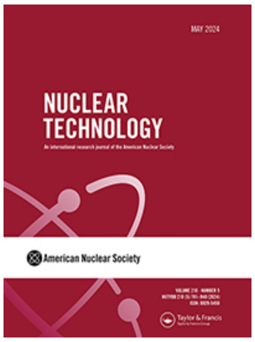

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Neutronic Analysis of the Conceptual Molten Uranium Breeder Reactor Using MCNP and SCALE Tools

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Abstract — *The Molten Uranium Breeder Reactor (MUBR) is a radically new reactor concept with a mixed-energy spectrum. MUBR is fueled with molten uranium metal in large-diameter fuel tubes and is cooled by circulating molten uranium fuel through a heat exchanger. The reactor has heavy water as moderator, and the reactivity of the reactor is primarily controlled by the voiding effect of the moderator through an innovative control cavity structure design. Because the MUBR design is vastly different from most existing fission reactors, neutronics analysis must be performed for many different combinations of design parameters to identify viable and optimum design configurations. To facilitate the neutronics analysis, a proprietary program called MUBR6gen is being developed to provide a pipeline tool to expedite the process. MUBR6gen employs two well-established neutronics codes, i.e., MCNP and SCALE, to perform standard neutronics calculations for MUBR by automating input preparation and output processing. In addition, MUBR6gen ensures consistency of the MCNP and SCALE inputs and compares the outputs of the two codes to warrant the simulation results. Augmented with MUBR6gen, standard neutronics analysis was carried out on a small-scale MUBR design, which serves as a model problem in the paper. The neutronics performance characteristics of the model reactor were obtained and discussed in a code-to-code pattern. An overall very good agreement between the results of the two neutronics codes was established. Based on the success of the model problem analysis, further neutronics analysis using MUBR6gen was extended for a set of MUBR variant designs. Meaningful and promising fuel cycle analysis results for the 10 different designs were achieved and discussed. These results are used to identify the best MUBR candidates in terms of fuel lifetime and utilization efficiency for future applications.*

Keywords — *Molten Uranium Breeder Reactor, control cavity structure, MCNP, SCALE.*

Note — *Some figures may be in color only in the electronic version.*

I. INTRODUCTION

The Molten Uranium Breeder Reactor (MUBR) is a radical uranium-plutonium fission reactor concept proposed by Mann and Pop.[1,2] The physics feasibility of MUBR was preliminarily confirmed by an early study performed at Oak Ridge National Laboratory (ORNL) with the conclusion that the concept could work but that the configuration should be optimized.[3] MUBR is an advanced uranium-based fission reactor and has significant technical advantages compared to other commonly known advanced reactors such as those recommended in

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the Generation IV reactor roadmap.[4] The underlying idea of MUBR is based partly on concepts of existing CANDU reactor[5] and molten salt reactor designs.[6]

With the MUBR design concept, fission is caused at around 45% by thermal neutrons, around 35% by fast neutrons, and around 20% by intermediate neutrons. As a result, MUBR can operate on a breed-and-burn fuel cycle where the plutonium is bred and consumed in the reactor core at the same locations where it is bred. Breeding is effective in MUBR because of the harder neutron energy spectrum and the lower neutron loss, which also leads to an initial fuel loading with a low (2 to 3 wt%) fissile content.

Specific design features of MUBR include the following: The fuel is molten uranium metal in large-diameter fuel tubes instead of solid and thin fuel rods as that in water reactors; the fuel circulates through the core to an external heat exchanger during normal reactor operation; the moderator is heavy water; the reactor is controlled over a very wide fuel reactivity range with an innovative control cavity structure (CCS) design; and some fission products are continuously removed from the circulating molten fuel because they have a boiling point lower than the fuel temperature (1200°C to 1400°C) or are insoluble in molten uranium and float above the fuel as dross. Figure 1 shows a conceptual diagram of the MUBR design with the major components in the system level.

The MUBR design uses heavy water as a moderator and reflector to reduce neutron absorption and increase the conversion ratio.[1] The MUBR design also utilizes a gas cover to capture any fission products that will evaporate. MUBR operates with the molten fuel between 1200°C and 1400°C.[1] Because of such a high operating temperature, many of the fission products will evaporate. Because of this, the gas cover will allow the evaporated fission products to be removed from the fuel.

On the fuel side, MUBR can be fueled by either low-enriched uranium (LEU) with 2.0 to 3.0 wt% ^{235}U or a mixture of standard LEU (4.95 wt% ^{235}U) with 40% to 60% light water reactor (LWR) used nuclear fuel (UNF), which can be reduced to metal from its original metal oxides.[7] Because of the distinct breeding capability of MUBR, its fuel life can be longer than the reactor life, with the potential to achieve a life of hundreds of years. By then, over 90% of the energy is produced by direct or indirect fission of ^{238}U, and at the end of the reactor life, the fuel becomes an asset, not a liability, being potentially ready to be used in a similar reactor for another very long period of time.[1] Because most of the power is from the fission of the plentiful ^{238}U, the amount of uranium mined per megawatt hour is reduced by a factor of around 10, and the nuclear waste per megawatt hour is also reduced by a factor of around 10. The used fuel thus can be utilized as part of the initial fuel in a new MUBR. If the ending fuel reactivity is too high because of effective breeding, it can be downblended with depleted uranium to form an optimum initial fuel reactivity. If the ending fuel reactivity is too low because of insufficient breeding, it can be upblended with standard LEU (4.95 wt% ^{235}U) to achieve the optimum initial fuel reactivity.

In water-based nuclear fission reactors, the effective multiplication factor (i.e., k_{eff}) is controlled by a number of factors[8]: (1) fuel composition (changed by partial or complete refueling many times during the reactor life);

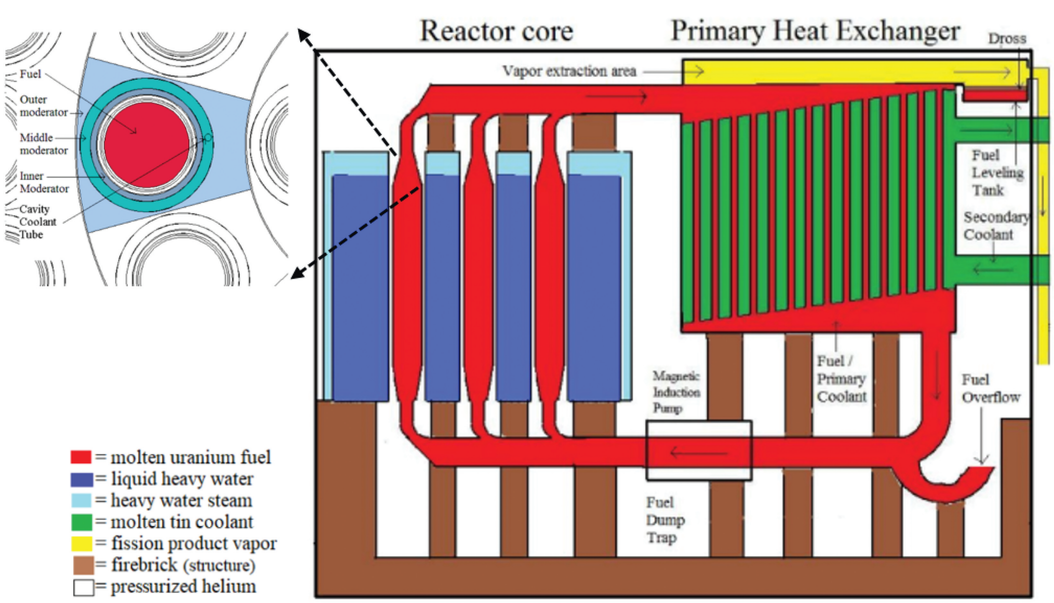

Fig. 1. Conceptual diagram of the main components in MUBR.

