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Numerical heat transfer analyses of wrapped wire bundle using different turbulent Prandtl number correlations

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The wrapped wire bundle is a common configuration in many nuclear engineering applications, such as in fuel assemblies of liquid metal fast breeder reactors. An accurate modeling of heat transfer mechanisms is thus crucial for ensuring the safe operation of advanced nuclear reactors that use liquid metal as a coolant. Low-Prandtl number fluids such as liquid metal exhibit several distinct characteristics that set them apart from conventional fluids. Owing to this characteristic, the thermal boundary layer of low-Prandtl fluids is considerably larger than the momentum boundary layer. As a consequence, the Reynolds-averaged Navier–Stokes (RANS) approach to model heat transfer of liquid metal accompanied by a constant turbulent Prandtl number close to unity cannot be considered completely suitable for the low-Prandtl number liquid metal. Hence, advanced techniques are thus needed. In the present work, in order to predict the temperature distribution and thermal performance in a wrapped wire bundle cooled by liquid Lead-Bismuth Eutectic (LBE), a CFD model investigating several turbulent Prandtl number correlations is adopted. In particular, the Generalised Gradient Diffusion Hypothesis (GGDH) correlation for the calculation of turbulent heat fluxes is considered. The turbulent Prandtl number correlation thus takes into account the effects of the temperature and velocity gradients on the turbulent transport properties of the LBE. The obtained results show that the heat transfer behaviour of the wrapped wire bundle in LBE is strongly influenced by the LBE flow velocity and the arrangement of the wires. The use of a turbulent Prandtl number correlation improves the accuracy of the predictions and provides insight into the mechanisms that govern the heat transfer behaviour in the wrapped wire bundle. This study demonstrates the importance of accurately modeling the turbulent Prandtl number and the flow characteristics of the LBE to obtain reliable predictions of heat transfer in nuclear systems and highlights the potential applications of the CFD simulations in the design and safety assessment of nuclear reactors.

P02

Thermal Hydraulic Analysis of Inward and Outward Fuel Plate Buckling in a Typical MTR Reactor

ABSTRACT

In the present work, numerical study of inward and outward buckling of two successive fuel plates of a typical MTR reactor is investigated using CFD code. Fuel plates buckling results in partial blockage of the hot channel. Both buckling towards the inside and outside are considered. The simulation is conducted for different blockage levels of the nominal flow area, i.e., 0%, 20%, 40%, 50%, 60% and 70% for inward buckling. Blockage levels of 0%, 20%, 40%, 50%, 60% and 70%, 80%, and 90% are considered for outward buckling. The impact of the flow field redistribution in four successive channels on the cooling capacity of each channel is investigated. The obtained results show that for inward buckling ratio greater than 50%, critical phenomena will occur which could affect the clad integrity. Moreover, for inward buckling of 70%, the maximum clad temperature in the blocked channel reaches the value associated with the onset of nucleate boiling at the operating pressure. On the other hand, for outward buckling of 90%, critical phenomena which could affect the clad integrity will occur.

P03

The impact of graphite swelling on the fuel cycle and safety of graphite moderated MSRs

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The Molten Salt Reactor (MSR) was identified as one of the candidates for the Generation-IV reactors. Among the various MSR concepts, the graphite-moderated MSR has attained the highest Technology Readiness Level and is closest to commercial deployment. One of the main challenges of this MSR concept is associated with the significant initial shrinkage and subsequent graphite swelling [1] which can greatly impact the reactor behaviour. For instance, graphite expansion will affect both the neutronics, by locally changing the fuel-to-moderator ratio, and the thermal-hydraulics, by impacting the flow cross-sectional area [2]. Also, replacement of the graphite core represents a significant economic penalty. For these reason, it is important to develop computational models that can accurately predict this phenomenon taking into account its dependence on graphite temperature, local neutron flux, spectrum and salt flow characteristics.

The objective of this PhD work is to develop and implement a high-fidelity methodology and associated code for coupled thermal-mechanics, thermal-hydraulics and neutronics to model the graphite dimensional variation during irradiation and to evaluate the impact of graphite swelling on both safety and fuel cycle of graphite moderated MSRs.

GeN-Foam, a solver based on OpenFOAM C++ library, has been chosen for this study due to its capability to model challenging multiphysics effects that are difficult to simulate using traditional codes [3].In this preliminary study, a GeN-Foam model for the MSRE primary circuit is developed and a coupling is established with the Serpent Monte Carlo model. Additionally, a model for predicting graphite temperature in porous-medium simulations will be developed using a heat resistance approach, and its accuracy will be validated against experimental data from the MSRE benchmark conducted at ORNL (Oak Ridge National Laboratory). Finally, a model for graphite swelling will be implemented in the thermal-mechanic sub-solver of GeN-Foam and this solver will be used to evaluate the temperature-induced deformation of the main structures.

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P04

Safety Analysis of the Ghana Research -1 Post Core-Conversion for Small and Large Reactivity Transients.

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Abstract

The core of Ghana's only research reactor has been converted from highly enriched uranium (HEU) to low-enriched uranium (LEU). This conversion has led to some significant changes in reactor parameters which may affect variation in the temperature and heat distribution in the core during reactivity transients. The Program for reactor transients (PARET/ANL) was used to model the LEU core of GHARR-1 under transient conditions of various reactivity insertions. The core was modelled in two channels: the hot pin and the average of the remaining pins. For small reactivity insertions of 1.94 mk and 2.1 mk, the reactor power peaked at 30.2 kW and 34.2 kW respectively. And the coolant temperatures were 52.42 ◦C and 54.24 ◦C respectively, far below the saturation temperature of 100 ◦C at a pressure of 1 atm, hence boiling is not expected to occur in the core. The peak clad and fuel temperatures were found to be 62.73◦C and 63.24 ◦C for 1.94 mk and 65.53 ◦C and 66.10 ◦C for 2.1 mk respectively. These clad and fuel temperatures are far below the melting points of the Zicaloy-4 clad material and UO2 fuel. For large reactivity insertions of 5.1mk and 6.71 mk, the coolant temperatures peaked at 87.6 ◦C and 98.8 ◦C respectively, which is close to the coolant saturation temperature of 1008 ◦C, hence boiling may occur under such large reactivity insertions. For reactor transients it was observed that temperature rise for LEU core was lower than that in the previous HEU core. The results obtained were found to agree with the available experimental results.

P05

Investigation of C5G7 Monte Carlo solutions with Serpent

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The first phase of the C5G7-TD benchmark consists of a series of neutronic transient exercises, and thus far, benchmark participants have reported results using deterministic and hybrid codes [1]. In this study, we explore the application of the Serpent code to investigate a Monte Carlo solution for the benchmark exercises, presenting the first purely stochastic solution.

The Serpent transient simulation is conducted in two main steps. First, a criticality calculation is performed to generate the initial neutron source. This is followed by a transient phase where the defined time interval utilizes the initial neutron source. The final state at the end of this phase is saved as a new neutron source, which is then used in the subsequent time step. This process continues until the final time step of the transient is reached. Population control, as defined by the user, is applied to each time step or sub-step [2].

