RESEARCH ARTICLE

REVISED Molten Uranium Breeder Reactor (MUBR) and Its Development Steps

[version 2; peer review: 2 approved with reservations]

Neal Mann [1], Zeyun Wu [2], Mihai (Mike) G. M. Pop [3]

[1]Neal Mann and Associates, Washington, DC, 20002, USA
[2]Virginia Commonwealth University Department of Mechanical and Nuclear Engineering, Richmond, VA, 23284, USA
[3]Areva College of Experts (Retired), Alexandria, VA, 22309, USA

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Abstract

Background

The Molten Uranium Breeder Reactor (MUBR) is a radical new mixed-energy spectrum breed and burn reactor concept. The MUBR is fueled with molten uranium metal fuel in large fuel tubes instead of thin fuel rods, and is cooled by circulating the molten fuel through a heat exchanger. The purpose of this research is to evaluate MUBR configuration variations with SCALE (RSICC request/license 203869) to show that the results are robust and that the simulated burnup is at least 10 times greater than the initial fuel fissile content without any refueling and holds up even if some of the assumptions are off.

Methods

A proprietary computer program uses parameters to generate MCNP (RSICC request/license 176034) or SCALE input files and initiate MCNP or SCALE burn simulations and other analysis. This allows relatively fast comparison of different parameters to verify that the design is robust.

Results

MUBR SCALE burn simulations through 120 years of fuel life show a burnup of 35% of the initial fuel mass when the initial fuel was Low Enriched Uranium (LEU) (3% U-235).

Conclusions

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1. **Andang Widi Harto**, Universitas Gadjah Mada, Yogyakarta, Indonesia

2. **Güven Tunç**, Gazi University, Ankara, Turkey

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These results suggest that if it can be constructed and operated reliably for a long time, the MUBR would be a true breed and burn reactor. Compared to Light Water Reactors (LWR) this would provide many advantages for the nuclear industry and the world such as: reactor operation without shutdowns for refueling or fuel manipulation, no need for special fuel, 7 times as much energy per ton of fuel, one seventh as much Used Nuclear Fuel (UNF) per megawatt hour of electricity, no UNF storage required every two years, increased energy security, and reduced nuclear proliferation risk. While development of the MUBR concept from a concept to a prototype reactor will be costly, the advantages suggest that the concept merits further study.

Keywords
Molten Uranium Breeder Reactor, Control Cavity Structure, LWR, UNF

Corresponding author: Neal Mann (nealgmann@gmail.com)

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REVISED **Amendments from Version 1**

Changes from the initial paper submission include simulation results of several new variations to show robustness of the results and emphasizing discussion of details of the breed and burn design and what is needed to develop it from a simulated concept to a prototype reactor. Figure 1 has been replaced to match the simulated reactor, figures 2 and 3 have been updated to make them easier to read, figure 4 has been eliminated as it is essentially a duplicate of the new figure 1. Figure 5 has been updated and renamed as figure 4 and figure 6 has been updated and renamed as figure 5. A new figure 6 has been added to show a graphic display of the main results. Section 2, Potential benefits of MUBR has been eliminated and section 2 is now methods, section 3 is now results, and section 4 is now conclusions and future work and includes much of the former section 2 on benefits of the MUBR. Because many new simulations have been added to show that the design is robust, they are summarized in section 3 – results and the detailed tables of the results are in section 5, an appendix of result details for interested readers. A new reference has been added.

Any further responses from the reviewers can be found at the end of the article

1. Background

The Molten Uranium Breeder Reactor (MUBR) is a radical uranium-plutonium mixed energy spectrum breed and burn fission reactor concept proposed by Mann et al.[1–5] It is an advanced uranium-based fission reactor that has significant advantages over other commonly known advanced reactors. The MUBR operates 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 created. Breeding is effective in the MUBR because of the large fuel tubes, the harder neutron spectrum, and low neutron loss, which also leads to an initial fuel with a low fraction of fissile content (2.3 to 4.5 wt.%, the initial fuel composition varies with the reactor size because larger sizes have lower peripheral neutron loss). The MUBR concept is based partly on concepts from existing CANDU[6] and Molten Salt Reactor (MSR) designs.[7] The specific design features of MUBR are detailed later in this section but are summarized as follows: (1) large fuel tubes instead of thin fuel rods, (2) the fuel is molten uranium metal, which is cooled by circulating it through the core and an external heat exchanger, (3) the moderator and reflector are heavy water, (4) the reactor is controlled over a very wide fuel reactivity range by negative feedback, and (5) some fission products are continuously removed from the circulating molten fuel because they evaporate from the fuel at the high fuel temperature (1200 °C–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 major components at the system level. Each fuel tube is surrounded by control cavities that are closed at the top and sides and contain heavy water moderator. The cavities are open at the bottom so that the heavy water can travel freely to and from the bottom heavy water reflector and are closed at the top and sides; thus, a bubble of heavy water steam is trapped at the top of each cavity.[8] The size of the bubble is controlled by negative feedback and essentially provides a void coefficient of reactivity that can vary from 0 to 100% of the cavity size.

The five main features of the MUBR are discussed in more detail in the 5 subsections below.