(2) insertion of or positioning of neutron-absorbing control rods; (3) use of burnable poisons (neutron absorbers that decrease their absorption rate with burnup to match the decrease in fuel reactivity); (4) temperature coefficient of reactivity; and (5) void coefficient of reactivity, which is important but usually limited in water reactors for safety reasons to prevent void occurrence in the coolant flow. However, reactivity control in MUBR is mostly provided by the void reactivity coefficient of the liquid heavy water, which serves as the moderator for the reactor. The void (moderator steam) can be small or as large as the entire moderator. This provides a very large reactivity control range (over 10 000 pcm) between all liquid moderator and all moderator steam. In addition, instead of being absorbed and wasted, excess neutrons are under-moderated, so they are preferably absorbed by resonance capture in ^{238}U to produce ^{239}Pu and thereby contribute to the conversion ratio of the reactor. There is no safety concern of rapid reactivity control for MUBR because the fuel is not cooled by the moderator; rather, it is cooled by circulating the molten uranium metal fuel through a heat exchanger outside of the reactor core. There is a heat shield between the fuel and the moderator to reduce thermal radiation and conduction. The control method is very sensitive and very fast because the energy that boils the liquid moderator does not come from heat transfer but rather comes from the fast neutrons that reach all of the moderator in less than a microsecond, and the moderator temperature is always at the boiling point.

There is no engineering design for MUBR yet because the design is still in the stage of determination of the best reactor configurations. The MUBR concept contains large-diameter fuel tubes instead of thousands of thin fuel rods. Each fuel tube is centered in a CCS filled with heavy water liquid and steam that serves as the moderator for MUBR. One distinct feature of MUBR is that the reactivity of the reactor is controlled by adjusting the ratio of heavy water moderator liquid to moderator steam to provide proper reactivity control. This feature is achieved by the innovative CCS design.[9] The array of CCS/fuel tube structures is surrounded by a heavy water reflector on all sides. For simplicity of construction and analysis, all the CCS and fuel tubes are designed with the same size. To build a larger reactor with higher power rate, one just needs to add more CCS/fuel tube assemblies to the design. Figure 2 illustrates a conceptual diagram of the CCS configuration with a brief description of the reactivity control mechanism through the CCS system provided on the side.

All the aforementioned unique design features of MUBR influenced the conventional nuclear simulation tool used to analyze the idea. Because MUBR thus far

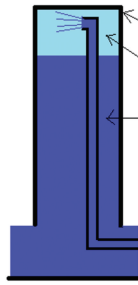

Fig. 2. Conceptual diagram of the CCS design in MUBR.

is only a reactor concept, the first step of the evaluation is the neutronics feasibility of the concept. At the neutronics analysis stage, a vast number of designs, materials, fuel variations, and knowledge uncertainties need to be analyzed and evaluated. Two well-established neutronics simulation tools, i.e., MCNP[10] and SCALE,[11] are adopted to perform the MUBR analysis. Both computational codes are widely used and respected in nuclear reactor physics applications. They both require an input file that describes the geometry and materials in the situation to be analyzed and demand some directions on what analysis is required. Not surprisingly, the two input files have different formats, and their output files containing the results are even more different in format. These differences pose some limitations on the analysis that can be done. Fortunately, both codes can do a basic evaluation of the multiplication factor and the changes in the fuel composition over time with fuel burnup.

To provide greater confidence in the analysis results for this radical MUBR concept, a QuickBasic (QB64) language-based program,[12] Neal Mann Inc. proprietary MUBR6gen.exe, has been developed to automate both MCNP and SCALE inputs and executions, read and analyze the output files, and create a log file entry and a report of the results. The input files are built based on a large number of parameters that describe the details of the situation and what analysis is desired. In this work, the latest versions of both codes, MCNP6.2[10] and SCALE6.3,[11] are employed in MUBR6gen.

With the computational modeling and simulation process facilitated by MUBR6gen, the work presented in this paper first discusses the basic MUBR design concept, followed by a thorough calculation of standard neutronics performance of a 334-MW(thermal), small-scale, 19-tube MUBR (referred to as "model problem"). A comparison of the MCNP and the SCALE results is purposely presented through the study of the model problem to provide credibility of the entire analysis. The model problem study is followed by more extended analysis of MUBR, including a discussion of unknown factors that may influence the standard design and a discussion about how simulations with various assumed values of the parameters affect the sensitivity of our results to changes in those parameters. These discussions are connected to potential effects of the MUBR concept on reactor operation, the fuel cycle, disposal of existing UNF, the environment, and power economics. This technical content of the paper is completed with a short summary of the modeling and simulation limitations that exist in the current computational tool set, which provides some caveats on using and interpreting the results presented in the paper. Conclusions based on the current analysis and future work with regard to continuing endeavors on the MUBR design are outlined in the last section of the paper.

II. ANALYSIS METHOD AND MUBR6GEN PROGRAM

The analysis of MUBR is performed essentially with two well-known neutronics analysis codes: MCNP[10] and SCALE.[11] MCNP is a Monte Carlo method–based radiation transport code developed by Los Alamos National Laboratory. SCALE is a nuclear modeling and simulation tool package developed by ORNL. The Monte Carlo–based transport code module KENO[13] included in the SCALE package is the main computation tool for neutronics calculations in SCALE. Both MCNP and SCALE can be used for reactor physics, critical safety, and depletion (i.e., fuel burnup) analysis and for investigating the sensitivity of key parameters in nuclear reactor design. Because of the nature of Monte Carlo modeling, MCNP and SCALE offered the ability to create the complex structure of the MUBR design. In addition, SCALE was selected particularly because of its specific feature that can filter/remove fission product materials and store them elsewhere.[14] This feature cannot be directly realized in MCNP but can be mimicked by running a very large number of steps and adjusting the fuel composition at the end of each step, which is a process that is very slow and not very accurate.

Since two computer codes are used in parallel in MUBR calculations, a code-to-code verification working philosophy is implemented all through the analysis procedure. Along the analysis, each code will execute the same tests with the same input parameters. Comparing the results obtained from the same reactor configurations will provide valuable information for further studies of the reactor design. The MUBR design was originally created in MCNP and is now also built in SCALE. The burnup simulations based on MCNP6 for a MUBR configuration confirmed the primary neutronic feasibility of the reactor.[1] SCALE can be utilized to confirm these results and implement new features for more accurate analysis.

Since there are so many possible variations of the reactor concept, developing a systematic methodology to conduct and pipeline a possibly large number of simulations becomes necessary. For this purpose, the Neal Mann Inc. proprietary computer program named MUBR6gen has been developed to streamline the analysis. It uses a large number of parameters to specify the details of the desired geometry, materials, and design objectives. The program handles both neutronics codes in parallel, which means it generates input files for MCNP or SCALE, executes MCNP or SCALE, reads the MCNP or SCALE output file, extracts the useful information from the file, produces a report, adds a line to a log file, and can repeat the process with new input based on a systematic change of some parameters or the results of the MCNP or SCALE analysis. The MUBR6gen parameters have default values coded in the program, but MUBR6gen can also read a small parameter file that specifies values for any number of parameters. Parameter values can also be specified in the command line when executing MUBR6gen: command line parameters override both the default values and any values specified in the parameter file. While the input file formats for MCNP and SCALE are vastly different, MUBR6gen creates input files that describe exactly the same geometries and materials for both codes because they are all created from the input source. This allows us to analyze exactly the same reactor configuration in both MCNP and SCALE and compare the results consistently. Figure 3 illustrates the major modules and workflow involved in the MUBR6gen program.