To assess the accuracy of the Serpent results, a comparison was made with the RMC results [3]. RMC is a hybrid deterministic-stochastic code and serves as the closest alternative to a Monte Carlo solution. The comparison reveals good agreement between Serpent and RMC for both global and local parameters. The dynamic k shows a difference ranging from 30 to 200 pcm, while the relative fission rate exhibits less than 9% variation. These results can be further improved by increasing the number of simulated neutrons in the system. It's worth noting that beta-effective results were not presented since Serpent currently lacks the implementation of beta-effective calculations for transients.

While it is true that Serpent simulations are computationally expensive compared to deterministic codes, with computational times per exercise ranging from 52h to 120h using 760 processors, it is important to consider that this benchmark was initially designed for deterministic codes using homogenized cross sections. Therefore, the Serpent results, although noisier, require significant computational resources compared to faster and more precise deterministic results. Nevertheless, it is crucial to acknowledge the capability of Serpent to model complex systems without introducing approximations, which provides a valuable advantage in certain scenarios.

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P06

Numerical simulation of an experimental device for reducing height irregularity during NTD of silicon in the WWR-K reactor

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A nuclear reactor is a complex engineering and technical installation that generates radiation. Because of this, the experimental measurement of the radiation characteristics of a nuclear reactor is a difficult task, and in some cases technically impossible. The modern development of calculation codes and tools makes it possible to determine the radiation characteristics of a nuclear reactor with sufficient accuracy and reliability. Due to this, numerical simulation of physical processes occurring in a nuclear reactor has become one of the main scientific research methods [1].

The WWR-K reactor is a multipurpose research reactor with many vertical and horizontal irradiation positions used for a wide range of scientific and applied problems. One of these tasks is R&D to develop a method of neutron-transmutation doping of silicon ingots of various diameters and heights [2, 3]. After the design of the experimental device was developed, detailed neutron-physical calculations were carried out to determine the energy and height distribution of neutrons in it. The experimental device consists of an irradiation channel and a profiling screen. The simulation was performed using the Monte Carlo method. The calculation study is based on the development of a detailed numerical model that takes into account the physical processes of the interaction of neutrons with a silicon structure in an experimental device.

The results showed a well-defined energy and height distribution of neutrons within the experimental device, providing crucial insights into its performance and safety. In this way, the simulation results will make it possible to optimize the parameters of the device, provide more uniform doping of silicon with neutrons, and offer recommendations for improving the efficiency of the process of neutron transmutation doping of silicon at the WWR-K nuclear reactor.

Keywords: silicon doping, WWR-K reactor, experimental device, Monte Carlo method, height irregularity

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P07

Abstract template for Joint ICTP-IAEA Workshop on Open-Source Nuclear Codes for Reactor Analysis

Berta Bürger, János Gadó

FUROM-FBR is a fuel performance code developed by the Centre for Energy Research, Budapest, for the simulation of fast reactor fuel behaviour. The code was developed on the basis of the FUROM code which serves for thermal reactor fuel performance in the institute and at the MVM Paks NPP VVER-440 units for years.
FUROM-FBR is capable to calculate fuel behaviour in helium, sodium, lead and lead-bismuth cooled reactors. Basic coolant data and power history have to be provided by the user. The axial distribution of fast neutron flux can be specified by the user or can be calculated using appropriate correlations. On this basis the code is enabled to calculate the temperature distribution in the fuel rod in open and closed gap conditions, the mechanical phenomena as thermal expansion, densification, swelling, creep, elastic and plastic deformations and also fission gas release. The migration of plutonium, pores and oxygen vacancies were built in the fast reactor version.
The original FUROM code was validated by using internationally available experimental results for thermal reactors. Similar validation process for FUROM-FBR was nearly impossible so far because of the unavailability of fast reactor data.
In the EU PuMMA project different aspects of plutonium management are investigated. A work package is devoted to the validation of fast reactor fuel behavior calculations to experiments performed with high Pu content fuel. The geometry and material composition as well as the irradiation specifications of three fuel pins were provided within the framework of the project.

In the first phase of the task, calculations were performed with the use of several different fuel performance codes – including FUROM-FBR – and their results have been compared with each other. Moreover, sensitivity of different parameters was studied in order to be prepared for the comparison with experiments.

In the second phase of the task the results of calculations will be compared with the experiments and conclusions for the need of further code development will be drawn.

P08

Abstract template for “Transmutation of minor actinides in thermal and fast neutron spectra”

Rania Sharaf [1], Anna Therekhova [1,2], Фан Тхи Фыонг Чанг[2]

Several approaches have been proposed for managing spent nuclear fuel and mitigating its radioactivity, which is primarily attributed to minor actinides that remain hazardous for hundreds of years following unloading from the reactor. One such approach involves utilizing specialized or power nuclear reactors to transmute minor actinides into fission fragments, as part of a closed fuel cycle that promotes the reproduction of fission nuclei.

The study aimed to examine the transmutation of minor actinides and evaluate various approaches for reducing the radioactivity of spent nuclear fuel. One promising method involves transmuting minor actinides into fission fragments, which can be achieved through specialized or power nuclear reactors.

To investigate this, the researchers developed mathematical models of fuel assemblies using the "Serpent" software package *(a computer program developed by the State Scientific and Technical Center in Finland in 2004, which employs the Monte Carlo method to enable highly precise calculations of reactor systems and their operation)*, which included:

1) fuel assemblies of the Rbec reactor with UO_2 fuel without MA,
2) fuel assemblies of the Rbec reactor with MOX fuel without MA,
3) fuel assemblies of the WWER-1000 reactor with UO_2 fuel without MA.

They then conducted neutron-physical calculations of these fuel assemblies, which allowed them to track the changes in minor actinides over the course of reactor operation. The accumulation of isotopes of curium, americium, and neptunium is a common occurrence during the operation of reactors. These elements are primarily responsible for the elevated levels of radioactivity observed in spent nuclear fuel and its byproducts after processing.

The findings of this study suggested that transmutation of minor actinides is more efficient in the thermal spectrum of neutrons as opposed to the fast spectrum. The researchers attributed this difference to the fission cross section of minor actinides, which is greater in the thermal spectrum compared to the fast spectrum.

P09

CFD simulation of debris bed formation in the unheated DEFCON-S experiment

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When severe accident occurs in a light water reactor, molten corium can be collected in the reactor cavity. If debris bed is not properly cooled, it can react with the basement concrete and cause melt through which can threaten the integrity of the containment. In the analysis of severe accidents phenomena, it is important to interpret such phenomena using CFD codes. In this study, the formation of a non-heating debris bed formation experiment, DEFCON-S, is simulated using the commercial CFD code, STAR-CCM+. To determine which between the default values provided by STAR-CCM+ and the modified constants effectively represent the interactions between inter-particles and particle-wall, the simulation is performed for two cases with different rolling resistance coefficients. In the comparison between the results obtained with the default values and the results obtained with the modified rolling resistance coefficient, it is observed that the results obtained with the modified coefficient closely resembles the results of DEFCON-S. The modified rolling resistance coefficient leads to a diameter error of 4.6% and a height error of 3.8%. In further studies, the analysis under the same condition will be simulated using an open source CFD code, OpenFOAM. Based on the comparison and assessment of results between STAR CCM+ and OpenFOAM, upon confirming the validity of utilizing OpenFOAM for analysis, OpenFOAM will be utilized for the analysis of severe accident phenomena using its enhanced flexibility and the ability to customize various parameters.