1.1 Large fuel tubes

The MUBR has a small number (19 in the base case discussed in this paper) of large fuel tubes instead of the many thousands of thin fuel rods in conventional light water reactors. Figure 1 is rendered by the SCALE view feature and shows some of the key features; therefore, it can be considered as an accurate representation of the simulated MUBR. As shown in Figure 1, the large fuel tubes are arranged in a hexagonal array in the top view, and the fuel is shown in red. The large diameter portion of the fuel tubes shown has an inside diameter of 42 cm with a pitch (center to center spacing) of 66 cm. The fuel tubes are tapered at the top and bottom, where they leave the reactor core and reflector. Each fuel tube is surrounded by a hexagonal control cavity around the wide portion of the fuel tube. The reflectors are located on the sides of the array of control cavities, as well as above and below them. The moderator and reflector are heavy water, shown in dark blue, but at the top of each cavity is a trapped bubble of moderator steam, shown in pale blue (there is also a layer of moderator steam above the top reflector to stabilize the reflector and moderator pressure). The top view is on a cross-section slightly above the middle of the reactor core, so it shows the control cavities at a level that shows the steam bubble in the control cavities of the central tubes but below the steam bubbles in the side control cavities. The MUBR fuel is molten uranium metal cooled by circulating it through a heat exchanger and fuel pump, as shown on the right side of the diagram in Figure 1.

Figure 2 illustrates the neutron flux distribution in a representative fuel tube of a MUBR from a neutronics simulation using the ORNL SCALE code.[9] The neutron flux is generally higher at the bottom of the fuel tube, where the moderator in the control cavity is liquid, and the fast neutrons are well moderated. At the top, the moderator is low density moderator steam; therefore, the thermal neutron flux is lower (for this simulation, the liquid moderator level was near the center of the fuel tube height). The radial fast neutron flux is fairly uniform. Thermal neutrons are absorbed quickly in the fuel;

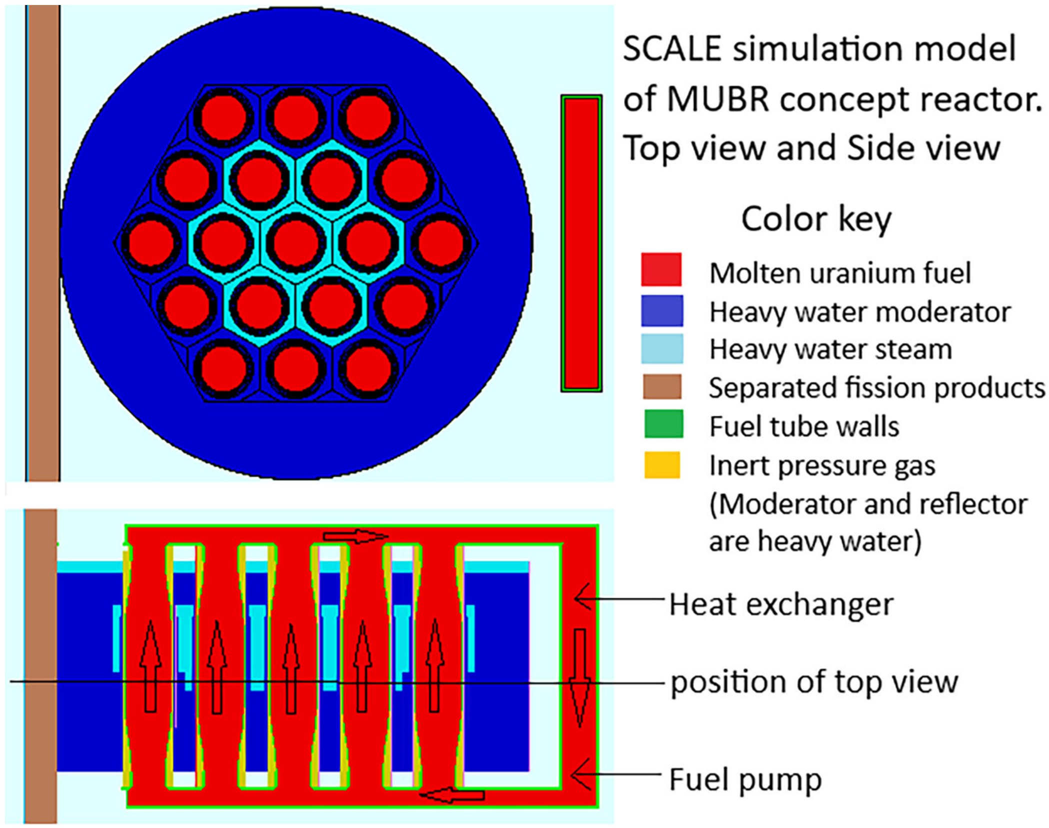

Figure 1. A conceptual diagram of the main components in the MUBR.

