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Benchmarking the HENDL-3.0 data library by simulating a sodium-cooled test reactor

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ABSTRACT

Advanced computation to deal with sophisticated nuclear systems and their process mainly relies on competent nuclear data libraries. The library could significantly alter the simulation results. In this study, the competence of a Hybrid Evaluated Nuclear Data Library, called HENDL, has been strengthened by employing it in performing calculations for integral neutron physics parameters. For this purpose, SuperMC code was used to model and simulate an SFR (Sodium-cooled fast reactor), BFS-62-3A – the critical experiment performed at BFS2 facility in Russia. The critical assembly in its full heterogeneous configuration containing four fuel-zones was modeled. The calculations were performed to estimate the criticality, reactivity effects, spectral indices and fission rates in radial direction. The sensitivity study was also carried out to see the influence of any change by the experimental uncertainties in the material data described in the critical experiment. The accuracy and trustworthiness being the vital properties that any neutron data library should possess were thus testified and proved thereafter for the aforementioned fast spectrum advanced reactor. The results so obtained were in good agreement with the experiment and the study hence enabled the HENDL library to be validated/benchmarked.

1. Introduction

Future of the global nuclear sector is mainly dependent on achieving a sustainable nuclear energy's development (World Energy Resources Report, 2016). This could happen by setting a goal to obtain highest possible nuclear safety and by making an optimization of the utilization of uranium resources (Wu et al., 2016). The advanced nuclear reactors, including the ones with fast neutron spectrum and metal coolant, under the theme of Gen IV concept (Igor, 2016), are gaining due weightage as they are capable to deal with both of the above-stated issues i.e. Utilization of natural uranium is possible in fast spectrum and the selection of coolant (liquid-metal coolant in our case) in fast reactors is one of the crucial aspects that directly relates to design, safety, economic and technical characteristics of the nuclear system.

The development and design phase of nuclear systems is supposed to meet the high standards of reliability, efficiency and safety. These integral factors being very costly in practical scenario cannot be compromised. Before any practical implementation, modeling and simulation using nuclear tools/software are hired to attest the accuracy of any nuclear data for which benchmarking of a data library is inevitable (Liem, 2012). The BFS-62-3A, a test reactor (or a critical assembly), is a full-scale model of Russian BN-600 hybrid core reactor that has been in employment to predict, verify and validate various nuclear tools/software (Manturov et al., 2006). The critical assembly, being one out of other zero-power critical assemblies maintained at BFS-2 critical facility, is the most convenient and feasible configuration with multi-zoned core for prediction of neutronic parameters to test the accuracy of and validate nuclear software systems including the codes and data bases. In their studies many researchers have used BFS-62-3A critical experiment, including Jamil Z. et al. to validate a Monte Carlo code (Jamil, 2018), SuperMC; Hazama T. et al. to verify a nuclear analysis system for fast reactors (Hazama et al., 2004); Rachamin R. et al. to validate the DYN3D-Serpent code system for an SFR core (Rachamin and Kliem, 2017); etc.

To ensure the simulation results to be reliable, presence of accurate data library according to the specific nature of a problem is essential. The HENDL point-wise data library is one of the evaluated nuclear data libraries that has been employed in the design and development of various nuclear fusion and fission based

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projects. The library has been developed in different problem-specific sub-versions (or sub-libraries) and is in continuous improvement process, the detailed introduction can be seen in succeeding section. The testing, validation and application of the HENDL has been carried out in many research works. For instance; Zou J. et al. designed and developed HENDL/FG that is a 1200 fine-group nuclear data library for advanced nuclear systems (Zou, 2012); Chen C. et al. conducted a testing and application of continuous energy neutron cross-section library HENDL/MC (Chen, 2013); etc.

In this study, it is intended to further the benchmarking of HENDL/MC-3.0 data library using a MOX-fueled sodium-cooled fast reactor (SFR) – the critical experiment BFS-62-3A fetched from the International Reactor Physics Experiment Evaluation project (IRPhE). It involves a prediction of neutronic parameters using Super Monte Carlo (SuperMC) code. The sole purpose for this validation/benchmarking is to testify its competence in prediction and analysis of phenomena in a fast spectrum sodium-cooled reactor like BFS-62-3A. This is an effort to improve and have the accurate nuclear data obtained that would be available for researchers/experts to fulfill the needs in the developmental studies of nuclear systems. Further it is noteworthy that in this paper we further investigated some of the findings of our previous work [published article in references at (Jamil et al., 2018)] while rest of the contents in this paper are new and could be considered as a complement of chain/sequence of our investigating work, as a whole, that was performed employing the BFS test reactor.

The contents in the paper are set up as follow: section 2 describes to the readers an adequate knowledge of HENDL library, an overview of the BFS-62-3A critical experiment and a brief introduction of the SuperMC code. While section 3 deals with the methodology used, results obtained and respective discussion on the measurements performed during benchmarking simulations. Next two sections including 4 and 5 are reserved to express conclusions and acknowledgement respectively while references are provided in the last section.

2. The data library, critical experiment and the MC code

This section deals with the description of HENDL neutron data library used in this study, the BFS-62-3A critical assembly and the Monte Carlo code employed to perform modeling and simulation.

2.1. Hendl-3.0

Hybrid Evaluated Nuclear Data Library, called HENDL, is a comprehensive, reliable and accurate neutron data library developed by FDS Team at INEST, CAS in China (Zou, 2010; Xu, 2009). To cope with the specific-natured problem in nuclear application, the library has four sub-versions; including HENDL/MC, HENDL/MG, HENDL/CG and HENDL/FG. With an energy group structure of 366-neutron/42-gamma, the HENDL/MG is multi-group sub-library; with an energy group structure of 27-neutron/21-gamma, the HENDL/CG is a coarse-group sub-library; and with an energy group structure of 1200-neutron/42-gamma, the HENDL/FG is a fine-group sub-library that could be used up to 150 MeV.