While the main purpose of this paper is to present the neutronics analysis of MUBR, the methodology and development of the MUBR6gen program are paramount for the MUBR project because the MUBR concept is so different from other conventional reactor concepts that it requires much more intensive analysis since all of the design details are new and all variations of these details need to be analyzed and validated. A new design concept requires new and fast analysis, so a new analysis methodology is desired. This is the main reason we introduce the MUBR6gen program in this paper and include sufficient analysis results based on the program to show that the concept is valuable and to provide enough design

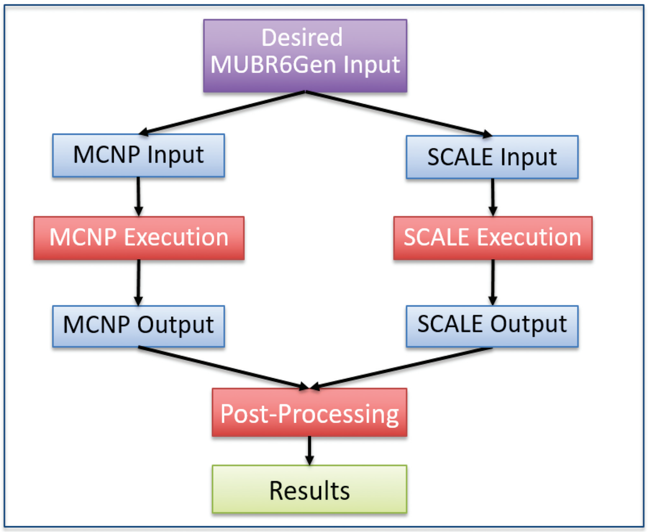

Fig. 3. Flowchart of the work tasks inside MUBR6gen.

details to convince one as to the plausibility of the concept. This is demonstrated in Sec. III with a presentation of standard neutronics analysis for a model problem based on one MUBR design.

In addition, MUBR6gen integrates the neutronics and fuel burnup calculation modules, enabling rapid analysis of many different reactor configurations as well as allowing sensitivity analysis of issues where fundamental knowledge is missing. These questions include the maximum operating temperature of the proposed silicon carbide fuel tubes, other possible fuel tube materials, and the rate of separation of fission products from the circulating molten fuel. For reasons of cost or availability, it also allows study of the effects on neutronics of substituting different materials. This is demonstrated in Sec. IV with more extended neutronics analysis of various MUBR designs.

III. MODEL PROBLEM AND STANDARD ANALYSIS

With the current analysis capabilities of the MUBR6gen program, we will perform neutronics analyses on 10 different MUBR conceptual case designs that range from a single-tube prototype reactor to an 80-tube, grid-scale reactor. In this section, we provide detailed neutronics analyses using one of the case designs as the model problem, which is referred to as the 19-fuel-tube, small-scale reactor. The primary purpose of the model problem study is to provide a code-to-code confirmative investigation that serves as a reliable base to support all other calculations performed by the MUBR6gen program. In later sections, we will offer results yielded from all 10 MUBR variant cases for comparison purposes.

III.A. Model Problem Description

The model problem is a small-scale MUBR conceptual core consisting of 19 fuel tubes. It has the same major characteristics as other sizes of the MUBR design studied. Both the MCNP and the SCALE models of the 19-tube core are generated by MUBR6gen. Since the computational models are developed with the same high-level engineering parameter inputs, the consistency of the model configurations and material compositions is retained for both models. The 19 fuel tubes are distributed in the core in a hexagonal lattice as shown in Fig. 4 with the characteristic design parameters of the core summarized in Table I. The fuel tubes are 42 cm in diameter and 190 cm in height in the core region. During normal operation, the ~1475 K (~1200°C) molten uranium would flow through these tubes, which are made of multilayer silicon carbide. The molten fuel is heated to up to 1400°C as it rises through the tube.

Figure 4 offers both a top-down view and a side view of the model problem core configuration. The side view is through the center plane of the core. A horizontal line hovers over the side view to indicate the axial position of the top-down view. The separation regions between the heavy water moderator and the moderator steam can be clearly seen from the side view of the core. The heavy water slows down the neutrons in the lower part of the core. As the fuel rises to the upper part of the core, where the moderator becomes heavy water steam, the neutrons are very undermoderated because of the very low density of the heavy water steam. As a result, the fission rate is greatly reduced in the upper region of the core. Figure 4 is rendered by the SCALE model, but similar graphic views of the core model can be generated by MCNP as well.

III.B. Effective Multiplication Factor (k_{eff})

In the MUBR design, the reactivity of the reactor is controlled by changing the amount of liquid heavy water moderator in the CCS surrounding each fuel tube. When there is less liquid moderator in the cavity, the fast neutrons are less moderated and preferentially captured by ^{238}U due to resonance absorptions. This reduces k_{eff} and increases the conversion ratio, diverting excess neutrons from fission of the fissile content and largely converting ^{238}U into fissile ^{239}Pu.

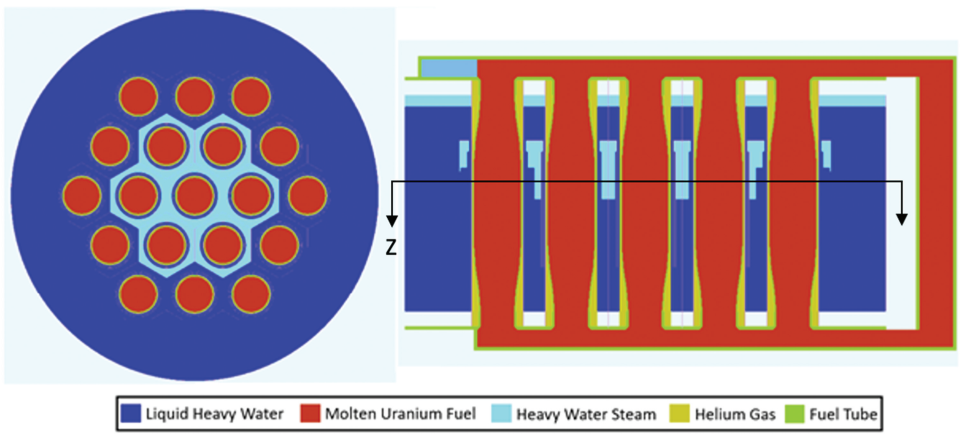

Fig. 4. Top-down view (left) and side view (right) of the model problem.

TABLE I

Characteristic Design Parameters of the 19-Tube, Small-Scale MUBR Core

Parameter	Value
Reactor power [MW(thermal)]	334
Average power density (W/cm^3)	34
Fuel cycle length (yr)	≥60
Number of fuel tubes	19
Fuel tube diameter (cm)	42.0
Initial liquid D_2O%	45.0
Fuel type	LEU or LEU + LWR UNF
Fuel inlet/outlet temperature (K)	1480/1680
Moderator material	Heavy water
Moderator temperature (K)	440
Fuel tube material	Silicon carbide
Gas cap	Helium or argon

Along with the steady-state neutronics simulation, the criticality status is achieved by adjusting the percentage of the control cavity occupied by the liquid heavy water. Neither MCNP nor SCALE can achieve this automatically. Instead, we realize the critical status with a semiautomatic criticality search procedure. A heavy water level for the transition of liquid heavy water to heavy water steam is manually specified in each cavity to obtain the corresponding k_{eff}; then, the heavy water height can be adjusted appropriately before the simulation. This process is repeated until the computed k_{eff} is close enough to unity. MUBR6gen.exe automates this iterative search process as user directed. The initial liquid heavy water occupied in the CCS for the model problem is 45%. Table II summarizes the calculated k_{eff} values for the model problem at the beginning of the cycle. Two alternative fuel options, LEU fuel and a mixture of LEU + UNF fuel, are considered in this calculation. The UNF

TABLE II

The k_{eff} Values Generated from Two Codes for the 19-Tube, Small-Scale MUBR

Fuel Type	MCNP	SCALE	Difference
LEU (2.98 wt% ^{235}U)	1.00075 ± 0.00016	1.00099 ± 0.00021	0.00024
LEU (4.95 wt% ^{235}U) + 40% UNF	1.00336 ± 0.00014	1.00161 ± 0.00020	0.00175

composition we used in this work was generated using a pressurized water reactor model based on the SCALE libraries and ORIGEN module in the SCALE code system.[11]