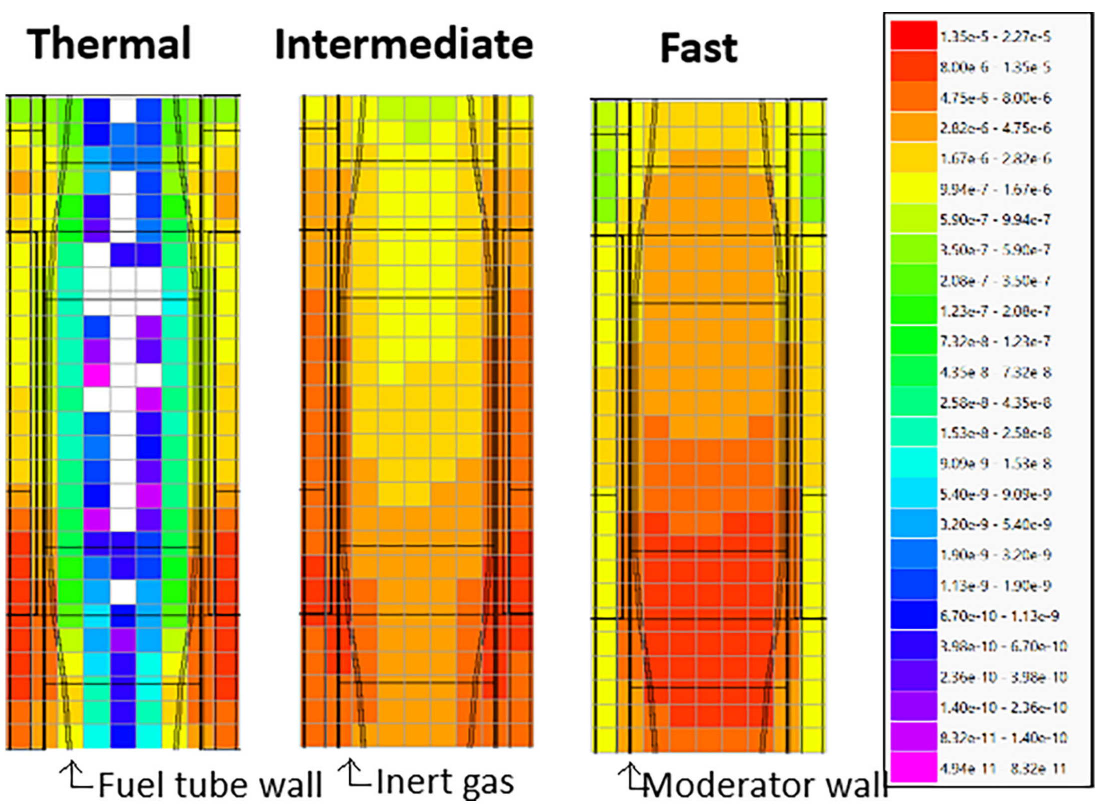

Figure 2. Neutron flux in a representative fuel tube.

therefore, the thermal neutron flux is very low in the large central portion of the fuel tube and high only in the moderator and fuel near the sides of the fuel tube.

The flux distribution and the high conversion ratio of the MUBR can be understood by thinking about some details revealed by the simulations. With MCNP simulations, each time a calculation of K_{eff} is done, MCNP includes in the output file some results of the K_{eff} simulation. These include: the estimated K_{eff} and the estimated error; the % of fissions caused by thermal (MUBR ~ 45%), intermediate (MUBR ~ 20%), and fast neutrons (MUBR ~35%); the energy cutoff for each range; and the average number of neutrons produced per fission (MUBR ~ 2.50), From the energy cutoffs the energy of the fastest thermal neutron is 0.625 ev (electron volts) and the energy of the slowest fast neutron is 100,000 ev. The ratio of these values is 160,000 to one. Since the energy is proportional to the square of the speed and the square root of 160,000 is 400, the speed of the slowest fast neutron must be 400 times the speed of the fastest thermal neutron. This means that in the time it takes for the fastest thermal neutron to travel the 2 cm between the moderator and the fuel a fast neutron travels at least 8 meters. In a conventional light water reactor with fuel rods having a diameter of around 1 cm, the emitted fast neutrons only travel a short distance before leaving the fuel and entering the moderator. In the MUBR with a fuel tube diameter of 42 cm, the average distance traveled by a fast neutron before it leaves the fuel is around 40 times as much as in a light water reactor so the probability of causing a fast fission is much higher. This is why light water reactors are thermal reactors and the MUBR is a mixed spectrum reactor. Because there is around 45% thermal fission and 35% fast fission, the ration of fast fission to thermal fission is around 0.78. However, the fast fission is caused by fast neutrons from all fission events so each thermal or intermediate fission causes a rapidly diminishing cascade of fast fissions. If the cascade was not rapidly diminishing the fast fissions would be greater than half of all fissions but it is actually nearer to a third. This cascade of fast fissions dies down to negligible amounts after around five steps. Each fast fission costs one neutron but produces ~ 2.5 neutrons for a gain of around 1.5 neutrons per fast fission. Because there is so much U-238 (initially 97%) and many of the fast neutrons have enough energy to cause fast fission of U-238, some of the fast fissions will be of U-238, reducing the number of fissions of fissile isotopes required to reach the simulated power level and therefore reducing the number of converted fissile atoms required to maintain the conversion ratio. This discussion is not intended as a calculation of any values during the burnup, it is intended to help in understanding why the SCALE burn simulations give the results shown in the section on results.

1.2 Molten uranium metal fuel

MUBR Fuel is molten uranium metal cooled by rapidly circulating it through fuel tubes, a heat exchanger, and a MHD fuel pump. The fuel is always well mixed and has the same composition throughout the reactor. This allows (and requires) all the fuel to be treated as a single body (or mix) for the simulations. To maintain the fuel temperature within a tolerable range, the reactor power per fuel tube is limited by the flow rate of the circulating fuel. The heat transferred from each fuel tube is the specific heat of the fuel multiplied by the fuel density multiplied by the fuel flow rate multiplied by the temperature difference between the cooler fuel entering the bottom of the fuel tube and hot fuel leaving the top of the fuel tube. The specific heat and fuel density are the material properties of the fuel, over which we have little control. The flow rate is determined by the hydraulics of the fuel circuit and pump pressure. This can be adjusted by changing the power supplied to the pump (up to the design limit). Uranium melts at 1132 °C at atmospheric pressure; therefore, the inlet fuel temperature must be high enough to allow the fuel to flow freely. The simulated low-fuel temperature is 1200 °C. A higher outlet temperature is limited by the ability of the fuel wall material to withstand it. The simulated fuel wall material is silicon carbide and the simulated upper temperature is 1400 °C for a temperature increase of 200 °C. The fuel composition changes slowly with burnup, which in turn changes the thermal and fluid properties of the fuel. The simulated power is based on 8 MW thermal power for each ton of fuel and the simulated MUBR base case has around 9,839 L of fuel with a mass of 167,274 kg and a thermal power of around 1336 MW.