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P10

Verification and Validation of Coupled Monte Carlo Neutronics and Sub-Channel Thermal Hydraulics Codes

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The proposed research aims at enhancing numerical methods used for research reactor safety analysis through performing benchmarks of coupled Monte Carlo neutronics and 3D thermal hydraulics code. Various methodologies have been developed for the coupling of monte carlo codes, such as SERPENT and MCNP, and sub-channel codes, such as CTF, for use in nuclear reactor modelling. The goal of this research project is to develop a coupling methodology using SERPENT and CTF, use it to perform modeling, calculation, and analysis, and then to verify and validate the reliability of its use through comparing it to available literature results and to available experimental measurements of chosen facilities. Exisiting verification and validation benchmarks for research reactors are available through the IAEA CRP on "Benchmarking against Experimental Data of Neutronics and Thermohydraulic Computational Methods and Tools for Operation and Safety Analysis of Research Reactors"[1]. Uncertainty analysis will additionally be performed to quantify the reliability and accuracy of the modelling method, and to aid in development of best estimate modelling approaches.

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P11

Determination of Affected Steam Generator During Primary-To-Secondary Leakage Accident

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One of the possible criteria for launching the automated management system of the primary-to-secondary uncompensated leakage (PRISE) is the exceeding of the absorbed dose rate setting near the main steam lines of the affected steam generator (SG) at VVER-1000. During the initial event, the generated alarm signal causes the switching on the relevant safety systems, that ensures further transfer of the reactor to a safe operational state [1]. Meanwhile, the system has to identify and localize the affected SG correctly.

^{16}N is the most informative marker for controlling the PRISE leakage in SG at the nominal operating level. The method allows monitoring the mass flow rate trend of the PRISE leakage and organizing the automated control of the PRISE accident. Thereby, ^{16}N activity in different positions of the primary and secondary circuits has been estimated by using neutron-physical and thermohydraulic models developed in MCNP6.2 and RELAP5/MOD3.2. Based on the obtained transport time of the main indicator and the steam activity, the dependences of the detected gamma radiation on the activity in the detection area for reference radionuclides, that correlate with the mass flow rate during the PRISE accident, have been obtained by modeling the "source of ionizing radiation – main steam line – Geiger-Muller counter" system [2].

The different types of detectors such as semiconductors, scintillation detectors and gas-filled counters that could be used for the affected SG identification have been analyzed [3]. The calculated detector responses show that the most effective indicator for launching the automated management system of the PRISE accident is the GIM-204 detection unit in contradistinction to SGLM-202 and BDMG-04-02. Besides, the layout of the main equipment in the detection area does not allow to prevent the crosstalk appearance at the unaffected SG without the applicability of the additional modification of the detector responses.

In conclusion, the paper presents the results of the absorbed dose rate dependences on the mass flow rates during the PRISE accident for different operational states such as from 20 to 100% of the nominal power level, different combinations of the continuously operated reactor coolant pumps, etc. Additionally, the method for the treatment of the detector responses from detection units, mounted on the adjacent main steam lines, has been proposed.

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P12

Validation of Couple Neutronics and Thermal Hydraulics Codes through Steady State Thermal Hydraulics Parametric Calculation of TRIGA Mark II Research Reactor

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TRIGA Mark II research reactor is under operation in Bangladesh since 1986. The reactor operation life is currently being considered to extend as it has already passed long operating hour burning its fuels. However during design and modification, the reactor core should be designed in such a way, the safety parameter DNBR should not be allowed to drop below a chosen value under a high heat flux transient condition for the most adverse set of reactor conditions. Before doing detail safety study of the modified TRIGA core, it is mandatory to validate codes with the help of the verified data. Hence the impact of thermal hydraulics safety parameters on DNBR of TRIGA MARK II reactor have been studied in the present work with neutronics code SRAC [1] and thermal hydraulics code EUREKA [2]. At the beginning axial power profile at the hottest rod, along with surface and bulk coolant temperature profile was evaluated and compared with that of PARET [3] work. Once they are benchmarked, the impact of flow rate, inlet temperature and power on DNBR was computed by means of a suitable correlation as defined in EUREKA code. Over the length 0.381 m of the hottest channel the DNBR varies, with a minimum at the axial center. The peak heat flux occurs at the axial center of the hottest rod; therefore the DNBR is minimum at this position. All results matched significantly with that of PARET. The results proved the efficacy of couple SRAC and EUREKA codes for couple neutronics and thermal hydraulic study. The codes could thus further be employed in reactor transient study of the present core or detail design and safety parametric study of the modified TRIGA core.

Keywords: TRIGA; Thermal Hydraulics Safety; Computational Codes; Validation

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P13

OpenFOAM Applications for Nuclear Reactor Thermohydraulics

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At the "Jožef Stefan" Institute Reactor Engineering Division we have been applying OpenFOAM for different thermohydraulics phenomena encountered in nuclear reactors.

Experiments of interaction of a vertical jet with a helium-rich layer, were simulated by Krpan et al. [1]. It was shown that in regions with possible occurrence of Kelvin-Helmholtz and Rayleigh-Taylor instabilities, turbulent diffusion is underpredicted and turbulence model fails to replicate some phenomena. For this reason, a model to dynamically prescribe turbulent Schmidt and Prandtl numbers was proposed, which greatly improves the results.

The paper by Tekavčič et al. [2] considered simulations of an isothermal counter-current stratified flow case in the channel of the WENKA experiment using a morphology-adaptive multifield two-fluid model. Turbulent flow conditions near an interface were considered with asymmetric turbulence damping on the gas side of the interface. The comparison with experiment shows that this approach offers improved prediction of turbulent kinetic energy in the liquid, but with a cost of diminished accuracy of the predicted velocity profiles in the gas.

Turbulent flow in a fuel rod bundle was studied by Mikuž and Tiselj using Large Eddy Simulation (LES) [3] and URANS approach [4]. The results were compared with the measurements of the MATiS-H experiment performed at KAERI. LES has been found as a pretty accurate tool for simulations of the flow through the rod bundle, if sufficiently fine mesh is applied. In URANS the best agreement is achieved when resolved and modelled fluctuations are assumed to be uncorrelated, and appropriately summed up.

Single-phase heat transfer has been studied in a fully developed turbulent duct flow, which is in contact with a hot wall by Tiselj et al. [5]. Simple geometry of the experiment and fully developed velocity field are convenient for accurate numerical simulation, which was performed with wall-resolved LES approach.