The results shown in Table II indicate very good agreement between the SCALE- and MCNP-calculated k_{eff} values. The first row is a comparison of LEU fuel containing 2.98 wt% ^{235}U. There is a nearly negligible percent difference of ~0.024% between the results from the two codes. The second row utilizes fuel that is a mixture of LEU fuel enriched with 4.95 wt% ^{235}U and 40% UNF. This case has a percent difference of ~0.175% between the two calculations, which is acceptable considering the inevitable statistics errors associated with Monte Carlo simulations. It is noteworthy to mention that the Doppler Broadening Rejection Correction (DBRC) option[15] appears to be available in SCALE but not available in the version of MCNP that we used in this work (MCNP6.2). However, all the calculations that we performed in SCALE have the DBRC option disabled by default, so the results that we obtained from both codes are consistent as far as DBRC is concerned. Thus, we trust that both cases display close agreement between the MCNP and the SCALE results of the k_{eff} predictions for the model problem.

III.C. Neutron Flux Distribution

We first inspect the neutron flux distribution over the entire MUBR core region. The three-dimensional neutron flux distribution for the small MUBR is generated from both MCNP and SCALE. For clarity, the energy mesh is split into three energy bins designated as fast energy range (20 MeV to 100 keV), intermediate energy range (100 keV to 0.625 eV), and thermal energy range (0.625 eV to 1E-7 eV), respectively. Note that SCALE does not allow one to designate 0 for the lowest-energy limit, so 1E-7 eV is used for the lowest cutoff energy boundary in SCALE as suggested by the code manual. Other than that, both codes use the same energy group structure for flux tallies. To improve the accuracy of the flux tally results, the calculation is executed with sufficient neutrons and generations to ensure that the relative uncertainty for the majority of the flux in every energy bin is under 1%.

Last, in order to view the three-dimensional flux distribution comparably in each direction, the number of increments in the axial direction is increased to match the distance of the increments in the radial (x and y) directions. Figures 5 and 6 give a top view and a side view of the flux distribution in the MUBR core region, respectively. Both figures contain the results from MCNP and SCALE. These results are presented in a code-to-code comparison manner such that overall good agreement on the flux distribution between the codes is clearly demonstrated. Note that apparent asymmetrical distributions are observed in the reflector area for both the intermediate and the fast neutrons. This is attributed to the asymmetric reactor component setting for the system. The fuel return path through the heat exchanger and the fuel pump is outside the reflector located on the right side of the reactor core as shown in Fig. 1.

In the preliminary analysis, we did not perform detailed thermal-hydraulic calculations of the design. However, it is noteworthy to point out that we considered that the cooling of each control cavity is more or less the same as all the other adjacent cavities because they share a common cooling flow. We assumed that each cavity was heated proportionally to the total fast neutron flux in the cavity, which is obtained by the fast neutron flux integrated over the cavity volume. As mentioned earlier, the control method of MUBR assures that the heating in each cavity is the same as the cooling, so each cavity has the same total fast neutron flux and therefore roughly the same rate of total fission in the fuel tube, which assures that the exit temperature in each fuel tube is the same as all the others.

This phenomenon is well confirmed by the flux distribution result shown in Figs. 5 and 6. Within the core and mostly at the bottom of the core, the outer fuel tubes have a lower fast neutron flux than the central tubes, while the central control cavities achieve their integrated fast flux at a lower liquid moderator level and have a lower fast neutron flux at the top of the fuel tubes than the outer fuel tubes. As a result, all tubes have the same integrated fast neutron flux. They are just distributed differently between the inner tubes and the outer tubes.

On the other hand, one drawback that is identified in the overall flux distribution is that the region with no thermal neutron flux mainly exists in the large central region of the fuel tubes, which prohibits us from a detailed understanding of the flux condition in these tubes. To have a close inspection of the flux at the tubes, we need a side view and a top view of the neutron flux distributions in some individual tubes. To achieve this, the flux surrounding several of the fuel tubes was analyzed closely. The center fuel tube is surrounded by six other fuel tubes, which is a unique location for the reactor layout. The corner fuel tube that is surrounded by three other fuel tubes was analyzed along with the fuel tube that is surrounded by

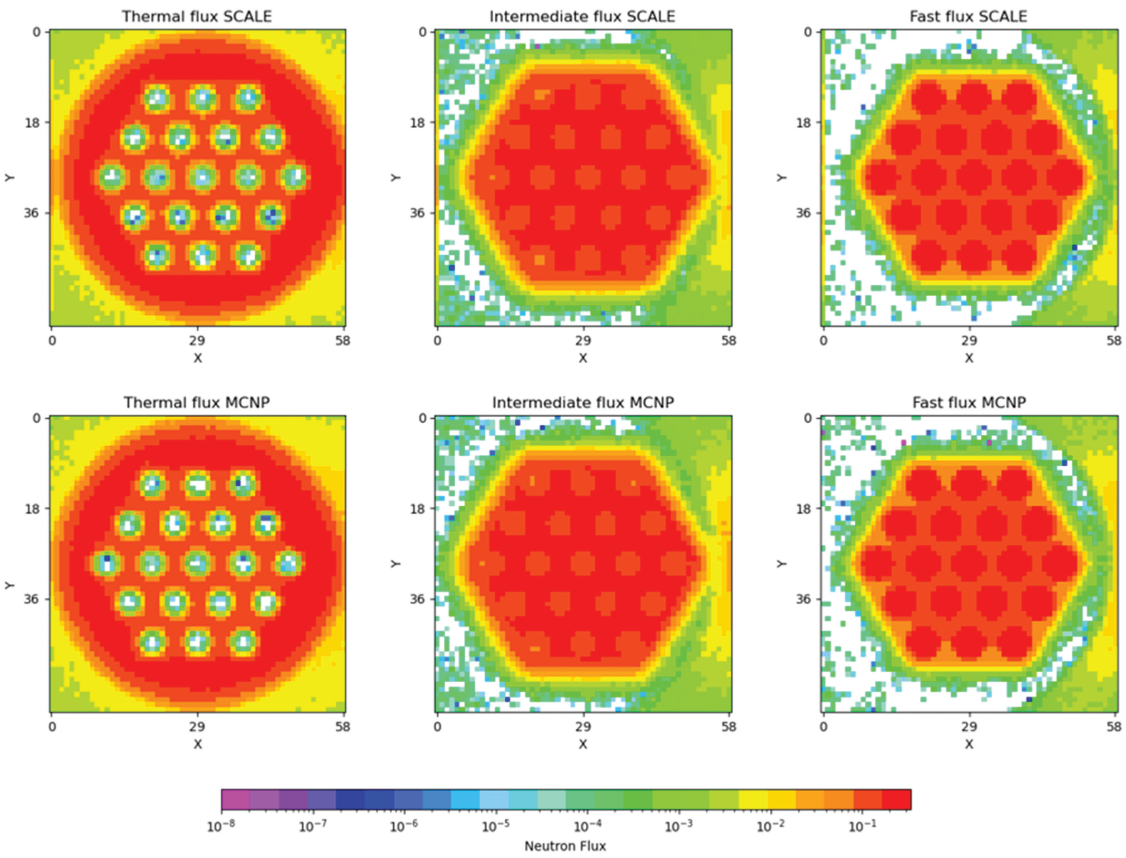

Fig. 5. Top view of small MUBR flux distribution by MCNP (bottom) and SCALE (top) at the midplane of the core.