The simulated fuel for most of the simulated cases is Low Enriched Uranium (LEU) which is 3.000% U-235 and 97.000% U-238 (plus trace amounts of other uranium isotopes). To produce this fuel from mined uranium requires fuel enrichment where X tons of mined uranium with 0.72% U-235 are converted to 1 ton of fuel with 3.00 % U-235 and X - 1 tons of depleted uranium. If we assume that the depleted uranium has 0.20 % U-235, then the total U-235 is X * 0.72 = 1 * 3.00 + (X – 1) * 0.20. Simplifying this gives X * (.72 - .20 = 3.0 -.20 which gives X * .52 = 2.8 which gives X = 2.8/.52 = 5.38 so it takes 5.38 tons of mined uranium to produce each ton of MUBR fuel (3.00% U-235). The same calculation for LWR fuel (4.95% U-235) shows that it takes about 9.13 tons of mined uranium to make 1 ton of LWR fuel, or 1.7 times as much mined uranium per ton of fuel. The MUBR initial fuel can also be mixture of 50% lightly treated LWR UNF and 50% LWR uranium fuel (LEU 4.95% U-235). "Lightly treated" means that the UNF is reduced from oxides to metal which is the same process that turns uranium ore from uranium oxides to uranium metal. MUBR fuel created by this mixture still requires 0.50 * 9.13 = 4.565 tones of mined uranium per ton of mixed MUBR fuel (15% less than the 5.38 tons of mined uranium required for the usual 3.00% MUBR fuel) and it allows the MUBR to dispose of 0.5 tons of LWR UNF for each ton of MUBR fuel. Since the LWR UNF is mostly U-238 and the MUBR gets most of its energy from the U-238, this fuel option allows the MUBR to produce several times as much energy from the LWR UNF than it produced when it was originally used in a LWR.

1.3 Heavy water moderator and reflector

Heavy water is used as the moderator and reflector because it has a very low neutron absorption cross section and leads to low neutron loss. The low neutron loss occurs because the hydrogen atoms in the water have already absorbed a neutron, their cross section for absorbing a second neutron is very low, and the oxygen also has a very low neutron absorption cross section. This low neutron loss increases the conversion ratio and decreases the required fuel fissile content. This allows CANDU reactors to use natural uranium as fuel instead of the enriched uranium fuel in other reactors. This is a major factor in allowing the MUBR to have a true breed and burn fuel cycle and to have fuel with a relatively low fissile content.

1.4 Reactivity control structure

The conceptual control cavity shown in Figure 3 is a cylinder; however, the simulated control cavities are a hexagon with a hole down the middle for the fuel tube and a heat shield. The control cavities are closed at the top and sides but open into the bottom reflector at the bottom. Most of the heavy water moderator is in the array of control cavities, with a bubble of moderator steam trapped at the top of each control cavity. The moderator is heated by fast neutrons and cooled by a flow of cool moderator, which is pumped into the top of each cavity. The temperature of the liquid moderator is at the boiling point of heavy water at the pressure of the moderator steam. The rate at which steam is produced is proportional to the fast neutron flux because the heat shield blocks the thermal conduction of the heat. The rate of steam condensation is proportional to the moderator cooling flow rate multiplied by the temperature difference between the cooling flow and the steam temperature. If steam is produced faster than it condenses, the trapped steam bubble increases in size and displaces the liquid moderator into the reflector, where it moderates fewer neutrons, reducing the reactivity and fission rate. If steam condenses faster than it is produced, the steam bubble decreases in size, and liquid heavy water flows into the cavity, increasing the neutron moderation, and thus the reactivity and fission rate. Consequently, the fission rate and reactor power are proportional to the moderator cooling flow rate. The control method is very sensitive because a small mass of boiled water creates a large volume of steam and displaces a large volume of the liquid moderator. It is very fast because a change in the neutron production rate is transmitted by the fast neutrons to the moderator within a fraction of a microsecond. The practical effect of this control method is that excess neutrons are under-moderated and absorbed by resonance capture in U-238 to convert it to fissile Pu-239 to increase the conversion ratio, instead of being wasted by absorption in the neutron absorbers used in traditional control methods. The reactivity control range achieved by CCS is very large (~13,000 PCM) because of the large difference in reactivity between all liquid moderator and all moderator steam. This is not a safety concern because the moderator does not cool the fuel, unlike thermal reactors. Because each fuel tube has its own control cavity with its own liquid moderator level, the control method adjusts the power of each tube to nearly the same level, which flattens the radial power distribution.

Functionally, less liquid moderator means the neutrons are under moderated and the ratio of intermediate neutrons to thermal neutrons is increased. Intermediate neutrons have an increased probability of resonance capture in U-238 which converts the U-238 to Pu-239, increasing the conversion ratio while decreasing K_{eff} because the neutrons captured by U-238 are unavailable to cause thermal fission of the fissile U-235 (and Pu-239 and Pu-241). In all nuclear reactors operating at constant power the K_{eff} is exactly 1.00000 otherwise the reactor will quickly either fizzle down to no power or