The experiment performed in the MISTRA cylindrical vessel at the CEA, was simulated by Zajec and Kljenak [6]. It involved gradual heating of structures near the vertical walls to induce natural convection and mixing of the atmosphere and erosion of a previously established helium layer in the vessel upper region.

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P14

The Integrated Analysis of ITU TRIGA Mark Research Reactor

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Energy Institute, Istanbul Technical University

In this study, the detailed safety analysis of ITU TRIGA MARK-II was achieved. In order to do that, multi-physics approach was adopted by coupling MCNP6.2 Monte Carlo code for fuel element heat flux generation, ANSYS Fluent computational fluid dynamics code for heat transfer analysis, and NJOY nuclear data generation code for cross section and thermal S (α, β) scattering law data generation at desired temperatures. In addition, MAKSXF tool of MCNP6.2 was included for the generation of $S(\alpha, \beta)$ data with increments of any temperature interval and OTF tool of MCNP6.2 was used for fitting hundreds of cross sections at the desired temperature. The results showed that safety limits were not violated. After the receiving neutronic and thermal hydraulic results that were well agreed with the experimental data, thermal stress analysis for the standard stainless steel clad-low enriched TRIGA fuel was performed. The coupled thermo-structural method was adopted to find the stress, strain, and displacement of the components of the hottest TRIGA fuel element. In order to perform this analysis, two different computational models were proposed to treat fuel-clad interaction. The results confirmed that the thermal displacements and stress are relatively low compared to the safety limits and material limits under normal operating conditions.

P15

Developing an Open-Source Simulator Codes for Nuclear Reactors (Neutronic and Thermal-hydraulic analysis) Using Python language

Tariq Malatim

Libyan Atomic Energy Establishmen

In the Libyan Atomic Energy Establishment's Simulation and Computational Department, closed-source codes written in FORTRAN are utilized for neutronic and thermal-hydraulic analysis of the reactor. While these codes are highly efficient and accurate, they have licensing drawbacks and cannot be adapted to simulate next-generation reactors without external programs. Consequently, a simulator project was initiated in 2019 to develop our own code. Initially, C# and C++ were employed as programming languages during the first stage. However, after years of work, Python was deemed a suitable alternative due to its open-source environment and greater flexibility. The first stage of programming of a simple neutronic code that employs MC method and linking it with ENDF library was successfully completed, which is a promising thing. Additionally, the vast library of open-source Python codes on Github can be highly beneficial.

P16

Calculation of X-ray spectra characteristics and kerma to personal dose equivalent Hp(10) conversion coefficients: Experimental approach and Monte Carlo modeling

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(a): Faculty Ibn Tofaïl, Kenitra, Morocco

Abstract

This work aims to establish some X-ray qualities recommended by the international Standard organization (ISO) using the half-value layer (HVL) and Hp(10) dosimetry approaches. The HVL values of the following qualities N-60, N-8-, N-100, N-150 and N-250 were determined using various attenuation layers. The obtained results were compared to those of reference X-ray beam qualities and a good agreement was found (difference less than 5% for all qualities). The GAMOS (Geant 4-based Architecture for Medicine-Oriented Simulations) radiation transport Mont Carlo toolkit was employed to simulate the production of X-ray spectra. The characteristics HVLs, mean energy and the spectral resolution of simulated spectra have been calculated and turned out to be conform to the ISO reference ones (difference less than the limit allowed by ISO). Furthermore, the conversion coefficients from air kerma to personal dose equivalent for simulated and measured spectra were fairly similar (the maximum difference less than 4,2%).

Introduction

Ionizing radiation are used in a wide variety of fields such as radiotherapy, industry, research, education, etc… Nevertheless, this radiation can have harmful effects on human health. The radiation dose received by workers exposed to external ionizing radiation must be accurately determined in order to verify the conformity to the radiation protection requirements. In order to achieve this, the dosimeters used to detect the radiation dose should be calibrated according to internationally recognized standards.

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In this study, the X- ray Narrow-beam radiation qualities (N-60 to N-250) have been validated using both HVL and Hp(10) dosimetry approaches.

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Methodology

1- Equipments:

The measurements were performed with a Philips X-ray machine model MXR-32026-90, which has a pure tungsten anode with an angle of 20° and inherent filtration of 3mm of beryllium. The high voltage that can be applied to this X-ray tube varies from 10 to 320 KVp. The amount of the generated charge by the X-ray photons was measured with a spherical ionization chamber PS-50 connected to PTW Unidos electrometer. Concerning the attenuators, aluminum and copper sheets with purity higher than 99,98% were used. The experimental set-up is shown in Fig.1.

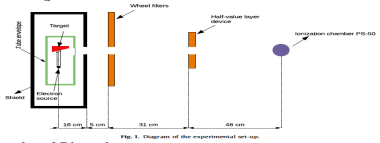

Fig. 1. Diagram of the experimental set-up.

2- Result and Discussion:

2-1- X-ray spectra characteristics:

The first and second half-value layers are determined for all the beam qualities of the N series, going from 60 to 250 kVp. The HVLs values obtained from the experiment and simulation were compared to the reference values (ISO). The obtained results are summarized in Table 1. According to the criterion of ISO 4037-1, if the first and second HVLs in a given material agree within < 5% for two X-ray beams, then these two beams can be considered as equivalent. The differences between the experimental, simulated and reference values for the first and second half value layers are shown in Figs. 2 and 3, respectively. As can be seen, the comparison in these parameters indicates a deviation inside of that allowed by the ISO. The maximum deviation between experimental and reference values of HVL_1 and HVL_2 is 3.64% and 4.11% respectively.

Quality	First HVL (mm)			Second HVL (mm)		
	Experiment	Simulation	ISO	Experiment	Simulation	ISO
N-60	0.24	0.23	0.24	0.26	0.27	0.26
N-80	0.60	0.60	0.58	0.63	0.64	0.62
N-100	1.13	1.09	1.11	1.17	1.12	1.17
N-150	2.43	2.37	2.36	2.56	2.49	2.47
N-250	5.38	5.25	5.19	5.45	5.27	5.23

Table1: First and second of the series results from the experiment simulation with GAMOS and reference values (ISO).