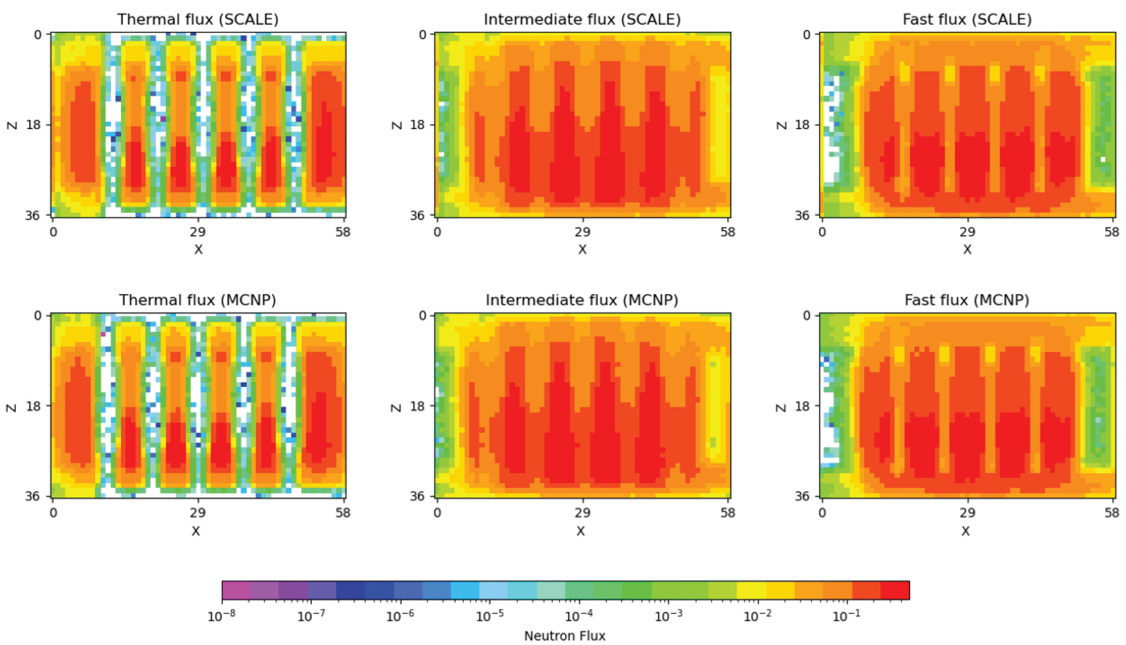

Fig. 6. Side view of small MUBR flux distribution by MCNP (bottom) and SCALE (top).

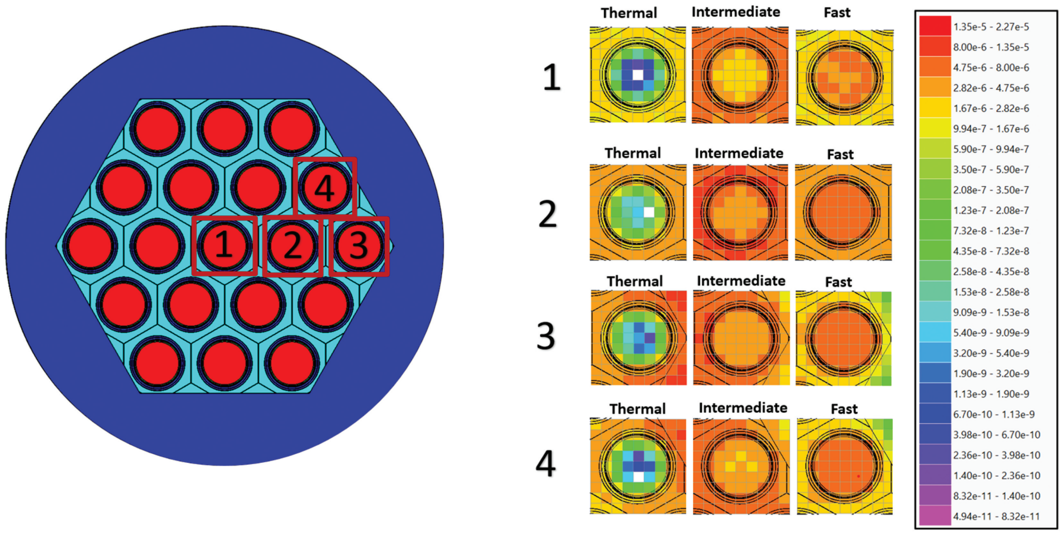

Fig. 7. Top view of neutron flux distribution in some individual fuel tubes.

four fuel tubes. Three separate meshes were generated in SCALE for these specific locations. Close views of the flux distribution in some individual but representative fuel tubes in MUBR are shown in Figs. 7 and 8, respectively. These results are generated from SCALE only. Similar results can also be obtained from MCNP calculations.

III.D. Energy Spectra

To understand the unique neutronics characteristic of MUBR, it is interesting to examine the flux energy spectra of the core. This can be achieved by performing an additional neutron flux tally calculation based on a more refined energy group structure. The standard SCALE 252-energy-group structure[16] is used for this purpose. Figure 9 shows the core-averaged energy spectrum of MUBR generated by both MCNP and SCALE, respectively. Note that the statistical errors associated with every Monte Carlo flux tally are also included in the plot; however, they are too tiny to be viewable in the figure. Though some small discrepancies are noticed in a few energy bins, especially in those thermal and fast flux peak locations, the spectrum characteristics receive overall good agreement in the two codes, which verifies the correctness of the spectrum calculations in both codes.

The MUBR spectrum becomes more insightful when it is compared to the typical energy spectra for a conventional thermal reactor and fast reactor.

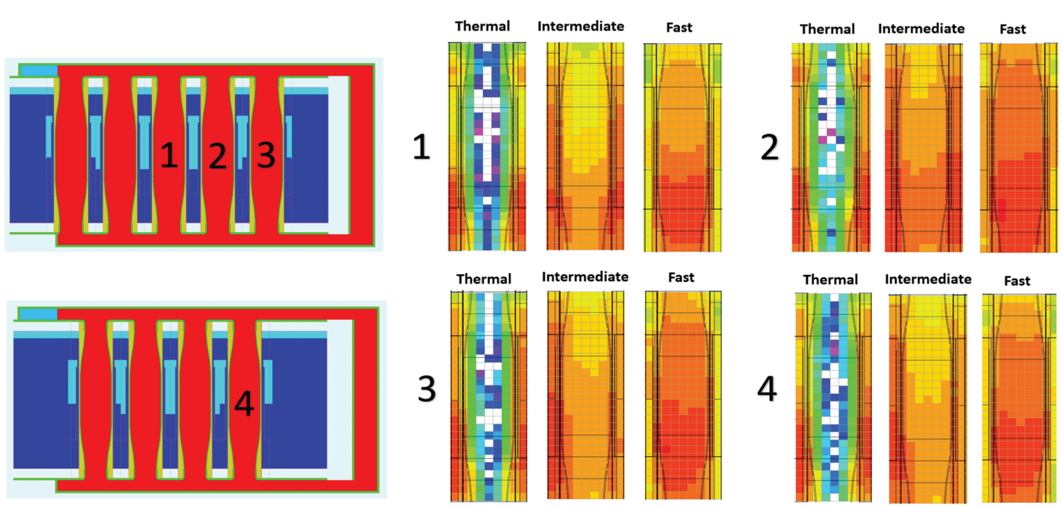

Fig. 8. Side view of neutron flux distribution in some individual fuel tubes.