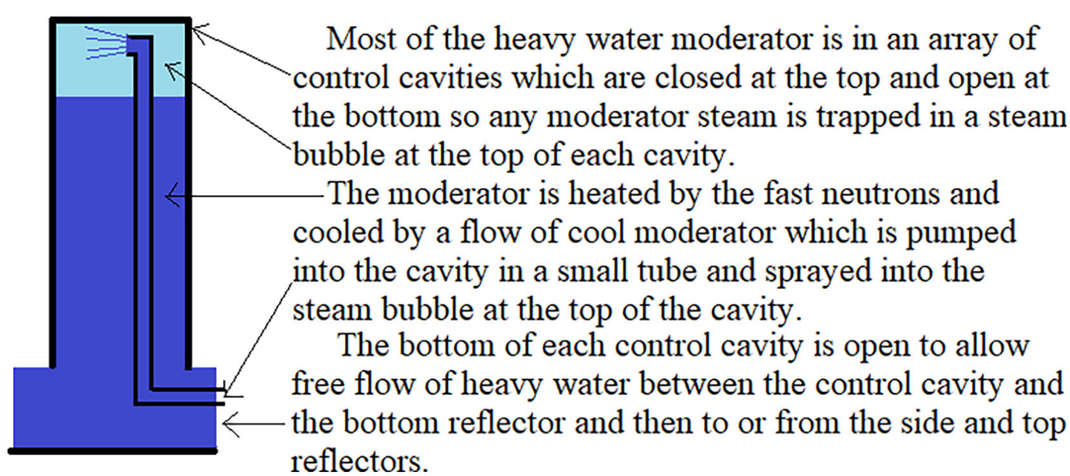

Figure 3. Conceptual view of a MUBR Control Cavity Structure (CCS).

the power will rapidly increase until it melts down. K_{eff} is maintained at exactly 1.00000 by a combination of naturally occurring negative feedback (void coefficient of reactivity, thermal coefficient of reactivity, etc.) and operator controlled effects such as changing the position of neutron absorbing control rods or adding soluble burnable poisons (neutron absorbers) to the moderator. In the MUBR this tradeoff of neutrons between K_{eff} and the conversion ratio causes some issues with the accuracy of long SCALE burnup simulations. To run a SCALE burnup simulation the input file specifies the composition, temperature, and density of all the materials in the reactor. Then the total burn time to be simulated is specified as one or more sub steps with the duration and power specified for each sub step. Then scale will create a neutron flux map (and calculate K_{eff}) and use the flux map to determine the change in fuel composition during the step. It will not change the temperature or density of the fuel or the moderator and will not change the position of control rods (in conventional light water reactors) or the height of the dividing line between liquid moderator and moderator steam (in the MUBR). In light water reactors this is not significant because decreasing the number of thermal neutrons does not change the flux map much and the fuel evolution by much. In the MUBR, if the simulated K_{eff} is greater than 1.00000 then in the physical MUBR some more neutrons will go to conversion and less to K_{eff} so the actual conversion ratio will be higher than the simulated conversion ratio and the actual fissile content will be greater than the simulated fissile content. The erroneous fissile content will cause additional errors in each succeeding step. If the simulated K_{eff} is less than 1.00000 then the actual fissile content will be lass than the simulated fissile content which will be propagated through subsequent steps. The error is slight if the simulated K_{eff} is close to 1.00000 but may become significant over a long burnup like 120 years.

1.5 Continuous fission product removal

The MUBR design contemplates the possibility of the continuous removal of fission products from circulating molten fuel. Some fission products may evaporate from the molten fuel at the top of the fuel circuit. Because uranium metal is denser than fission products, some fission products may be insoluble in molten uranium and may tend to rise to the top of the fuel circuit as dross. In either case, it may be possible to separate them from the circulating fuel, which has a temperature of 1400 °C at the top of the circuit.

Because the solubility of fission products in hot molten uranium is not well known, the current analyses assume no removal of any floating dross from the fuel. On the other hand, the boiling points of fission products are well known, so an estimate of the probable evaporation rate of each fission product can be made. Evaporation rates are likely to be proportional to the concentration of each fission product (which is equivalent to a radioactive decay rate), so SCALE can simulate the evaporation behavior. The formula used to estimate the evaporation rate of each fission product is based on the boiling point of the product relative to the simulated fuel temperature at the top of the fuel circuit and the overall removal efficiency factor. This factor is expressed as the number of days required to evaporate from the fuel half of a hypothetical element with a boiling point equal to the fuel temperature. Running burn simulations with different values of this parameter allows us to evaluate the effect of fission product removal on the long-term burnup simulation of the MUBR. The baseline conservative value of this parameter is five days, which is equivalent to saying that it would take five days for half of the alcohol to evaporate from a bucket of alcohol and water mixture heated to the boiling point of the alcohol.

Because the MUBR burn simulations contemplate a burn of over 35% of FIMA (Fuel Initial Metal Atoms) and each fission produces 2 fission product atoms, at the end of the burnup there will be 70 fission product atoms for each 100 atoms of initial fuel and the atoms not split will be reduced to 65 for each 100 atoms of initial fuel so the fission product atoms will outnumber the fuel actinide (heavy metal) atoms. Many of the fission product isotopes are strong neutron absorbers, so their presence could greatly reduce the fuel reactivity and the fuel life.

The proposed method of removing some fission products by evaporation requires that the top horizontal portion of the fuel circuit is not full of molten fuel, the fuel level is lower than the top of the horizontal pipe and there is an inert cover gas above it. The cover gas is circulated through the top of the fuel circuit and then a condensation column where the evaporated fission products are condensed out of the gas flow and collected while the cover gas goes back through the top of the fuel circuit. This physical arrangement also allows any insoluble fission products to accumulate as dross on top of the fuel in this part of the fuel circuit.

2. Methods

Previous work described a shell program (MUBR6gen.exe) which is driven by a large number of parameters. Default values of all the parameters are hard coded in MUBR6gen but all default parameter values can be overwritten with values contained in a parameter file. During the execution of MUBR6gen, the command line parameters can override both the default values and parameter file values of any parameter. MUBR6gen uses the parameters to determine the MUBR configuration to be studied and what is to be done, and generates the appropriate input file for either SCALE (version 6.3.1) or MCNP (version 6.2.0)[12] as requested. It then executes SCALE or MCNP, reads the output file produced, adds a line to a log file of program executions, produces a report summarizing the results, and can repeat the process with new values of some parameters. This allows the rapid evaluation of how changes in the parameters (such as fuel tube diameter, fuel enrichment, control cavity liquid moderator level, etc.) affect different simulated values. As described in the previous papers, the tool was used to maximize the conversion ratio in high burn simulations. This led to the base MUBR breed and burn reactor configuration studied in this paper.