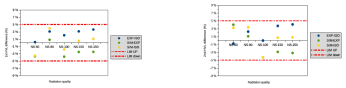

Fig 2-3, Difference in 1st HVL and 2nd HVL between results obtained from the experiment simulation and reference (ISO)

All radiation qualities meet the ISO criteria. Additionally, the maximum deviation between simulated and experimental values of HVL_1 and HVL_2 is 3,71% and 4,04% respectively. These results indicate that the performance of GAMOS can be considered adequate to simulate the experiment. The same comparison between the simulated and reference values indicate a deviation less than 3,97% for HVL_1 and 4,27% for HVL_2. Another comparison between the simulated spectra and the reference values was made in terms of mean photon energy spectra resolution. This comparison was summarized in Table 2 and the simulated fluence spectra are shown in Fig.4. The ISO 4037-1 states that the agreement for mean energy and spectral resolution must be within $< 5\%$ *and* $<$ 15% respectively. The difference in these parameters between the reference values and simulated results are shown in Fig.5 and Fig.6. All simulated spectra conform the ISO requirements. These results

Quality	Mean energy (keV)		Resolution (%)	
	Simulated	ISO	Simulated	ISO
N-60	48.28 ± 2.70%	48	34.84 ± 5.02%	36
N-80	65.63 ± 1.26%	65	30.79 ± 3.93%	32
N-100	81.85 ± 2.76%	83	25.94 ± 4.92%	28
N-150	117.26 ± 1.81%	118	36.32 ± 2.99%	37
N-250	204.9 ± 1.36%	208	27.94 ± 2.11%	28

Table 2: Mean energy and spectra resolution of narrow (N) radiation quality of simulated spectra and ISO reference.

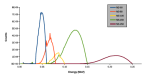

Fig. 4. X-ray spectra simulated by GAMOS.

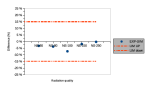

Fig.5. Difference in spectra resolution between reference values and simulated spectra

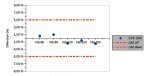

Fig.6. Difference in mean energy between reference values and simulated spectra.

Conclusion

the present work, we have established some reference X-ray spectra recommended by the ISO 4037-1 standard, especially the narrow-spectrum series. The obtained results are conform to the standard values of ISO within the permissible tolerance limits of ±5% for the first and second HVLs. Furthermore, the generation of X-ray spectra has been simulated using the GAMOS Monte Carlo code. The HVLs of the simulated spectra are matching with the experimental results. They are also compatible with the standard values of ISO 4037. Moreover, the X-ray spectra produced with GAMOS present a mean energy and a resolution that agree with those of the reference radiations.

Additionally, the conversion coefficients for personal dose equivalent were calculated both experimentally and by Monte Carlo simulation. The calculated conversion coefficients for the measured spectra are in good agreement with the standard values of ISO 4037-3. The conversion coefficients for the simulated spectra present differences limited in the interval [0.14%- 4.22%] compared to measured values. This is in accordance with the standard values of ISO 4037-3.

As summary, the results of this study indicate that the ISO narrow-spectrum series have been successfully established. In addition, the Monte Carlo model used to simulate the X-ray spectra is valid and can be used in works requiring knowledge of the X-ray spectral distributions.

Acknowledgmen

This research was supported through computational resources of HPC-MARWAN (www.marwan.ma/hpc) provided by the National Center for Scientific and Technical Research (CNRST), Rabat, Morocco.

P17

Coupled thermal hydraulics and neutron physics analyses of the SCW-SMR reactor concept

Tamás Varju [1], Péter Mészáros [1], Zeno Bertesina[1], Csenge Antók[1], Boglárka Babcsány[1], Szabolcs Czifrus [1], and Attila Kiss[1]

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BME, as a consortium partner, takes part in the ECC-SMART project [1], the aim of which is to develop the pre-conceptual design of a supercritical water cooled small modular reactor (SCW-SMR). Among others, there are two working packages (WP), whose work focus on the thermal hydraulics (WP3) and neutron physics aspects (WP4) of this reactor type. These groups work in strong cooperation, as the concept is not only analysed separately, but also with coupled thermal hydraulics and neutron physics models and tools. Currently, we have three main fields of coupling within this project:

1. Apros – Serpent 2
2. Apros – SPNDYN
3. Ansys CFX – Serpent 2

Apros is a system thermal hydraulics code developed in Finland [2], Serpent 2 is a three-dimensional continuous-energy neutron and photon transport code also developed in Finland [3], SPNDYN is a finite element-based diffusion and SP3 code developed at BME by Boglárka Babcsány [4], and Ansys CFX is a high-performance computational fluid dynamics (CFD) software tool [5].

Models of the pre-conceptual core and reactor vessel design have been built with the above described tools, the coupling methods have also been set up and various coupled calculations have already been performed, but this work is still ongoing. The Apros – SPNDYN coupling is fully automated, using the built-in Phyton module of Apros and an additional Phython communication script developed at our Institute. Meanwhile, the two other couplings with Serpent 2, are semi-automated, which means that the two codes do not communicate directly with each other, but an indirect, offline data transfer is provided by Matlab scripts.

These coupled models have been used only for steady-state calculations so far. However, it is important to highlight that the SCW-SMR core concept uses the special supercritical water as moderator and coolant, and its flow path is quite complex and far from the usual, conventional setups, which means that even the steady-state calculation is challenging. To mention an example, the same water is used first as a moderator, which goes through the core two times, and then as a coolant, which goes through the core 7 times. Therefore, the coupled models are currently used to analyse the system behaviour and optimize the core concept in nominal operation states. However, we also plan to perform transient calculations as well in the future.

[1] ECC-SMART web page: https://ecc-smart.eu/.

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[5] Ansys CFX web page: https://www.ansys.com/products/fluids/ansys-cfx

P18

Tritium breeding in a Fusion Fission Hybrid Reactor concept

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The development of fusion-fission hybrid reactors (FFHR) [1,2], an old idea proposed together with pure fusion, has gained more consensus in recent years as an intermediate step before fusion energy production. An FFHR requires a reduced fusion power (about a hundred MW via D-T fusion reactions producing 14.1 MeV high energy neutrons) to drive a subcritical fissile blanket. By this process the applications are different: nuclear waste transmutation, fertilization of thorium or uranium for nuclear fuel, energy generation, tritium breeding for future fusion reactors, radioisotope production for medical application. The less stringent parameters of the fusion reactor in hybrid system respect to pure fusion ones and the availability of a controlled source of high energy neutrons to irradiate the high Z nuclear material make this solution very attractive.
One of the main issues regarding Fusion devices is the availability of Tritium for D-T reactions.
Tritium is a synthetic isotope of Hydrogen, and its supply is one of the main issues regarding future Fusion Reactors operation. It can be produced from Lithium through the reactions:

$^{6}Li + n\ (thermal) \rightarrow {}^{4}He + {}^{3}T$
$^{7}Li + n\ (fast) \rightarrow {}^{4}He + {}^{3}T + n$

FFHRs could have an important role in increasing Tritium breeding properties of pure Fusion devices since the neutron multiplication occurring inside the fission blanket could increase the neutron flux intensity and modify neutrons energy distribution, maximizing Tritium Breeding Ratio (TBR), defined as the ratio between produced and consumed Tritium masses.
We will investigate the possibility of producing Tritium in the fission blanket of an FFHR based on the coupling of available fusion and fission technologies. To do so, the neutronics and thermal hydraulics design of a dedicated test bed will be developed, taking inspiration from Lithium blanket proposed concepts for Fusion Reactors. Then, Lithium conversion properties of the test bed will be analyzed by FISPACT-II [3]. Results will be compared to those corresponding to Lithium breeding blankets, highlighting pros and cons of the Hybrid technologies.