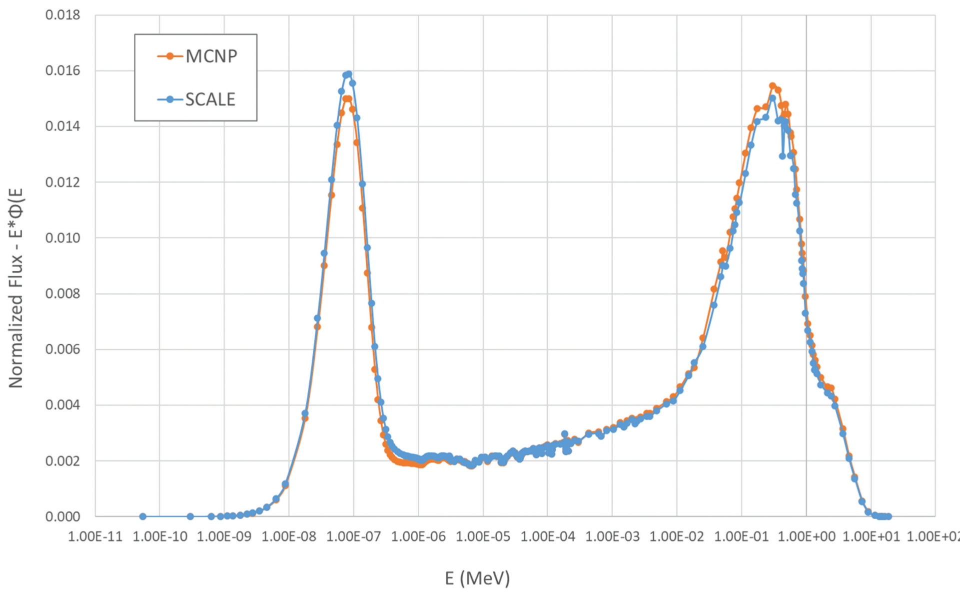

Fig. 9. Core average spectrum generated by MCNP and SCALE.

Figure 10 compares the MUBR spectrum to the assembly-wise spectra of a conventional LWR and a lead fast reactor (LFR).[17] The LWR and LFR spectrum results are generated from MCNP calculations separately. The unique features of the MUBR flux energy distribution are clearly demonstrated in Fig. 10. The figure also shows two extreme behaviors with two flux peaks at the thermal and fast energy ranges, which are similar to the LWR results. However, both peaks in MUBR are shifted to a certain level toward the intermediate energy range, and the magnitudes of these peaks are also higher than those of the LWR. In this regard, MUBR is apparently not a "fast" reactor; however, it is not a typical "thermal" reactor either.

These characterized features exhibited by the MUBR spectra can serve as a basis to explain the superior breeding ratio/conversion ratio calculation. Because of the spectrum-hardening tendency, MUBR favors ^{238}U neutron capture and also favors fast neutron fission reactions in the reactor.

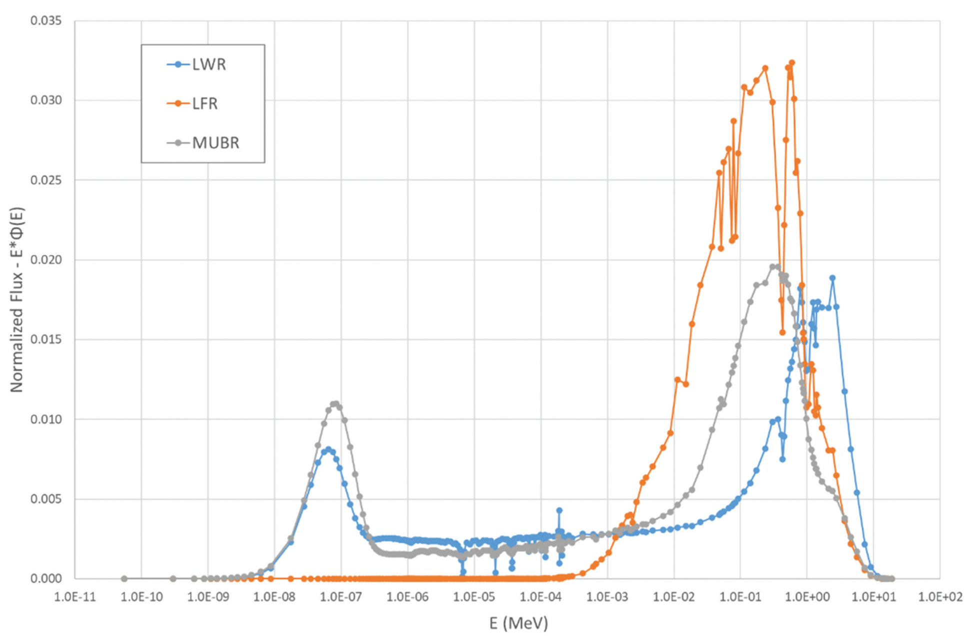

Fig. 10. MUBR spectra compared to those of some typical reactors.

III.E. Reactivity Control Range

As briefly outlined in Sec. I, MUBR features a large and flexible operational reactivity control element thanks to its innovative CCS design. This advantageous reactivity safety control feature can be justified with standard neutronics calculations. We calculate the reactivity control worth by incrementally adjusting the heavy water volume fraction in the CCS. This calculation is performed for both the LEU fuel and the mixed LEU + UNF fuel conceived at the beginning of the fuel cycle.

Figure 11 shows the reactivity control worth curves for the LEU-fueled MUBR. Both the MCNP and the SCALE calculation results are presented and show good agreement. As can be seen, a large range of reactivity control (~13 800 pcm) is achieved at this condition. In addition, the control worth curve is shown with a more linear behavior rather than a typical s-shaped control worth curve as normally seen in typical water-cooled thermal reactors.[18] This is because the mechanism of the CCS is largely based on a physical reaction (boiling) and achieves a more homogenous control pattern over the core.

Figure 12 compares the control worth curves for the LEU core and the mixed-fuel core. These results are based on MCNP calculations, but similar ones can be drawn based on the SCALE results. Relatively large discrepancies

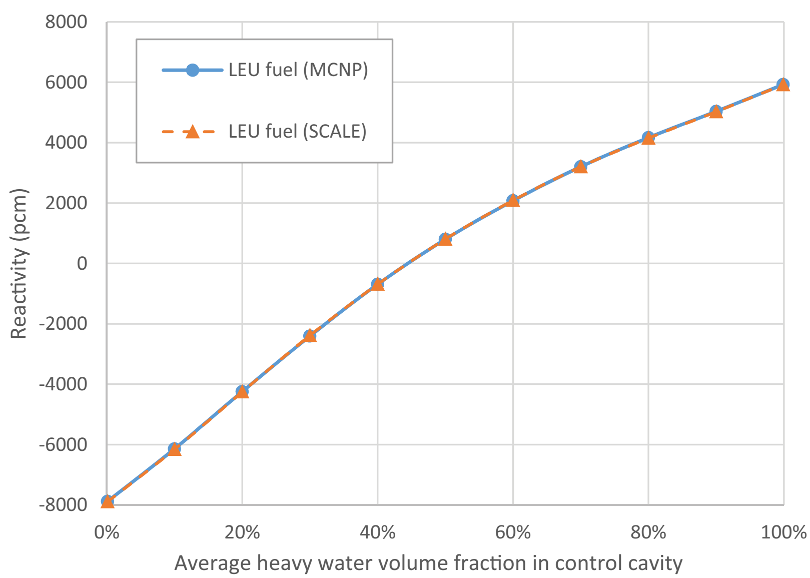

Fig. 11. Reactivity control worth of CCS in LEU-fueled MUBR.