MCNP does not have the capacity to run burn simulations which include continuous removal of fission products. Simulations to show K_{eff} at the limits of the control method were run with both MCNP and SCALE to show the reactivity range of the control method. Burn simulations were run with SCALE to determine what happens with a simulated burn which runs through fission of 35 % of the initial fuel atoms when only 3% were U-235 and the remainder (97 %) were U-238 and there was simulation of continuous removal of some fission products. Simulations were run with both MCNP and SCALE with no fission product removal.

In this paper, results of the burn simulation were examined to show what effect they have on evaluation of how the MUBR concept compares to other reactors. In addition, several new simulations were run with specific changes in the simulated design to show that the results are robust and the achieved burn is not much changed by various changes in the configuration or by potential errors in the assumptions.

3. Results

3.1 MUBR neutron flux

The three-dimensional neutron flux distribution can be mapped using both MCNP and SCALE once the K_{eff} value is determined. This was done in the case of the 19 tube SMR sized MUBR with a thermal power rated at approximately 1,338 MW used in the flux maps and burn simulations discussed below. Figure 1 shows a schematic view of the core configuration of the SMR sized MUBR. Figures 5 and 6 illustrate the top and side views of the base configuration flux maps produced by the MCNP and SCALE simulations, respectively.

The side view in Figure 1 shows the vertical position of the top view and shows in pale blue the regions where there is heavy water steam at the top of the reflector and at the top of the control cavities. The top view is at the level where the central control cavities have heavy water steam and the control cavities which surround the outside fuel tubes have liquid heavy water. The control cavities only surround the fat central portion of each fuel tube and the beginning of the tapered portion at the top and bottom of each fuel tube. The reflector region surrounds the array of control cavities on the sides and above and below the control cavities at the top and the bottom.

Figure 4 shows that the thermal neutron flux is high in the moderator, tapers off in the reflector away from the fuel tubes and rapidly tapers off in the fuel tubes so there are very few thermal neutrons in the large central portion of the fuel tubes. The intermediate neutron flux is high in the moderator, tapers off in the reflector and is slightly lower throughout the fuel tubes than in the moderator. The fast neutron flux is high throughout the fuel tubes, is slightly lower in the moderator and tapers off rapidly in the reflector.

Figure 5 shows again that the thermal neutron flux is very low in the central region of each fuel tube and in the moderator and is higher at the bottom where there is liquid moderator than at the top of the control cavities where there is moderator steam. The high thermal neutron flux at the bottom of the control cavities means that there is much greater thermal fission in the bottom outer portion of each fuel tube than in the upper portion of each fuel tube. The intermediate neutron flux is much more uniform than the thermal neutron flux but is lower in the fuel tubes than in the moderator and lower in the top of the fuel tubes than the bottom. The fast neutron flux is also higher at the bottom of the fuel tubes than the top and tapers off as it enters the moderator and reflector.

The MCNP simulations also provide the fission rate by neutron energy and indicate that each fission releases around 2.50 neutrons and that fission is caused approximately 45% by thermal neutrons, approximately 35% by fast neutrons, and approximately 20% by intermediate neutrons with some variation depending on the configuration and the stage of burnup. A fast fission event may occur from a fast neutron from a fission event provoked by a neutron of any energy. If the fast fission came from a fission event caused by a thermal or intermediate neutron, the fast fission is the start of a rapidly decreasing cascade of fast fissions. If the fast fission came from a fission event caused by a fast neutron, the fast fission is already part of a rapidly decreasing cascade of fast fissions. Since the fissions caused by fast neutrons are only 35% of all fissions, only a small part (less than 0.5) of the 2.50 neutrons emitted by a fission cause a fast fission. Since many of the fast neutrons have enough energy to cause fast fission of U-238 and they travel much farther in the fat fuel tubes than in thin fuel rods, there is much more fission of U-238 than in conventional light water reactors. This reduces the number of fissions of fissile isotopes required for the specified power rate and decreases the number of new fissile atoms that have to be converted to keep the conversion ratio near or above 1.00.