[1] H. Bethe, The Fusion Hybrid, Physics Today, May 1979
[2] R. PIOVAN et al., “Pilot Hybrid Experiment with Reversed Field Pinch as neutron source and double test beds: an innovative stage approach towards a full power Fusion-Fission Hybrid Reactor”, Technical Meeting on Synergies between Nuclear Fusion Technology Developments and Advanced Nuclear Fission Technologies (Vienna, Austria 6-10 June 2022)
[3] The FISPACT-II User Manual, Editors: Michael Fleming Thomas Stainer Mark Gilbert, UKAEA-R(18)001 February 2018

P19

Advancement of GeN-Foam for CHF Prediction and Validation against the OECD/NRC PSBT Benchmark

Yuhang Niu [1, 2] , Scolaro Alessandro [2] , Carlo Fiorina [3], Hao Qin [1], Yingwei Wu [1], Wenxi Tian [1], G.H. Su [1]

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Numerical simulations occupy an increasingly vital position in the realm of nuclear energy design and safety. In pursuit of a more precise and efficient representation of reactor physics phenomena, the Laboratory for Reactor Physics and Systems Behaviour (LRS) at EPFL has embarked on the development of an open-source nuclear reactor analysis code, GeN-Foam, which is built upon the OpenFOAM. Notably, GeN-Foam has been extended to encompass simulations of water boiling conditions[1]. Recently, GeN-Foam has been augmented to facilitate the prediction of critical heat flux (CHF) by means of employing the CHF look-up table model. This paper describes such extensions and preliminary validation which are based on the OECD/NRC PSBT benchmark.

The benchmark exercises encompass two aspects: fluid temperature and steady-state rod bundle of departure from nucleate boiling (DNB). In the fluid temperature exercise, the disparity between the calculated and measured values is assessed using the error temperature, as illustrated in Figure 1. In the steady-state rod bundle DNB exercise, the GeN-Foam-predicted DNB power of select cases is compared against experimental and other computational results, as depicted in Figure 2. The GeN-Foam results exhibit errors within the 10% range when compared to the experimental data.

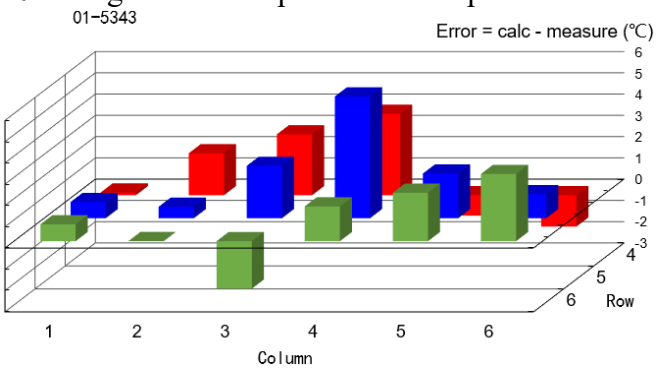

Fig. 1 case01-5343 of fluid temperature

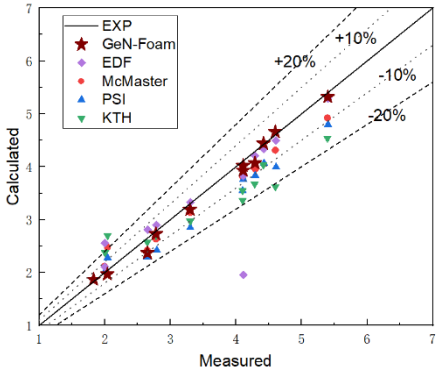

Fig. 2 DNB Power Results

Overall, this research showcases the advancements in GeN-Foam, providing valuable insights into its capability to simulate CHF and its preliminary validation against the OECD/NRC PSBT benchmark.

[1] Qin H, Fiorina C, Zhang R, et al. Extension of GeN-Foam to modeling of boiling water and validation against the OECD/NRC PSBT benchmark[J]. Nuclear Engineering and Design, 2023, 408: 112320.

P20

Poster Abstract

Monte Carlo radiation transport simulations have clearly contributed to improve the design of nuclear systems. Variance reduction methods are very important in this context, since they reduce the usual lack of tally statistics found in shielding or in-beam calculations. In this work we present a new approach by using intermediate surface sources with density estimation via the adaptive multivariate kernel density estimator (KDE) method. This concept was implemented in KDSource, a general tool for modeling, optimizing and sampling KDE sources, with a convenient user interface. The tool was used in a Monte Carlo simulation that modelled a neutron beam, showing good agreement with experimental results. Another simulation showed KDSource capability to couple optics and shielding simulations in a neutron guide system.

P21

Nuclear Data Influence on Integral Parameters in Sodium-Cooled Fast Models

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Nuclear data are a crucial aspect of a neutron transport simulation as a basic input parameter of a reactor system. These data are compiled in different libraries, and each library has different evaluations and processing codes applied, which leads to systematic errors.

Recent works [1, 2] show ENDF/B-VIII.0 systematically underestimates eigenvalues for water-cooled systems in depletion calculations compared to ENDF/B-VII.1, especially with high burnup (~60 MWd/kg) exceeding 500 pcm. Comparisons of the libraries [3, 4] for fast integral experiments demonstrate ENDF/B-VIII.0 generally underestimate eigenvalues while the results for ENDF/B-VII.1 represent a distribution around zero. Therefore, a different behavior of ENDF/B-VIII.0 can be expected for depletion in fast models too.

Following this, the work presents an assessment of nuclear data influence on functionals in the MET1000 and MOX3600 fuel assemblies, the sodium-cooled metal- and MOX-fueled models, respectively. The simulations have been conducted using OpenMC [5] with modern libraries prepared via OpenMC and NJOY [6] such as ENDF/B-VIII.0, JEFF-3.3, JEFF-4T2.2, JENDL-5, TENDL-2021, CENDL-3.2, BROND-3.1, and these have been compared to ENDF/B-VII.1. An ENDF/B-VII.1-based depletion chain has been used for all the cases.

As a result, although the integral experiments show an underestimation of eigenvalues, the simulation reveals the eigenvalues at the beginning are larger for ENDF/B-VIII.0 compared to ENDF/B-VII.1. The overestimation also can be generally seen in the Oak Ridge National Laboratory report [7]. The further comparison between ENDF/B-VII.1 and ENDF/B-VIII.0 reveals that the reactivity decreases faster using the latter library what is the case for light-water systems too. Despite this and a higher burnup, smaller discrepancies are present for the fast models. In addition, the other tested libraries such as JEFF-3.3/-4T2.2 and TENDL-2021 systematically overestimate the eigenvalues. The results for the reactivity coefficients are consistent including TENDL-2021 for both models. Considering the fuel composition, the largest deviations among the libraries for the actinides is present in the case of Am-243 and Cm. For fission products, Cs-134, Pm-147, and Sm-151 have the highest deviation. The deviations are also compared to the concentration uncertainties due to the nuclear data uncertainties obtained from the ENDF/B-VII.1 library using the random sampling approach via OpenMC.

