ref}\right] \leq 2,58 \cdot \sqrt{(u_{lab}^2 + u_{ref}^2)}$ Precision: $100\% * \sqrt{\left(\left(\frac{u_{ref}}{C_{ref}}\right)^2 + \left(\frac{u_{lab}}{C_{lab}}\right)^2\right)} \leq 15\%$ Where C is the activity concentration of ^{238}Pu (*lab*- value measured by LPD, *ref*- reference value) and u is the measurement uncertainty.
Linearity	Correlation coefficient $R^2 \geq 0,98$

All of the analyzed validation parameters met the established criteria which means that the method investigated is appropriate for determination of plutonium isotopes in urine samples. Accuracy of the method was also verified in an interlaboratory comparison organized by PROCORAD in 2013 - "Actinides in urine" and "Plutonium solution". Target values of plutonium activity concentration in two urine samples and results obtained by LPD were close. Detailed information about the validation process is included in Report B [1].

In June 2014 evaluation by the Polish Centre for Accreditation, related to the planned supervision and extension of the accreditation, took place at the Radiation Protection Measurements Laboratory. As a result of this evaluation the scope of the accreditation was extended by measurement of the activity of plutonium isotopes in urine using alpha spectrometry, in accordance with the operating procedure W-9.

Reference

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EPITHERMAL NEUTRON SOURCE AT THE MARIA REACTOR

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In 2001 a research programme aimed at producing a therapeutic neutron beam of parameters required in boron neutron capture therapy (BNCT) started at Świerk in Poland. To this end the concept of the in pool fission converter was developed [1], [2]; it consisted of 98 EK–10 fuel rods with 10% U-235 enrichment. A UO_2 dispersion in metallic magnesium was to be the fuel. As the use of a converter generates the necessity of shaping, moderating and filtering the neutron beam, all of the BNCT line was constructed in 2010. Completing the work coincided with the departure of the treatment method in Europe; other centres providing a beam were shut down. The main reason for this being that wide acceptance by the medical community had not been gained, due to many difficulties arising in treatment carried out using a reactor neutron beam. Finally, the nuclear fuel in the converter has not been activated despite the readiness for installation.

However, elsewhere work on BNCT has not been interrupted; what is more, the commercial production of equipment (cyclotrons), prepared for installation in hospitals, was announced in Japan, which would solve the constraints arising from the reactor-based treatment. Considering the various unique features of BNCT (including the fact that it is incommensurable with available tumour therapy, because it gives a solution in the case of tumours, for which any other treatment could be used) and the estimated treatment price, which does not exceed the rest of the patient's treatment methods, it should be assumed that introduction of this therapy for European hospitals is highly probable [3].

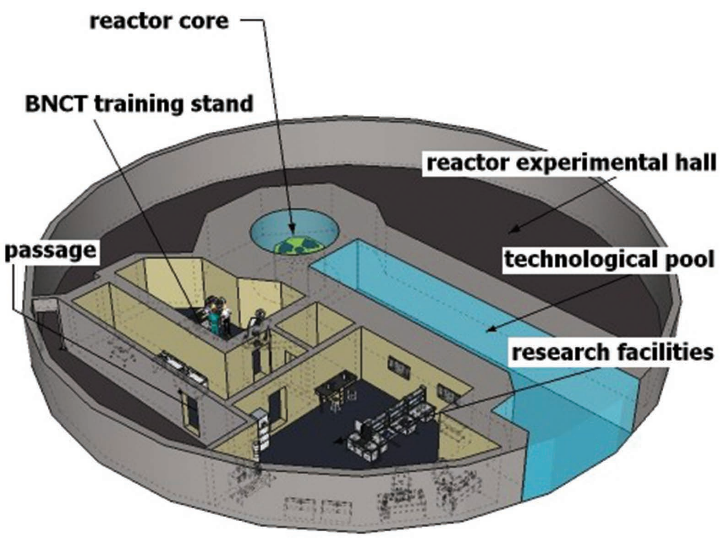

Fig. 1. Fission converter based research/training station at the MARIA research reactor.

In 2013 in Poland a programme aimed at generating a neutron beam for many different applications was resumed [4]- it would be the only epithermal neutron beam of flux density exceeding 10^9 n cm^{-2} s^{-1}in Europe. There are eight horizontal research channels at the MARIA reactor, two of them were allocated to a training and research station (see Fig. 1). In 2014 it was decided to construct a new fission converter, powered by uranium fuel plates made for this purpose.

A series of studies are being carried out to prepare the neutronic, thermal – hydraulic and engineering design of the converter.

BNCT combines many different fields of research. Beyond the irradiation facilities the second important issue is the boron carriers. Neutron capture ^{10}B-containing compounds should cause preferential killing of tumour cells and induce therapeutic effects. There is a strong group of scientists in Poland with significant achievements in this area and they report the need for irradiation experiments on the use of boron compounds. Also, for the development of many other research areas it is necessary to conduct research using a high intensity epithermal neutron source, e.g.: dosimetry, electronics, technical equipment, radiobiology, molecular biology.

Special recombination ionization chambers have been developed specially for epithermal neutron beams from fission converter sources [5]. Radiation effects of BNCT are associated with a four-dose-component radiation field - boron dose (from the $^{10}B(n,\alpha)^7$ Li reaction), proton dose from the $^{14}N(n,p)^{14}C$ reaction, neutron dose (mainly fast and epithermal neutrons) and gamma-ray dose (external and from the capture reaction $^1H(n,\gamma)^2$ D). The secondary charged particles generated in tissue have a LET spectrum resulting from the beam composition and therefore a microdosimetric characterization of the beam can be useful for monitoring its quality and possible variations with time. A set of specially designed recombination chambers makes it possible to determine the total absorbed dose, dose components and a few other dosimetric quantities. Therefore, using one of they in the beam monitoring cauld be of special interest in accelerator based BNCT beams.

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