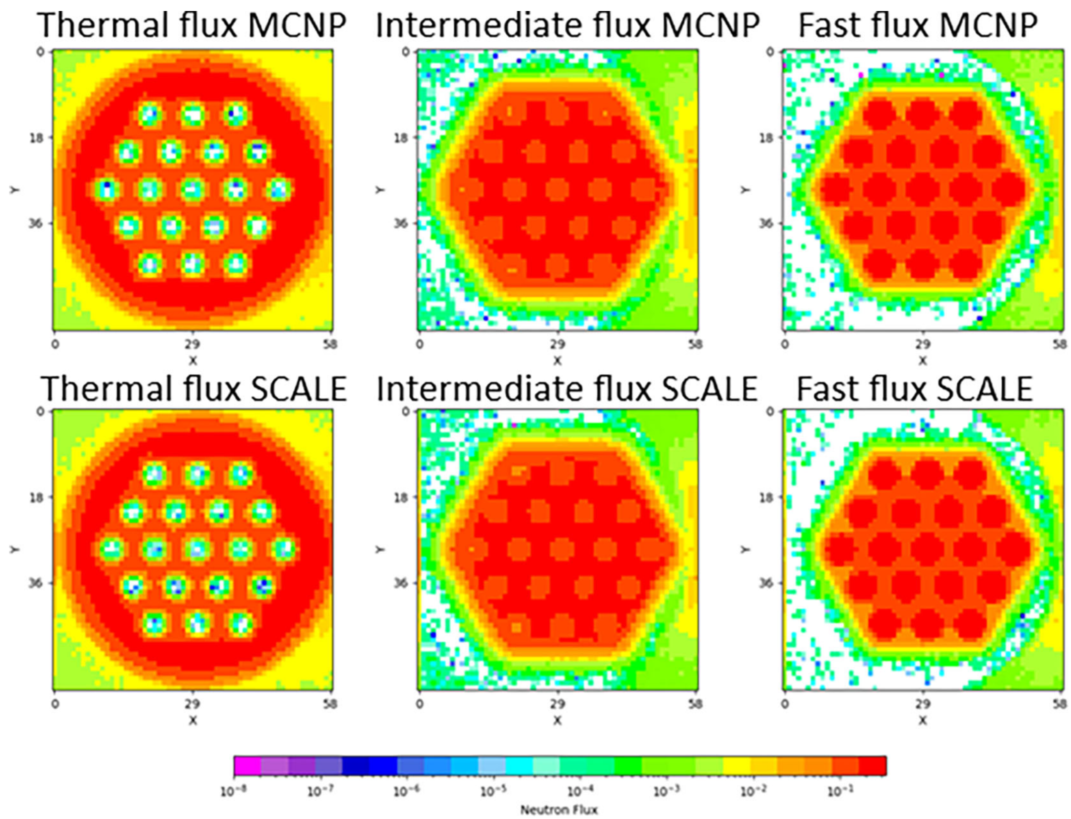

Figure 4. **MCNP and SCALE flux maps in a horizontal cross section (top view).**

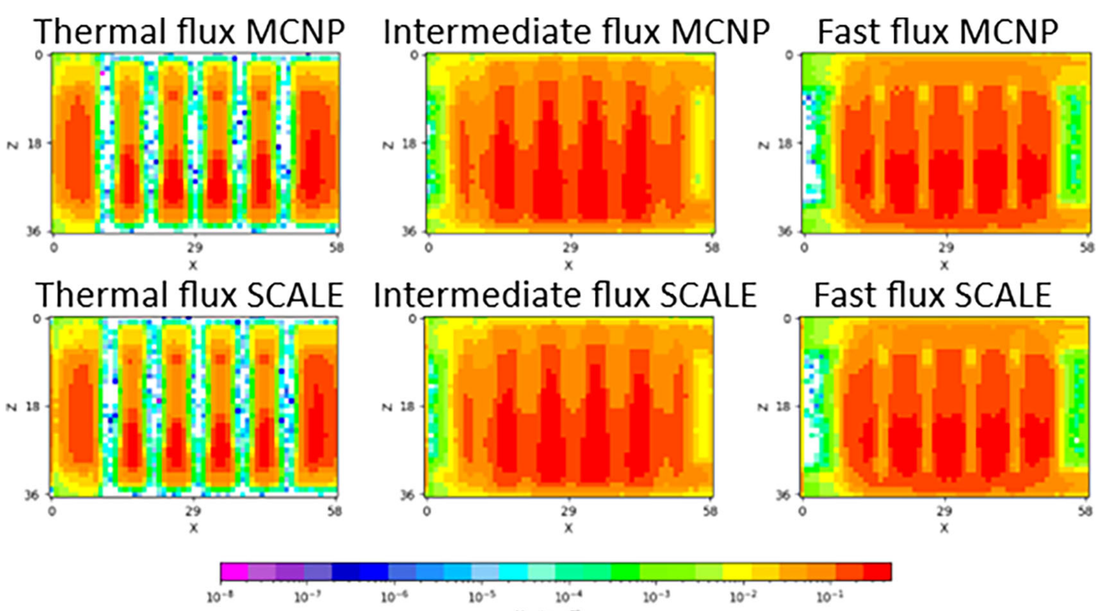

Figure 5. **MCNP and SCALE flux maps in a vertical cross section (side view).**

Because the MUBR is heavy water moderated and has low neutron loss, the fuel fissile content can be low compared with light water reactors. However, because most of the fissile content of the fuel is in the center of the fat fuel tubes and is not exposed to much thermal neutron flux, the fissile content required is higher than the unenriched uranium used to fuel conventional heavy-water (CANDU) reactors.

3.2 MUBR burn analysis

The results of the MUBR analysis with SCALE and MCNP are presented in this section. These simulations were performed for the 19 fuel tube SMR version of MUBR with 167274 kg of initial fuel that contains LEU with 3.00 weight% U-235 and produce a thermal power of 1,338 MW. The large diameter central part of each fuel tube has a fuel diameter of 42 cm and a fuel wall thickness of 1.00 cm. The center to center spacing of the fuel tubes is 66 cm.

Table 1 below demonstrates the wide control range of the MUBR control method by simulations with 0.1% liquid moderator in the control cavities, followed by 99.9% liquid moderator using both MCNP and SCALE. Tables 2 through 11 and 14 and 15 show the results of burn simulations with variations to show that the results are robust.

As shown in the table, for both the SCALE and MCNP simulations with essentially no liquid moderator, K_{eff} was approximately 0.92900, or a deficit of 7100 PCM. For both simulations with essentially all liquid moderators, K_{eff} was approximately 1.06500 or a surplus of 6500 PCM. Both results suggest that the effective control range by negative feedback is over 13,000 PCM for this MUBR configuration.

Figure 6 below shows the principal results of a 120 year SCALE burn simulation of the basic MUBR configuration.