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[2] K.S. Kim, W.A. Wieselquist, J. Nucl. Eng., **2**, 318-335 (2021).
[3] W. Zwermann, I. Hill, Ann. Nucl. Energy, **178**, 109354 (2022).
[4] O.N. Andrianova, et al., Izv. Wys. Uch. Zaw. Yad. En. **1**, 71-81 (2021).
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[6] R. MacFarlane, et al., LANL, LA-UR-17-20093 (2016).
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P22

Breed-and-Burn LFR core design

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[3]Laboratory for Advanced Nuclear Energy, Institute of Innovative Research, Tokyo Institute of Technology, 2-12-1-N1-19 Ookayama, Meguro-ku, Tokyo 152-8550, Japan

The global demand for energy is constantly increasing, and reliable, sustainable energy sources are urgently needed. Nuclear energy is one such source, and many thermal reactors are currently in operation. As a result, a huge amount of depleted uranium has been stored, and reprocessing plants or fast reactors operating in breed-and-burn (B&B) mode are necessary for effective utilization of uranium source. In this work, we investigated the feasibility of a Rotational Fuel-shuffling breed-and-burn reactor with Nitride fuel and Lead coolant with neutronic and thermal-hydraulic analyses. To establish a high neutron-economy B&B mode, the core is established with a Rotational Fuel-shuffling scheme. The present study confirms the feasibility of a small LFR design functioning in B&B operating mode by optimizing core height and operating parameters. We kept the maximum temperatures of fuel, cladding, and coolant under their constraints by performing a steady state thermal-hydraulic analysis for the hot channel of a shorter core with higher power.

P23

Sensitivity analysis is an essential step in validating nuclear safety codes like SIMMER-III. LIFUS5/MOD3 experimental campaign test E5.1 was conducted to improve the database for the system and boundary conditions to validate lead-lithium and water chemical reactions in the SIMMER-III system code.

The SIMMER-III code's predictions have been compared with experimental data from the LIFUS5/MOD3 test E5.1 for different critical parameter values to predict the reaction rate and its characteristics numerically and accurately. The reaction rates, loss coefficients, pressures, and temperatures of fluids are the most important input parameters for the LIFUS5/MOD3 test E5.1 simulation in SIMMER-III. Other important parameters include physical properties, boundary conditions, initial conditions, and model parameters. These could include physical properties, boundary conditions, initial conditions, and model parameters. Metrics of varying parameters used to quantify the impact of variations in input parameters on the code's predictions have been identified. Appropriate ranges for each input parameter are considered based on experimental data. These ranges should encompass both nominal values and potential variations that may occur during the LIFUS5/MOD3 test E5.1. The sensitivity analysis systematically varies the parameters within these ranges to assess their impact on the code's predictions.

Simulation results are analysed to assess the sensitivity of the code's predictions to the input parameters. The simulated results are compared with the experimental data from the LIFUS5/MOD3 test E5.1 to evaluate the code's accuracy and reliability. The sensitivity analysis results are interpreted to identify the most influential parameters and understand their impact on the SIMMER-III predictions. Compare the code's predictions with the experimental data to validate the code's performance. This analysis will help us gain insights into the strengths and weaknesses of SIMMER-III for the specific test case E5.1.

P24

Using open source nuclear codes to optimize the collimation delivery system of fast neutrons

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Abstract

In clinical practice, it is often necessary to concentrate neutrons emitted by a target in 4π space into a mono-directional beam. This process is crucial for increasing the particle flux density and optimizing the beam shape and cross-sectional area, while minimizing neutron absorption in structural elements. A collimator can be used to change the beam shape, significantly narrowing it and achieving optimal results. This paper presents simulation works using the MCNP5 code to investigate the feasibility of applying a narrow beam of fast neutrons, measuring 2 cm or less, in radiotherapy. The simulations were performed on the original design of an 8.5 x 8.5 cm2 collimator for treatment, located in the cyclotron laboratory of Tomsk Polytechnic University. The results show that the neutron energy spectrum remains nearly unchanged in the fast region, while the neutron flux increases by approximately 11% when using the collimator with a 2 cm aperture. The spatial distribution of fast neutrons is significantly narrower at a distance of 10 cm from the aperture compared to the original design of 8.5 x 8.5 cm2. The narrower and more intense neutron beam reduces damage to healthy tissue and decreases the treatment time, making the procedure more comfortable for the patient. Narrow beams offer the potential to make neutron beam radiotherapy safer and more accurate for the treatment of small and irregularly shaped tumors.

P25

In-vessel melt retention with external cooling roe VVER-1000

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[1](*Presenting author underlined)*

This article focuses on in-vessel melt retention with external cooling, a strategy critical for improving nuclear reactor safety during severe accidents. The study presents the calculation results of in-vessel melt retention with external cooling using a pressurized water reactor as a case study . The research includes a comparative analysis of two cooling strategies [[1]], one with a deflector and one without, to estimate the influence of the deflector on the results. The results were also compared with Melcor 1.8.6 code [[2]] to assess the reliability and validation of the BH models. The article discusses the limitations of the Melcor 1.8.5 code [[2]] and highlights the improved capabilities of the updated Melcor 1.8.6 code [[2]]. However, the lack of experimental data for certain heat transfer coefficients and the uncertainty associated with modelling severe accidents with different phenomena [[3]] still pose challenges for the accurate evaluation of external cooling of the reactor vessel. The study emphasizes the need for continued research in this field, taking into account foreign experiences and new results of experimental research.

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Coupling containmentFOAM and Modelica to tackle multi-scale challenges in SMR design and analysis

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The analysis of containment phenomena in SMRs is challenging, and has to take into account effects that present themselves at a large range of scales [1]. A proposed solution to tackle this multi-scale problem is to decompose it into multiple sub-domains, and solve these concurrently in a coupled approach (e.g. coupled CFD and system codes) [1, 2].
In this poster we present our approach to implementing a coupling infrastructure for containmentFOAM [3], an open source CFD code for containment analysis. An extension with the standardized and open co-simulation interface FMI [4] is discussed. Here, for improved simulation convergence, a semi-implicit calculation scheme is used. Furthermore, coupling with the system modeling solution OpenModelica [5] is shown, which is used to produce system thermal-hydraulic (STH) lumped-element models.
Based on the described simulation infrastructure, verification cases and their results are given for the implemented coupling between CFD and STH domains. Moreover, a technological demonstrator is presented: a pressure suppression safety system as employed in current light water-based SMR concepts.

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Benchmarking and validation of open source code for Fusion Neutronics studies

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OpenMC is an open source neutron and gamma particle transport code based on Monte-Carlo technique which utilizes constructive solid geometry and continuous-energy cross-sections [1]. Fusion neutronics requires extensive neutron and gamma radiation transport simulations for calculation of various nuclear responses related to radiological safety and machine design. Fusion neutronics simulations are performed with monte-carlo particle transport codes such as MCNP, TRIPOLI etc.