In this work, we used HENDL/MC (or HENDL-ADS/MC) that is a point-wise data library (ACE format) and could be employed by a Monte Carlo code. This library was primarily developed to perform nuclear analysis of an Accelerator-Driven Subcritical system (ADS). With 408-nuclide cross-section files for actinides, fission products, and structural materials; the library is capable to deal with the neutrons of energies up to 150 MeV. The library has been in employment to be testified in various critical safety benchmarks. The high neutron energy cross-sections of the library have been validated by performing a series of high-energy neutron shielding experiments using an indigenously developed system called VisualBUS – A system for multi-functional neutronic calculation and analysis. Chen C. et al. in their study have carried out a testing and application of HENDL-ADS/MC (Chen, 2013).

Another sub-version of HENDL, called HENDL-ADS/MG that is based upon the evaluated data of TENDL-2009, JENDL-He, has been developed and tested by (Zou, 2010) to deal with the energies as higher as 150 MeV. They have validated the library using JAEA's 800 MW ADS benchmark for critical safety and shielding analysis.

2.2. The critical experiment – BFS-62-3A

Associated with the Institute of Physics and Power Engineering (IPPE) in Russia is a full-scale critical facility called BFS-2 (Big Physical Facility-2). This is one of the largest critical test facilities, in operation, in the world that allows a flexible and convenient working with various available simulated coolants including gas, water, Na, Pb and Pb-Bi. The full-scale simulations on a variety of fast-reactor cores up to 3 GW (electric) could be performed with a desired choice of core-blanket configurations. In this study, one of the most attracting critical assemblies – the BFS-62-3A – was selected that contains MOX-fuel, sodium coolant, UO_2 blanket and stainless steel reflector. Being a full-scale model of hybrid-fueled BN-600, the critical assembly was primarily hired to experimentally carry out the investigation to estimate the nuclear physics parameters and their respective uncertainty in the conversion of conventional BN-600 design in to a MOX-fueled design with partial stainless steel reflector (replacing 1/3 or the UO_2 blanket of the former design). With a hexagonal pitch of 5.1 cm, the critical assembly BFS-62-3A is composed of over 9000 cylindrical stainless steel tubes vertically installed on to an SS (stainless steel) plate (diagrid). Each tube with an outer diameter of 5 cm, wall thickness of 0.1 cm has a length of 317 cm. It is important to note that space between the tubes is filled with stainless steel dowels round in shape, 0.8 cm in diameter and about 310 cm in length. The inter-tube space could also accommodate small-sized detectors to measure the reaction rate distributions. The core with a diameter of 200 cm and a height of 100 cm is divided in to four zones in accordance with the type and enrichment level of the fuel elements as shown in Fig. 1. These are Low, Medium, Plutonium and High Enrichment Zones referred to as LEZ, MEZ, PEZ and HEZ respectively. Mirror symmetry is observed in the fuel cells of fuel zones with reference to the core's mid-plane. Attached to the core, the blanket in radial and axial directions extends from 50 to 70 cm. Core contains mock-ups of 6-SRs (safety rods) and 18-CRs (control rods) in its LE Zone such that 12 CRs are placed near to the border of MEZ and LEZ zones.

The assembly of the zero-power critical reactor contains stainless steel reflector. The reflector, being one-third of the blanket in 120° sector as depicted in Fig. 1, contains standard stainless steel tubes, each filled with an average of 24 SS-dowels. About 1100 empty tubes surrounding the assembly are installed in the area between the core and the vessel. Core contains two kinds of sodium pellets, laser and green, where the laser ones (also called "new") lying inside the 120° sector are free from hydrogen content and the green ones (also called "old") lying outside the 120° sector contain some hydrogen content (Manturov et al., 2006). Similarly, the U36% being one of the fuel elements in the form of pellets is in three qualities - brig, new and old - where brig-quality pellets are present in outer fuel-tubes installed in Low and High enrichment zones; new-quality pellets in HEZ; and old-quality pellets in LEZ. Within the core, the other fuel materials are Pu95% placed in PEZ only, U90% placed in HEZ and MEZ and UO_2-36% placed in LEZ only.