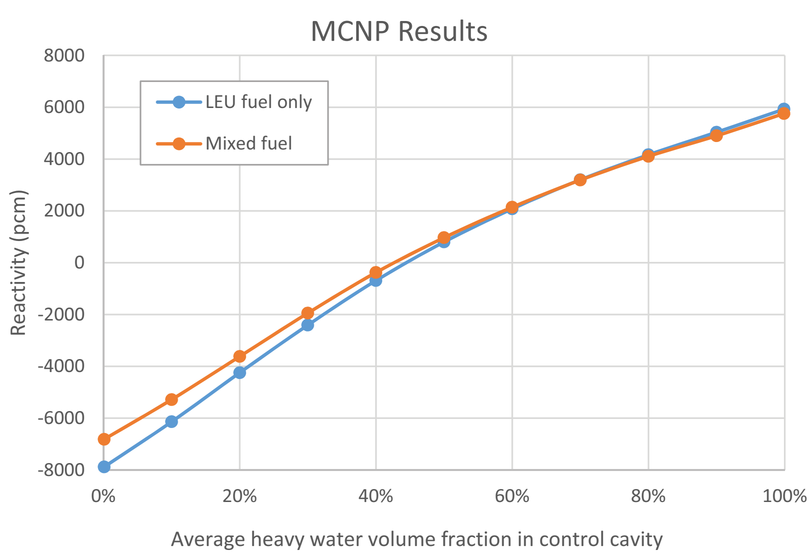

Fig. 12. Control worth curves for LEU and mixed-fuel cores of the MUBR design.

are noticed from the LEU core to the mixed-fuel core at conditions with lower heavy water fractions. The fissile isotope in the LEU fuel is only ^{235}U while the mixed LEU-UNF fuel contains ^{235}U, ^{239}Pu, and ^{241}Pu. Therefore, it is not surprising that the reactivity change is not the same in both fuels when the fuel is actually grossly undermoderated, which causes different reaction effects of the fuel compositions when neutron moderation is decreasing and energy spectra are hardening. Besides this difference, both curves have confirmed a large reactivity control range (>12 500 pcm) for the model problem core.

III.F. Burnup and Fuel Cycle Analysis

MUBR6gen performs the fuel burnup calculations at each major step with the correct liquid moderator level adjusted to ensure the critical status at the step. If the critical status is not maintained, the burnup calculation results would deviate because the ratio of neutrons causing fission to neutrons causing conversion is not correctly counted. This effect may be minor for small burnup but could be very significant for burnup for a few decades, which of interest to us as the desired fuel cycle for the MUBR design is 80 years.

The first 10 years of burnup calculations on the small-scale, LEU-fueled MUBR core (the model problem) were performed via the MUBR6gen platform with the burnup estimate up to ~75 MWd/kg HM. Figure 13 shows the evolution rate of concentration for some major actinides in the LEU core. Burnup results from both the MCNP and the SCALE codes are presented in a comparative manner to show generally acceptable agreement between the two predictions. Since MCNP does not have the fission product removal capability, the SCALE burnup results shown in Fig. 13 consider zero fission product loss for the purpose of achieving consistent results as that of MCNP. Note that the original unit of the nuclide concentration in the burnup calculation is atoms/barn·cm, but the numbers shown in Fig. 13 are normalized with the arbitrary unit for the purpose of better display. Also, note that some nuclide concentrations such as ^{236}U and ^{242}Pu were adjusted by a factor of 10 or 100 in order to realize a manageable scale to be viewable in the same plot.

III.G. Summary of the Model Problem Analysis

The model problem assembles a representative example of the MUBR design. It serves as a test bed for the MUBR6gen program to evaluate its capability of conducting standard neutronics analysis in a code-to-code verification mode. All the neutronics calculations presented and discussed above confirm that MUBR6gen successfully established a reliable and flexible MUBR analysis pipeline. It also builds up sufficient confidence to move the modeling and simulation efforts to the next phase and perform more extended analysis on a variety of MUBR designs.

IV. EXTENDED ANALYSIS OF OTHER DESIGNS

The flexibility of the MUBR6gen tool has allowed evaluation of the MUBR concept in many size variations ranging from microreactors through small modular reactors (SMR) to large grid-scale reactors with many variations in materials and fuels. The results of this type of analysis can confirm that certain results are robust in that they occur in both MCNP and SCALE analyses over a wide range of simulated reactor sizes and fuel fissile

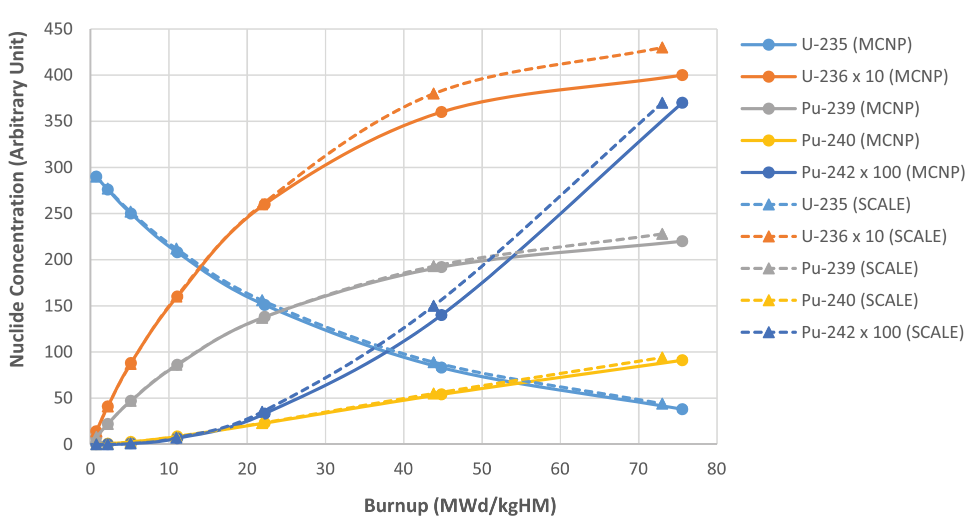

Fig. 13. Major actinide concentration changes along with burnup in MUBR.

contents. MCNP does not provide the ability to simulate continuous fission product removal from circulating fuel, so long-term burnup analysis with continuous fission product removal can currently be modeled only in SCALE.[14] We have gained confidence in the SCALE results because the MCNP and SCALE results are close when simulating burnup with no fission product removal. Also, a preliminary investigation shows that with different rates of fission product removal simulated with SCALE, the results are varied as expected with the changes in the simulated rate of fission product removal.

In the extended analysis, we analyzed 10 different MUBR configurations with the characteristic design parameters shown in Table III. The total power rate of each configuration is estimated based on the power per fuel tube limited by the rate at which the uranium can be pumped through the tubes. As the number of fuel tubes increases, the total fuel mass does not increase in exactly the same proportion because of different external geometries such as heat exchangers, fuel pumps, and connecting tube lengths that exist for different configurations.

The fuel tubes are configurated in a hexagonal array (regular for even cases and irregular for odd cases), so the reflector diameter is designed sufficiently large enough to contain the resultant hexagon of each case. Two alternative fuel loadings are considered for each configuration: LEU only or standard LEU mixed with UNF. The two alternative fuel loadings are of interest in this study because we intend to understand the breeding capability of MUBR at different configurations and to achieve the desired fuel cycle length with a given amount of UNF. In other words, we are interested in knowing the fuel life and how much of the reactor power comes from the fission of ^{238}U (either directly or indirectly) with various MUBR configurations.

In Table III, the initial liquid D_2O% (heavy water) column is the target initial liquid D_2O% level in the control cavity. The two columns next to it represent the two different fuel options and the fuel values that we estimate to lead the liquid D_2O% to be near the target. Note that all the design values shown in the table are proposed based on current neutronics estimations and are subject to change. In all cases the fuel tube diameter is 42 cm, and the reflector height is 192 cm, while only Case 0 has a different diameter for the control cavity.

Considering that the fission product is continuously removed from the core because of gas evaporation, the initial values for the liquid moderator level in each case should be adjusted to more optimum values based on the fission product removal results. These adjustments will require new initial values of ^{235}U wt% or UNF% in the fuel, which would reduce the initial ^{235}U wt% and increase the UNF%, giving even more advantageous results.