This work is carried out to perform the radiation transport simulations using OpenMC for benchmark problems so that it can be validated for fusion neutronics domain. SINBAD benchmark experiment conducted at ENEA for ITER shield blanket mock up is modelled in OpenMC, neutron and photon flux spectra are compared with experimental results as well MCNP calculations. The results show good agreement and capabilities of OpenMC to use in further neutronics calculations at the same par as MCNP.

[1] Benchmarking and verification of the OpenMC code for accelerator-based neutron source analyses Yuan Hu, Yuefeng Qiu , Ulrich Fischer , Yudong Lu , Fusion Engineering and Design, Volume 170, September 2021, 112512

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Preliminary Extension of OFFBEAT to TRISO Fuel

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TRISO(Tristructural Isotropic) fuel is a type of advanced nuclear fuel designed for high-temperature gas-cooled reactors. It is composed of small spherical fuel Kernels encapsulated in multiple layers of coatings. These layers provide a robust defence against the release of radioactive materials even under extreme conditions, which make TRISO fuel highly attractive for next-generation nuclear reactors due to its enhanced safety features and improved fuel performance.

To simulate the behavior of TRISO during reactor operation, the multi-dimensional fuel performance code OFFBEAT(OpenFOAM Fuel BEhavior Analysis Tool)[1] has been extended for the TRISO case. The material properties of porous carbon buffer, PyC(Pyrolytic Carbon) and SiC(Silicon Carbide) from PARFUME[2] have been implemented into the code, as well as their behavior models, including anisotropic irradiation dimension change, thermal expansion and creep. The original UO_2 models already present in OFFBEAT were used to simulate the material properties, densification and swelling of Kernels. Additionally, a dedicated gap plenum model was used to account for the gap formed by the shrinkage of buffer and PyC, as well as buffer's porosity. The Proksch model[3] was utilized to compute CO(Carbon Monoxide) produced by the reaction of oxygen and carbon. By incorporating the gap plenum model, CO production model and mechanism model of fission gas release, the gap behavior, internal pressure and heat transfer between buffer and inner PyC can be simulated.

After the extension of new models, a 1-D model of a TRISO particle has been built with wedge boundary conditions along transverse directions and it has been used to verify the code against the IAEA CPR-6 benchmark cases. The results are in line with the predictions of other codes, suggesting that OFFBEAT could be a promising tool to accurately simulating the performance of TIRSO under normal conditions. Currently, efforts are focused on 3-D simulations of TRISO. Future developments will include studying the diffusion of fission products, debonding between different layers[5], as well as the multi-dimensional and multi-scale coupling between TRISO particles and the pebble bed.

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Development of a novel subchannel code for wire wrapped subassemblies using FVM method based on OpenFOAM

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In order to realize the potential for finite volume method in the numerical area, it will be beneficial for build a more unified framework for both multi-physics and multi-scale analysis for nuclear reactor. And traditional subchannel analysis for reactor core gives both accuracy and speed, but it has less robust in some numerical conditions [1]. Therefore, a novel subchannel code using FVM method and corresponding discretization schemes within the OpenFOAM [2]. OpenFOAM is the free, open source CFD software. It's very mature in the simulation of CFD scale, but it also has ability to develop your own code.

In this work, we present the methods how we implement subchannel scale analysis for wire wrapped subassemblies based on OpenFOAM framework using FVM methods. We modify the mesh model, derivate the equations and implement closure model for wire-wrapped assembly. We inherit the solution scheme, discretization schemes, matrix solve method and parallel scheme in OpenFOAM, which save lots of time in developing new code. We validate the code using experimental results covering different pin numbers, different Reynold number and skewed or uniform power distributions. The results agree well with experiments and traditional subchannel code. This novel subchannel code shows an attractive and potential in the subchannel analysis of SFR

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Simulations of sodium spray fire accidents using analysis code NAFCALX

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Sodium fire, which includes sodium spray fire, sodium pool fire, and mixed fire, is one of the most significant phenomena that can occur in Sodium-Cooled Fast Reactors (SFR). It can cause temperature and pressure rises, as well as aerosol generation in the containment. To study sodium fire, the analysis code NAFCALX has been developed by establishing models for sodium droplet combustion, sodium droplet grouping, heat transfer, and sodium motion. By providing information such as containment volume, sodium leakage rate, transfer information, droplet distribution, and others, NAFCALX can predict the energy released by sodium spray fire, as well as the temperature and pressure peaks. Furthermore, it can provide detailed output regarding burning droplets, including droplet size, velocity, and mass. The code has been used to simulate sodium spray fire experiments such as FAUNA, SNL T3, ABCOVE, and KAERI.

The Korean Atomic Energy Research Institute (KAERI) has conducted a series of sodium spray fire experiments in a specially designed sodium fire experimental facility with a volume of 1.7 m^3. The leakage height is 1.80 m, and the mass of sodium injected is 120 kg. The initial sodium temperature ranges from 478.15 K to 603.15 K, and the spray duration is 6 seconds. The average droplet size is 0.06 cm.

The Hanford Engineering Development Laboratory (HEDL) has conducted the ABCOVE experiment in the Containment System Test Facility (CSTF), which has a volume of 852 m^3. For the AB5 test, the mass of injected sodium was 222.8 kg, and for the AB6 test, it was 204.7 kg. The initial oxygen concentration, initial sodium temperature, and initial gas temperature were similar for both tests. However, the AB6 test had a much longer duration (4780 seconds) compared to the AB5 test (872 seconds). The average droplet sizes were 0.103 cm for AB5 and 0.064 cm for AB6.

Germany has established the FAUNA facility to conduct a series of sodium spray fire experiments under different conditions. The container's volume is 220 m^3, with a height of 7.78 m and a diameter of 6 m. Sodium fire experiments were conducted with different total leakage sodium, recorded as FS4, FS5, and FS6 experiments, while maintaining the same sodium leakage rate of 57 kg/s. The masses of sodium sprayed were 30 kg (FS4), 40 kg (FS5), and 60 kg (FS6). An average droplet size of 0.2 cm was used.

Sandia National Laboratories (SNL) conducted the T3 sodium spray fire test in a 99 m^3 Surtsey large-scale steel container. The sodium spray had a leakage height of 5.3 m, and the temperature of sodium droplets in the spray was 473.15 K. The initial air temperature in the air space was 288.15 K, with an initial air pressure of 101.3 kPa and an initial oxygen concentration of 23.3%.

The predicted temperature and pressure peaks, as well as the trend of temperature change over time, were in good agreement with the experimental values. The calculation results also included temperature changes at different locations along the vertical height of the containment. It was observed that the location of the temperature peak primarily depended on the initial sodium temperature and the initial falling height. Using the code, the influence of droplet size changes and droplet position on burning rate calculations has been studied. Additionally, the code can calculate the proportion of the unburned sodium in the total sodium leaked. The formation of sodium pool fire is an important factor when simulating the sodium spray fire accident.