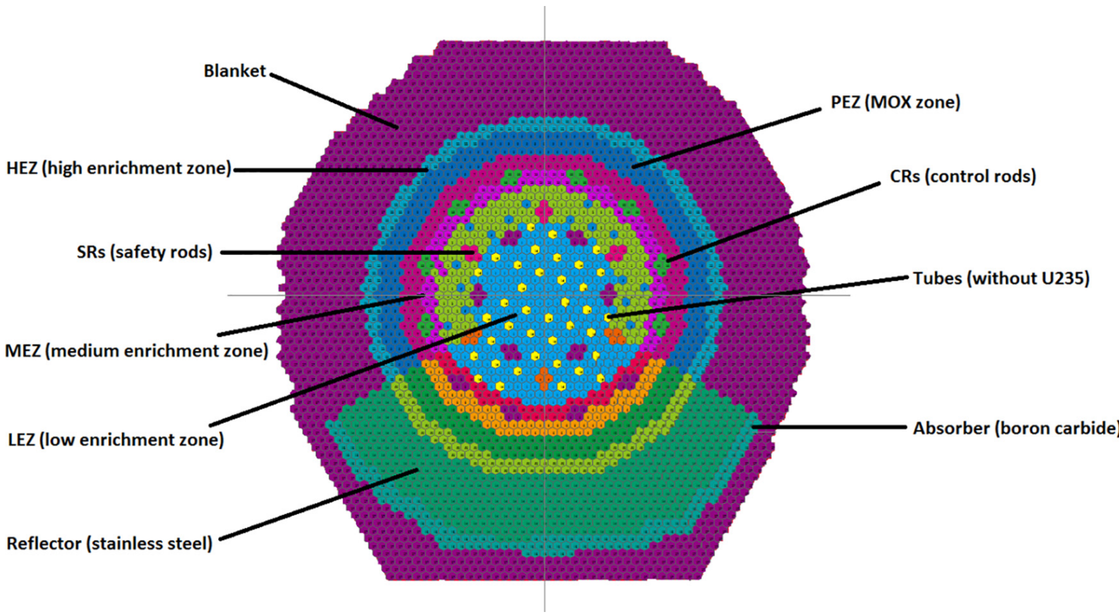

Fig. 1. A SuperMC's modeled-layout of BFS-62-3A assembly.

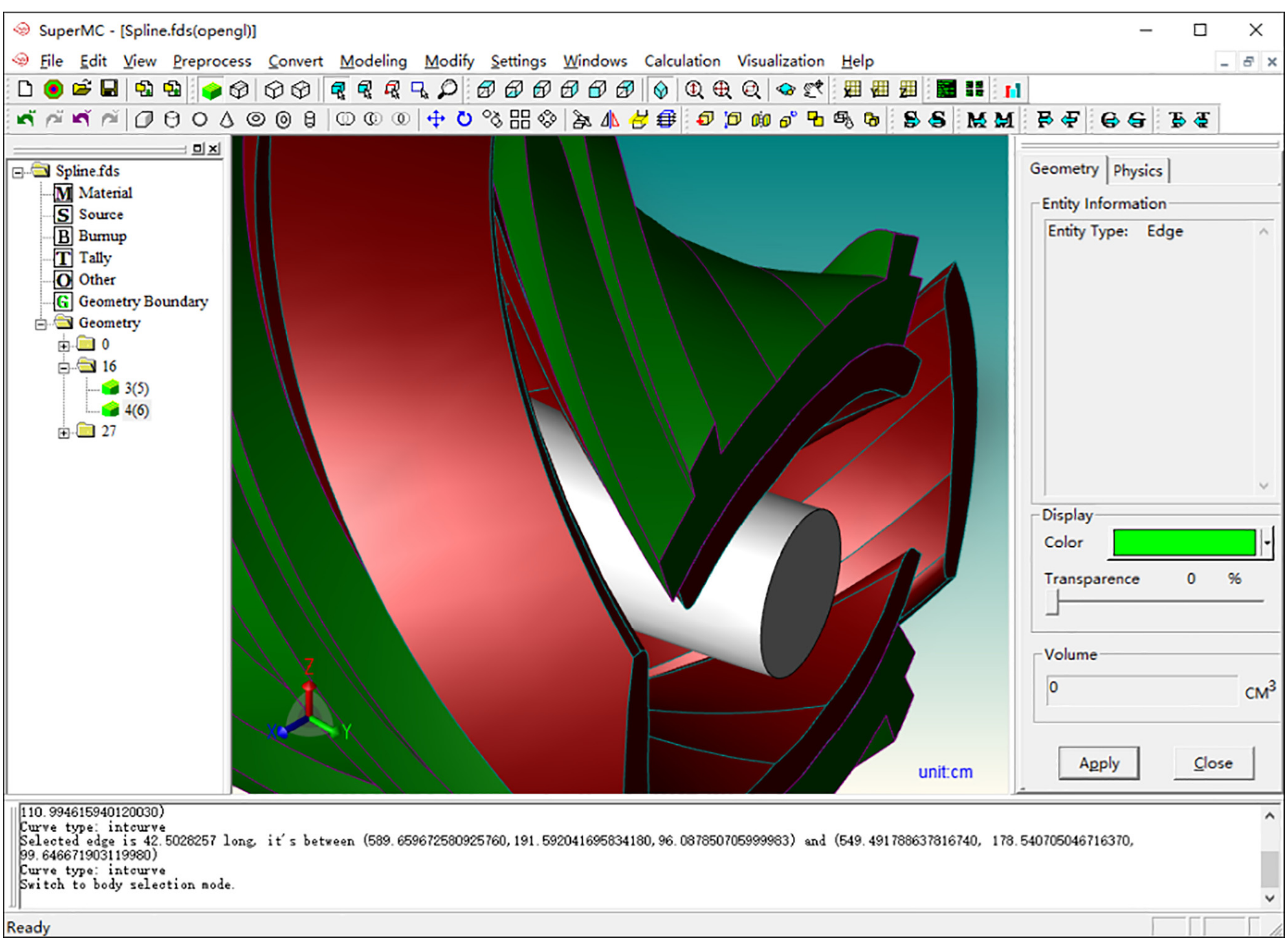

Fig. 2. A typical GUI of SuperMC.

2.3. An overview of SuperMC code

SuperMC being a Monte Carlo code, acronym for "Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation", is a program designed for nuclear and radiation simulation. It is accurate, intelligent and precise software to perform general-purpose integrated simulations for a variety of nuclear systems. The code, developed by FDS Team (INEST, CAS, China) is virtually self-sufficient in creating CAD-models without relying on other CAD software (Wu et al., 2015; Wu and Team, 2009). Diversified modeling features of the code are available to help users deal with complex 3D-geometries, parameterized hierarchical-structure, etc. A GUI (Graphical User Interface) of SuperMC is shown in Fig. 2. The code can do geometry-, material-, source-, process-, tally modeling, etc. To entertain users, calculations could be performed in different convenient ways that include serial-, parallel-, cloud- and FISPACT coupling calculations.