The performance for which we are striving is that MUBR will operate within the control range for 60 to 100 years without refueling, that most of the power comes from direct or indirect fission of ^{238}U, and that the fuel is good for 200 years or more. Therefore, we examined only the results with conservative fission product removal rates at this moment. The results however vary somewhat with reactor size; the smaller ones have higher peripheral neutron loss and slightly lower performance. For the reactor in which we are most interested [Case 4 with 19 fuel tubes and a nominal power of 334 MW(thermal)], we have the following results:

TABLE III

Characteristic Design Parameters of 10 Different MUBR Configurations

Case	Fuel Tubes	Power [MW(thermal)]	Reflector Diameter (cm)	Initial Liquid D_2O%	Initial LEU ^{235}U wt%	LEU 4.95 wt% + UNF%	Description
0	1	23	228	70.0	6.01	n/a	Prototype
1	3	72	256	20.0	4.47	13	Microreactor
2	7	139	316	35.0	3.62	24	SMR small
3	14	246	388	40.0	3.15	36	SMR mid
4	19	334	430	45.0	2.98	40	SMR large
5	30	493	504	50.0	2.74	49	Grid small
6	37	608	544	55.0	2.60	51	Grid mid
7	52	818	618	60.0	2.47	54	Grid large
8	61	961	658	65.0	2.32	56	Grid larger
9	80	1222	732	70.0	2.27	57	Grid largest

- With initial fuel that is 2.98 wt% ^{235}U at 100 years, the burnup is 7.30% of the initial heavy metal (IHM), and 99% was from fission of ^{238}U.
- With initial fuel that is 40% UNF mixed with LEU (4.95 wt% ^{235}U) at 100 years, the burnup is 7.30% of the IHM, and 97% was from fission of ^{238}U.
- With initial fuel that is 40% UNF mixed with LEU at double power at 100 years, the burnup is 14.6% of the IHM, and 96% was from fission of ^{238}U.

These results show that even with a modest-sized reactor, we can use UNF as 40% of the fuel, and we can get most of the power from fission of ^{238}U. So, at the end of reactor life, the used fuel still has almost as much fissile content as the initial fuel and is an asset, not a liability.

Additional simulations can be and have been performed only in SCALE using MUBR6gen to estimate the effect of fission product removal on fuel burnup and how sensitive the results are to different values of the hypothetical half-life of fission products due to removal by evaporation. The simulations have been performed with mixed LEU with LWR UNF fuel with three choices of decay rate: the slightly optimistic half-life of 2.5 days, the slightly pessimistic half-life of 10 days, and the very pessimistic half-life of 40 days. The results indicate that with the most pessimistic half-life of 40 days, even the microreactor would be able to operate without refueling for 127 years, fission 9.27% of the heavy metal, and generate 85% of its energy from direct or indirect fission of ^{238}U. All of the larger sizes were simulated to 255 years and fission 18.6% of the heavy metal with from 94% to 102% of its power from direct or indirect fission of ^{238}U (the largest case produces more fissile material than it uses). These results are much better with the less pessimistic values for fission product removal half-life. Based on these results, MUBR produces over 10 times as much power from each ton of fuel as a standard LWR, produces only one-tenth as much UNF per megawatt hour of electricity as a standard LWR, requires vastly less mined uranium, and can produce ten times as much power from LWR UNF as it originally produced.

V. LIMITATIONS OF CURRENT ANALYSIS

In the course of MUBR neutronics analysis, some modeling and simulation limitations have been identified. Some of these limitations are due to methodology deficits, and some of them are because of the infancy status of MUBR6gen program development. We summarize these limitations in the section to give the reader a clear heads-up on the interpretation of the simulation results.

First, MUBR contains a flowing fuel that is rapidly cooled by the primary heat exchanger, which makes all of the fuel in the reactor have the same composition at any point in time. The fuel-flowing effect cannot be simulated realistically in MCNP and SCALE. So, delayed neutrons are not properly treated, and some will drift outside of the core along with the flowing fuel and are not modeled in the present study. However, we believe that the actual effect of drifted delayed neutrons on the k_{eff} is small for MUBR and certainly well within the reactor control range.

Second, the current burnup analysis is not valid for long-term burnup because both MCNP and SCALE do not deal with any reactor control system simultaneously, and thus, the modeling state does not fully reflect the reality of a reactor configuration following control element adjustment to compensate the reactivity change due to fuel burnup. Our analysis compensates this effect partly by performing a long burnup in separate major steps. Each major step is then treated to operate at a steady state in MCNP or SCALE and may include minor steps. At the end of each major step, MUBR6gen reads the output file, determines the new fuel composition, creates a new input file, and performs a new kcode analysis to estimate the k_{eff} value. If it is not close enough to 1.00000, then the level of liquid moderator in the control cavities is adjusted accordingly, and the k_{eff} determination is repeated as necessary. Then, a new major burnup step is activated, and the process is repeated until all of the specified major steps are executed or no value of liquid moderator makes k_{eff} near enough to 1.00000, which means the new fuel reactivity is outside the range that the control system can handle. This burn analysis works equally well in MCNP and SCALE and provides fuel compositions at each burn step that are close enough so that we have confidence in the results.

Third, these results do not 100% reflect physical reality because in the physical MUBR, fission products evaporate from the surface of the molten uranium metal fuel as it passes from the reactor core to the heat exchanger, which cools the fuel and heats the heat transfer fluid to transfer the thermal energy to the power-generating system. The fission products include species of elements ranging from 30 to 70, which have different boiling points and solubility in molten uranium, both of which affect the rate of evaporation. MCNP currently has no provision for modeling separation of fission products from the fuel, whereas SCALE works around this problem by imposing a fictitious decay rate to the fission product and the rate can be specified separately for each element.[14] MUBR6gen can generate the SCALE input with a specified rate of removal for each element. In the absence of any experimental data about the rates of evaporation, one parameter is used as a prior to specify a half-life in days for a hypothetical element with a boiling point equal to the temperature of the molten uranium fuel as it enters the heat exchanger. For each real fission product element, the specified half-life is adjusted based on the difference between the actual boiling point of the element and the fuel temperature.

VI. CONCLUSIONS AND FUTURE WORK

This paper performs standard neutronics analysis on the novel MUBR concept using the MUBR6gen program, which essentially employs the MCNP and SCALE codes to do the neutronics calculations. The MCNP and SCALE simulation results presented in the paper indicate that the MUBR concept has very attractive neutronics features in that most energy from nuclear fission can come from the plentiful ^{238}U instead of the scarce ^{235}U. These results have major implications for the nuclear fuel supply chain, the nuclear waste issue, and nuclear plant operation. Additional SCALE-only simulations on fuel burnup analysis with continuous removal of some fission products show that with even modest rates of fission product removal, the results are much better, and the MUBR fuel life may extend for centuries, not decades.

At the present time, MUBR is only a reactor concept, and the analysis has been largely limited to neutronics and fuel burnup. These analysis results suggest that the MUBR concept has many advantages and is worth additional investigations and efforts to progress from a conceptual design to an engineering design. This will involve more detailed descriptions of the many components not discussed in the conceptual design such as pumps, insulation, sensors, and so on. These components may have a low effect on neutronics but a large effect on cost and reactor performance.

In addition, heat flow by conduction and radiation between different components needs to be analyzed. For fluid circuits, a thermal-hydraulic analysis must be done for each circuit at each power level. As other components are added, the neutronics analysis must be expanded to include the effects of radiation and neutron flux on each of these components. The costs of reactor materials and manufacturing also need to be evaluated. The results of all these analyses may suggest changes in the design, which will require repetition of all of the above analysis steps.

There are no experimental data on the rates of fission product removal by evaporation from molten uranium metal fuel, and experimentation on this topic is urgently needed. We have proposed the use of silicon carbide for the fuel tubes in the MUBR design. Experimentation is needed to determine the durability of silicon carbide fuel tubes in a high-temperature, high-radiation environment with circulating molten uranium metal fuel.

Disclosure Statement

No potential conflict of interest was reported by the author(s).

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