The SuperMC code has been applied in many projects dealing with the nuclear reactor design and safety evaluation studies – Particle transport calculations rendered very accurate results. Advanced nuclear systems based on fission, fusion and fusion-fission hybrid; including China LEAd-cooled Reactor (CLEAR), International Thermonuclear reactor (ITER), etc. have employed SuperMC for nuclear engineering design and analysis (Zhang, 2016).

The code has proven skills and credibility after being predicted, verified and validated through over 2000 international Benchmark tests and experiments. Numerical verification, for instance, has been performed employing IRPhEP, ICSBEP, BEAVRS, IAEA-ADS, TCA, etc. for reactor core physics (Wang et al., 2018); benchmarks from AAPM55, etc. for medical physics; SINBAD shielding benchmark, ITER reference models, FDS series fusion reactors, etc. for radiation shielding. SuperMC code has been/is in application by many international nuclear projects like International Fusion Materials Irradiation Facility (IFMIF), Fusion Nuclear Science Facility (FNSF), Fusion Demonstration Reactor in European (DEMO), High Power Laser Energy Research (HiPER), Korea Superconducting Tokamak Advanced Research device (KSTAR), Experimental Advanced Superconducting Tokamak (EAST), etc. More than 600 institutions around the globe are presently using the code that shows its good credibility and worth.

3. Monte Carlo simulations for benchmarking

In this section, we describe the detailed neutronic calculations performed for the estimation of criticality and spectral indices, reactivity effects including sodium void reactivity effects (SVREs) and control rod worths (CRWs), and reaction rates in the core and blanket in radial direction. Computational resources included 80 CPUs, SuperMC-3.1 code and HENDL-3.0 cross-section nuclear data. All the calculations were performed at room temperature (293 K), and the calculation run, except for reactivity effects measurement, employed 5.0×10^5 particles per cycle keeping 200 initial cycles skipped with a total of 2000 cycles. The prediction of reactivity effects employed 5.0×10^4 particles per cycle (skipping 200 initial cycles for a total of 2000 cycles). The β_{eff} that we obtained during calculations is 0.00602 (the same value was used to obtain the reactivity worth values in cents which were then compared with the experimental data).

As to the terminology used above related to Monte Carlo (MC) calculations; it is to mention for better understanding of readers that in "MC neutron transport" the neutrons are followed in a defined geometry and the interactions that they make with media are recorded. MC simulations move on generations (or cycles) wherein neutrons are generated and followed (one-by-one). Thus the neutron histories are contained by each cycle. The logic behind skipping the adequate number of initial cycles (called inactive cycles) is that the source, at the starting of simulation, might be away from "stationary converged distribution". So the simulation statistics is gathered/recorded in active cycles that start after the skipped cycles end and continue until the attainment of desired accuracy level.

3.1. Criticality measurement and spectral indices

3.1.1. Criticality

To reduce the calculation uncertainty, a good combination of number of neutron histories (5.0×10^5 particles per cycle) and available computational resources was used as stated in the beginning of section 3. The calculated effective multiplication factor is 1.0063 with a statistical uncertainty of 0.00002 while the experimentally measured value of k-eff is 1.0008 with an uncertainty of 0.003. The calculation using HENDL library slightly over-estimates the k-eff by 0.55% which is virtually a good agreement between the simulation and experiment. Just to give an idea, the k-eff obtained with ENDF7.1 library was 1.0044 with a slight over-estimation of 0.3% (as predicted by our previous work – see (Jamil et al., 2018)).

3.1.2. Spectral indices

At the centre of the core, spectral indices were measured by employing the fission chambers (small in size) in the experiment. Spectral indices being the ratio of fission rates (or cross-sections) are an important parameter to assess the breeding performance of a fast spectrum reactor i.e. ^{238}U turns to ^{239}Pu when exposed to fast neutrons. The ratio of ^{239}Pu fission rate to that of ^{235}U (termed as F9/F5) and the ratio of ^{238}U fission rate to that of ^{235}U (termed as F8/F5) are the desired dimensionless spectral indices that were calculated and the results are shown in Table 1.

The quantitative deviation of calculated results predicted by HENDL's cross-sections is 0.27% and 1.38% for F9/F5 and F8/F5 respectively. The experiment document has stated of some typical uncertainty values 1–1.5% for F9/F5 and 1.5–2.5% for F8/F5 (because of the errors in exact isotopic composition of the three isotopes and of thermal fission cross-sections). So, by taking the margin of experimental uncertainties in to account, the simulation predicted the spectral indices in close agreement with the experiment.

3.2. Reactivity effects measurements

3.2.1. Sodium void reactivity effect (SVRE)

The core should be having a negative sodium void reactivity that is considered one of the most important features for the safety of a sodium-cooled fast reactor (Tiberi, 2010). So, any reactivity shift/change of the critical assembly should be noticed and predicted precisely. In the BFS-62-3A experiment, the instrument called a "reactimeter" that includes three neutron detectors (BF_3 filled ionization chambers) was installed in the blanket region (radial) at about 30–40 cm from the boundary of the core (Manturov et al., 2006). To predict and analyze sodium void reactivity effects (SVREs) of the critical core, the sodium pellets were removed from the defined sub-regions of Low-, Medium-, Plutonium- and High Enrichment Zones i.e. the central conical part (60° sub-sector) of the key-region (120° sector) was voided with sodium-laser in radial direction. The sodium was also removed

Table 1
Spectral indices at the core centre: Measured and Calculated results.

Spectral indices	Experiment "E"	Simulation "C"	C/E
F9/F5	0.9370 ± 0.015	0.9345 ± 0.0116	0.9973
F8/F5	0.0202 ± 0.0004	0.0205 ± 0.0005	1.0149

from the upper axial blanket (UAB). Terminology could be apprehended by the readers that before removing the sodium, the core is unperturbed while it would become perturbed (the reactivity would become changed) after the removal of sodium and this perturbation eventually lead to the prediction of SVRE.

It is to note that 259 FRs (Fuel Rods) in all were voided from sodium with such a division: LEZ 118 FRs (52 pellets per FR); MEZ 37 FRs (50 pellets per FR); PEZ 58 FRs (36 pellets per FR); and HEZ 46 FRs (46 pellets per FR). The removal of sodium-laser pellets was carried out in a particular sequence; the LEZ was voided first, then MEZ, PEZ and HEZ respectively such that the previously voided sub-zone was kept voided i.e. the absence of sodium from the core was increased gradually. The calculated and experimental values for total reactivity effect (in cents) after voiding all four zones were −20.0 ± 0.8 and −28.4 ± 1.5 respectively. It can be seen that HENDL underestimates the SVRE by about 8 cents with a deviation of 29%. Similarly, the SVREs for the four zones from LEZ through HEZ were predicted to be −4.54, −1.33, −2.8, and −11.36 (all in cents) respectively. Since the higher deviations (in total effect and for some of the individual zones) have already been observed by Monte Carlo (SuperMC, MCNP, and Serpent) and deterministic codes (DYN3D) with different neutron cross-section libraries; as reported by other researchers like (Jamil, 2018), (Rachamin and Kliem, 2017), (Ivanov and Bousquet, 2016) and (Marinoni et al., 2012); so the predictions here in this work seem reasonably justified. Another point is that the sequence of voiding the zones if gets changed may affect the reactivity effects on individual and cumulative values [like if HEZ is voided first and then PEZ, MEZ and finally the LEZ (or any other sequence)]. It may lead to make further investigation that currently seems being out of scope for this work.

In addition to the above-stated argument, any minor error (by the extent of experimental uncertainties, for instance) in estimation of density, dimension and isotopic composition of the core may lead to receive the large discrepancies in the calculations as the experimental value of SVRE itself is significantly smaller in magnitude. Similarly choice of a data library is also a means to affect the simulation results.

3.2.2. Control rod worth (CRW)

Without introducing any control or safety rods (CRs or SRs), the core was classified as "unperturbed". As the CRs were introduced, the reactivity got changed and the status of the core became "perturbed". For evaluating the control rod worths (CRWs), the experiment defined seven positions, all lying inside the inner core/LEZ (both inside and out of the key region). It is to note that before performing this kind of control rod worth measurements, mock-ups of the control and safety rods are already there in the core. To measure CRW, some kind of control and safety rods containing absorber part and B_4C pellets (facing the effective fuel cells' height of the fuel rods in core) were used and introduced in the core; the readers may refer to the BFS-62-3A benchmark document for further details (Manturov et al., 2006). The results, being dependent on the sequence of loading the control (or safety) rods at defined positions of the core, are tabulated in Table 2.

Table 2
Control rod worths: Measured and Calculated results.

Control/safety rods	Experiment "E" (cents)	Simulation "C" (cents)	C/E
SR-2-2	−95.0 ± 1.4	−103.2 ± 0.4	1.09
SR-2-5	−87.8 ± 1.3	−92.4 ± 0.5	1.05
CR-1-3	−54.8 ± 0.8	−52.3 ± 1.0	0.95
CR-1-6	−54.8 ± 0.8	−51.6 ± 0.6	0.94
CR-3-5	−45.8 ± 0.7	−50.5 ± 0.8	1.10
CR-3-11	−45.3 ± 0.7	−44.9 ± 0.5	0.99
Centre	−56.3 ± 0.8	−56.4 ± 1.4	1.00

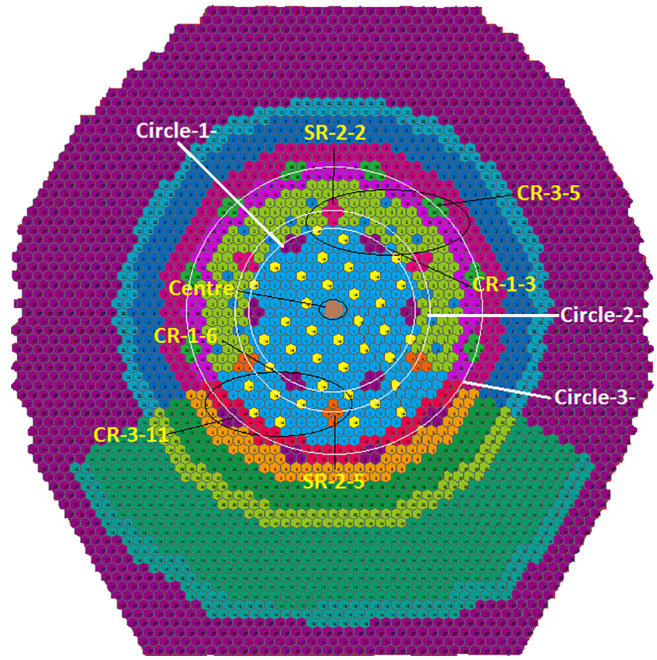

Fig. 3. Seven designated positions in the core for CRW measurement.

Fig. 3 depicts a clear view on the seven positions inside the core that were used to measure the control rod worths. As is obvious, four positions including centre, CR-1-6, SR-2-5 and CR-3-11 lye inside the key region while rest of the three are out of the key region. The calculations for both SR-2-2 and CR-3-5 show a bit higher discrepancies with an overestimation of around 8cents and 5 cents respectively with reference to the experimental data. In simulation, these two control and safety rods offer over-absorption of particles (neutrons) that brings more rod worth to the designated positions.

The SR-2-5, positioned in key region (in an alignment facing the SR-2-2), and CR-3-11, also positioned in key region (in an alignment facing the CR-3-5), both predicted the CRWs to be in close agreement with the measured data. The difference in the level of deviation of the predicted values for both of the safety rods, for instance, might be because of their presence in different core parts i.e. one lies in the key region containing sodium-laser while the other lies out of the key region containing sodium-green. The same has been described in the benchmark document and reported by other researchers; including (Jamil, 2018) and (Marinoni et al., 2012). It is of interest to notice that both CR-1-3 and CR-1-6, lying in laser-zone almost at the same distance from the centre in oppositely symmetrical direction, slightly underestimate the CRWs by 2 cents and 3 cents respectively. This shows under-absorption of particles (neutrons) that eventually brings less rod worth to the designated positions defined at the inner circular ring (White-colored circular rings have been marked in the Fig. 3 for the better understanding of the readers; where -1- is the inner most ring, -2- is middle ring and -3- is outer ring with the largest radius). The calculations agreed well with the measured data within 5.1% on the average that conferred a good match of results by employing HENDL library.

3.3. Fission rates

In the BFS-62-3A experiment, small pin-type fission-chambers (cylindrical in shape) were employed to measure the fission rates (also called reaction rates) for the three isotopes; including ^{239}Pu, ^{235}U and ^{238}U throughout the whole assembly. Through some manipulators, the fission chambers were inserted in to the inter-tube spaces and placed at the required height level to carry out the measurements within a cell (an interval of 0.5–1.0 cm was selected within each cell for measurement and the result was averaged over the whole cell). All of the radial fission rate distributions (RFRDs) were measured at 19 defined positions in the key region only (starting at 2.9 cm from core centre heading to the reflector ending at 144.3 cm). The fission rates in core and blanket region in radial direction were normalized to the reaction rate at the centre of the core and the predicted simulation results along with experimental data are plotted in graphs as shown in Figs. 4–6.

It is worth mentioning to note that all fission rates are taken for a 5% decreased density of stainless steel reflector. It is because we inferred from our previous study on the same BFS critical assembly that a 5% decrease in reflector density gave good simulation results (Jamil, 2018).

As can be noticed from Fig. 4, the fission rates for ^{239}Pu predicted by the code are stringently in agreement to the experiment. The average deviations of 2.1% in the core and 4.0% in the whole assembly come up with the calculation.

As can be observed from Figs. 5 and 6, the fission rates for ^{235}U and ^{238}U predicted by the code also agree well with the experiment. The simulation is responsible for average deviations of 2.2% in the core and 3.5% in the whole assembly for ^{235}U while

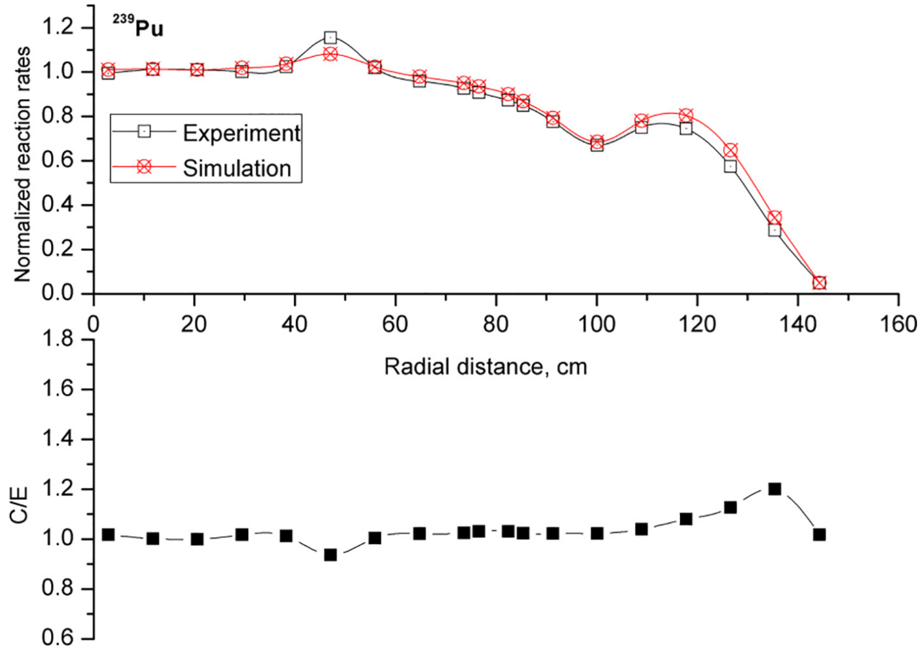

Fig. 4. The fission rates for ^{239}Pu in radial direction, normalized to unity at R = 0. The calculation errors are 0.4% and 1.6% in core and reflector region respectively.

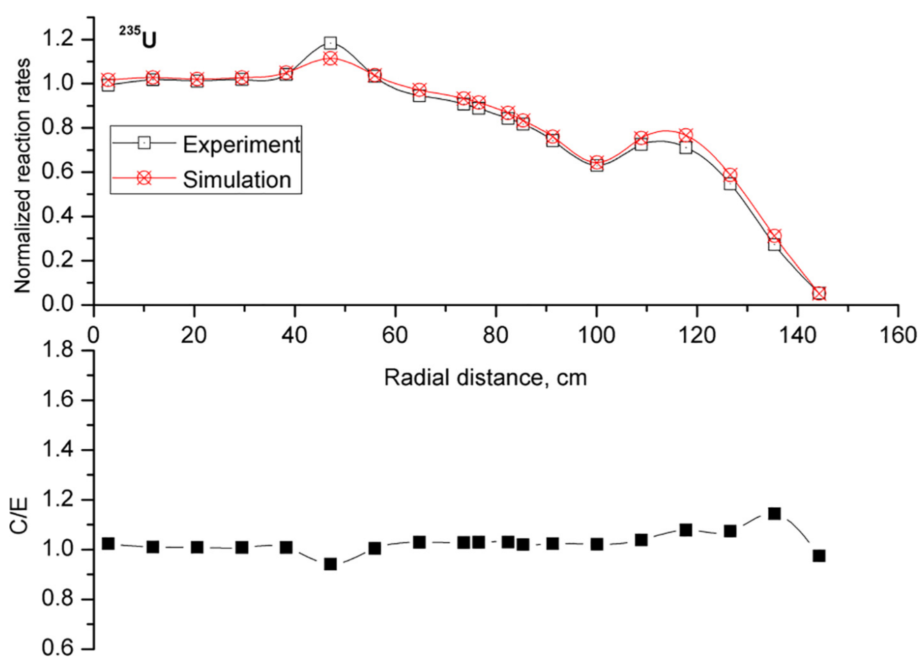

Fig. 5. The fission rates for ^{235}U in radial direction, normalized to unity at R = 0. The calculation errors are 0.4% and 1.0% in core and reflector region respectively.

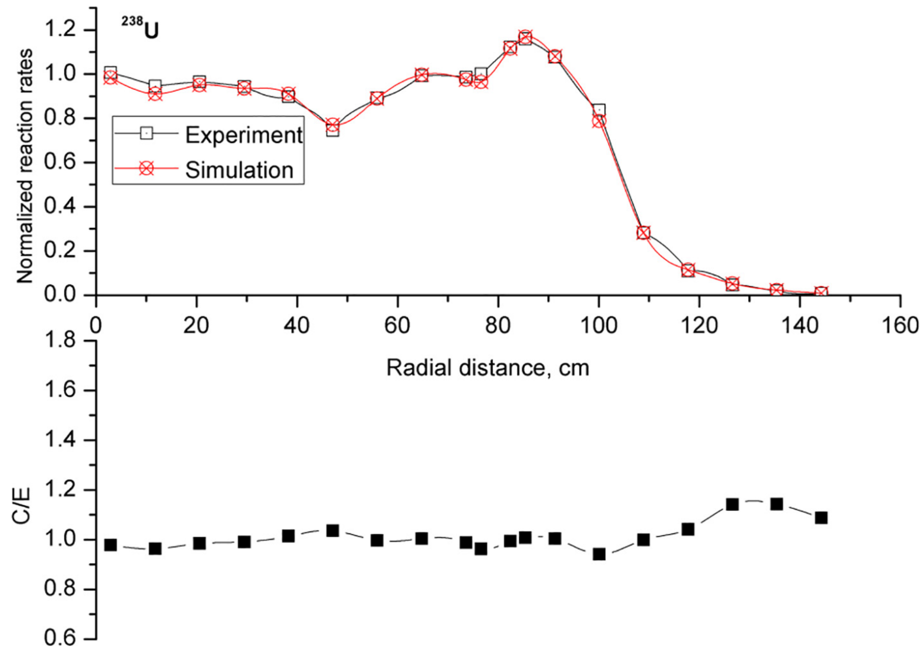

Fig. 6. The fission rates for ^{238}U in radial direction, normalized to unity at R = 0. The calculation errors are 1.0% and 3.7% in core and reflector region respectively.

the discrepancies are 1.9% and 3.6% in the core and assembly respectively for ^{238}U.

It is important to notice that at 47.1 cm, there is a sudden rise/increase in fission rates for ^{239}Pu and ^{235}U while a sudden dip/decrease for ^{238}U. The reason for this sudden change in fission rates is that the layers of two different kinds of fuel rods touch each other i.e. the position at 47.1 cm lying in LEZ is actually a junction for two different fuel rod types FR-1 and FR-10 (with reference to the Benchmark document reference (Manturov et al., 2006). The FR-1 type fuel rods (belonging to LEZ and containing fuel cells of fissile material) are located in the core's central part in key region while FR-10 type safety rods (belonging to key region) are present in outer side of LEZ towards the reflector and touching the MEZ. The FR-10 contains the cells with no B_4C material (only Na and SS pellets are there) along the active length of 73 cm (for Safety Rod). So at this point/position, the neutron spectrum gets softened and hence the sudden shift in the fission rates is observed. The softened neutrons being more prone to cause fission in ^{239}Pu and ^{235}U are owing to a rise in the fission rates as is obvious in Figs. 4 and 5. The scenario, however, is flipped (inversed) for ^{238}U as there would be a decrease/slump as shown in Fig. 6 (as the isotope ^{238}U has higher fission cross-section for fast neutrons i.e. the softened/moderated neutrons would be causing comparatively less fission events).

The reaction rates in blanket region are important to be discussed too. It is worthy to note that blanket region starts slightly ahead of 100 cm from the core centre and the first measurement point is at 108.9 cm lying in stainless steel (SS) reflector. So, four measurement points are in reflector while the last/fifth position at 144.3 cm lies in absorber part of the radial blanket. The calculated fission rates for all of the three isotopes are with some substantial deviation as compared with the experiment. It seems that neutron spectrum is imbalanced inside the reflector because overestimation of reaction rates for ^{239}Pu and ^{235}U seems prominent. A good reason for that could be a speculation as if the experimental data regarding SS reflector is not accurate and the same has been argued by many other researchers; including (Jamil, 2018); (Rachamin and Kliem, 2017), (Ivanov and Bousquet, 2016) and (Marinoni et al., 2012). The simulation offers over-moderation (and under-leakage) of the neutrons in reflector region that turns the neutron spectrum more feasible to cause fission with ^{239}Pu and ^{235}U isotopes and eventually their fission rates got increased. The ^{239}Pu, for instance, gives an over-estimated fission rates with a maximum deviation up to 20.0%. The simulation results could be further improved for reflector region by decreasing the reflector's density up to some appropriate level. The readers may refer to the investigations made by Jamil Z. et al. (Jamil, 2018) to better understand the effect of decreased reflector's density on fission rates.

It is to be considered that the experiment itself because of uncertainty in exact positioning the fission chambers in core and blanket region has stated of uncertainties in reaction rate measurements. For ^{239}Pu and ^{235}U, these uncertainties are about 1.5–2% in fuel-region and 3–4% in radial blanket while about 2–3% in fuel-region and 5–7% in radial blanket for ^{238}U (Manturov et al., 2006). Besides the stated sources of discrepancies in fission rates, the dominant component is of course the uncertainty in material density in reflector region (because of difficulty in packing the SS dowels that may lead to some considerable dispersion around the average value – "24 dowels per tube"). Keeping the margin of experimental uncertainties and the speculation of reflector's data of being inaccurate in to consideration, the simulation predicted the fission rates well and the calculated results seem justified.

3.4. Sensitivity studies

Any change in the input parameters like mass/density, isotopic composition of the nuclides, and physical volume (diameter and height) of the fuel pellets or other structural elements of the assembly may significantly alter the results of the measurements either performed by the experiment or running some code simulation. The BFS-62-3A critical experiment's document describes briefly about the standard errors/uncertainties in masses of the different component materials (fuel and structural material) (Manturov et al., 2006). So, it is important to take these experimental uncertainties in to account to study the corresponding effect on the performed simulation. For this reason, some selected materials were taken and their respective structures in the form of pellets

Table 3
Sensitivity prediction: Reactivity change by changing the density of component materials.

Component materials (pellets or dowel)	$\Delta\rho$, pcm
UO_2-dep	229.0
UO_2-36%	16.8
U-36% (old)	96.88
U-90%	132.5
Pu-95%	101.8
SS-dowels	3.9

and/or dowels were re-modeled by changing their mass-densities i.e. the mass (and hence the density) was decreased by subtracting the respective experimental uncertainty from the nominal mass value. It was to expect that the increase in the mass (and hence the density) by the respective uncertainty would render approximately the same quantitative impact on the reactivity of the core i.e. the results to be symmetrical by adding or subtracting the mass-uncertainty. So, only the decrease in mass was considered for this sensitivity studies. The study should in fact include the investigation on all of the component materials used in the experiment, but here we have a few of them; including UO_2-dep, UO_2-36%, U-36% (old), U-90%, Pu-95%, and stainless steel for dowels (SS-dowels).

Table 3 quantitatively describes the extent to which the reactivity of the critical assembly is sensitive to the experimental uncertainty in masses of the component materials. The largest impact on reactivity is associated with UO_2-dep that is well justified as this material in the form of pellets is abundantly available in all of the fuel cells belonging to the four fuel zones (LEZ, MEZ, PEZ and HEZ). The reactivity is also substantially sensitive to fuel materials; including U-90% (present in HEZ), U-36% (old) (present in LEZ) and Pu-95% (present in PEZ). The UO_2-36%, present only in LEZ with a few numbers of pellets, has comparatively less effect. The sensitivity of the reactivity can also be estimated for reflector's density. The SS reflector in the blanket region is composed of various standard BFS SS tubes, each filled with about 24 SS dowels (on the average). As we changed the nominal mass of one SS-dowel by its respective experimental uncertainty, the reactivity change observed was 3.9 pcm. It is clear that because of the uncertainty in exact packing of dowels in each tube, the mass uncertainty of the tube as a whole and ultimately that of the entire reflector would significantly be increased. The increased mass uncertainty of the reflector region will eventually lead to somewhat bigger reactivity shift in the core. The sensitivity studies reveal that material type, its quantity and its location inside the assembly all contribute to affecting the core's reactivity.

4. Conclusions

For a MOX-fueled fast reactor, the credibility and accuracy of a point-wise nuclear cross-section data library, the HENDL, was assessed by employing a Monte Carlo code, the SuperMC. The benchmarking of data library involved a comprehensive neutronic analysis of the critical assembly, the BFS-62-3A, by calculating integral neutron physics parameters. The reactivity effects were measured by calculating the SVRE and CRW. In the calculations, the former was underestimated by about 8 cents while the latter agreed well with the measured data within 5.1% on the average. While calculating the sodium void reactivity, the removal of sodium pellets was followed by a sequence i.e. removal started from inner core and ended at outer core (from LEZ through HEZ) such that the previously voided sub-zone was kept voided. If the order is reversed following a sequence starting from outer core to inner core, the total value of SVRE is expected to be nearly the same while the results for individual four zones are expected to be significantly different. It is to conclude that reactivity effects have a significant dependence on sequence of voiding the zones in case of SVRE measurements, and sequence of loading the control rods in case of CRW measurements.

The reaction rates for the three isotopes were estimated in the key region of core and blanket in radial direction. For ^{239}Pu, ^{235}U and ^{238}U respectively; the discrepancies of 2.1%, 2.2% and 1.9% on the average were observed in the core while the discrepancies of 4.0%, 3.5% and 3.6% on the average were found in the assembly (core + reflector). The spectral indices, criticality and sensitivity were all predicted well by the library.

The calculations agree well with the available experimentally measured data within acceptable limits that confirms the fidelity of the library. The study has benchmarked and validated the predictions of HENDL nuclear data library and strengthened its use in prospective research for advanced and Gen IV reactors based on fast neutron spectrum.

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