This graph shows the values of the most interesting variables in the MUBR SCALE burn simulation. The burn was simulated at a constant power of 8 MW per ton of fuel for 120 years in steps of 1, 2, 3, 4, and 5 years for the first 15 years and then 7 steps of 15 years for the remainder of the simulation. The graph is linear in both dimensions and is 0 to 120 years horizontally and from 0 to 3.35 vertically. The black vertical lines in the graph are the steps in the simulation and represent the times when SCALE reports the simulated values. The black horizontal lines in the graph are to highlight the values 1, 2, and 3. The green line at the top is the fuel fissile content (U-235 + Pu-239 + Pu-241). It starts at 3.000%, all of which is U-235. It rises to a maximum 3.233% at 60 years and doesn't fall below the initial 3.000% until after 90 years of burnup, showing that the average conversion ratio is greater than 1.0000 for the first 90 years and the simulated MUBR is operating on a true breed and burn fuel cycle. The fuel has enough fissile content to run another 30 years.

Table 1. The steady-state condition at different liquid moderator levels.

Code	K_{eff} @ 0.1% Liquid	K_{eff} @ 99.9% Liquid	K_{eff} difference
SCALE	0.92960 ± 0.00140	1.06490 ± 0.00110	0.13530
MCNP	0.92854 ± 0.00106	1.06690 ± 0.00115	0.13836

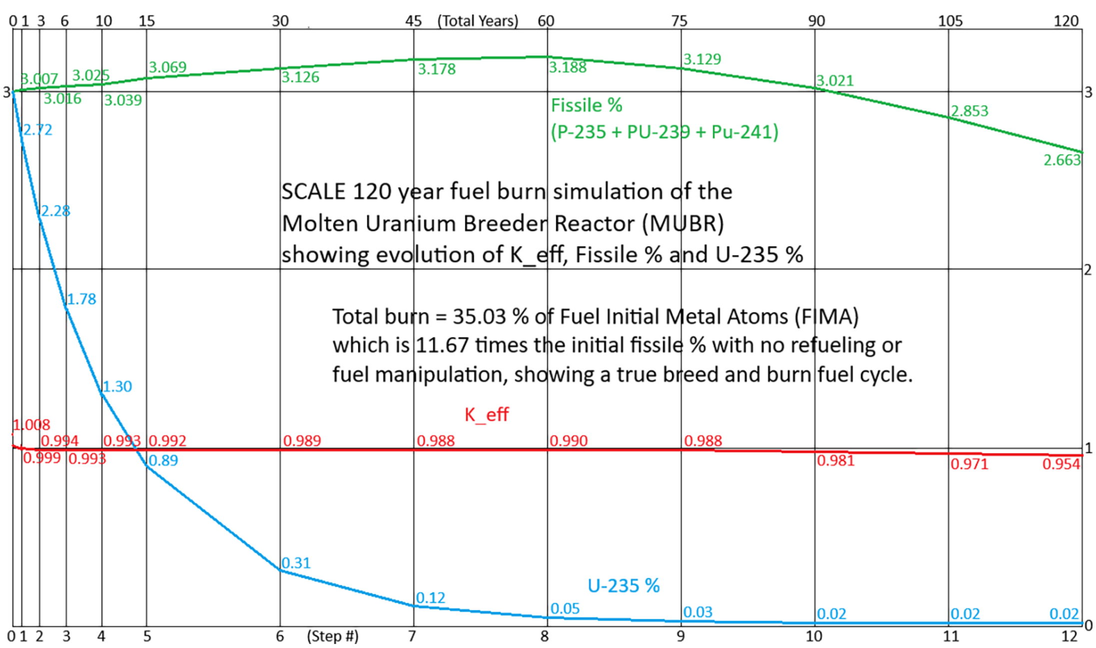

Figure 6. Principal results of SCALE burnup base case (input file inp9194p.inp).

The blue line is the U-235 content which starts at 3.000% and declines to below 1.0% during the first 15 years of simulation. It declines more slowly after that as most of the thermal fission is of the converted Pu-239.

The red line is the simulated value of K_{eff}. In a physical reactor operating at constant power, K_{eff} is always 1.0000, maintained by a combination of negative feedback (void coefficient of reactivity, temperature coefficient of reactivity, etc.) and operator controlled effects such as the use of neutron absorbers. SCALE is not capable of simulating any of this so the SCALE calculated value of K_{eff} will vary from 1.0000, the simulation is assumed to be valid if the simulated K_{eff} value is not too far from 1.0000. The MUBR control by negative feedback has a very wide range of control (greater than from 0.940 to 1.060 as shown in Table 1 above). In the graph K_{eff} starts at 1.008 and decreases to 0.998 in the first year, to.994 by year 3 and stays in the range 0.990 to 0.994 through year 75, then starts dropping off to 0.959 at year 120, still within the control range.

Table 2 shows the results of the SCALE simulation of the base case and was the source of the data used to create the graph in Figure 6 discussed above.

The simulated K_{eff} value was within 1,000 PCM of 1.00000 during the first 15 years of burnup and within 2,000 PCM through 90 years of fuel burnup and ended with a deficit of just under 4,600 PCM after 120 years of burnup, still well within the control range. During the first 90 years of burnup, the fuel fissile content remained above the 3.00% of the initial fuel, showing a true breed and burn fuel cycle with an average conversion ratio greater than 1.00000. During the next 30 years, the fissile content dropped to 2.66%, but still 88% as high as the initial fissile content. For comparison, in LWR the fissile content drops from 4.95% to around 1.43% or even lower in 6 years. The final burnup in terms of fissions per initial metal atom (%FIMA) is approximately 35.0258%, which is 11.67 times the initial fissile content compared to a burnup of around 5.5% in LWR. While the decrease in K_{eff} below 1.00000 will have some effect on the neutron flux distribution, for most of the reactor life this difference is small enough that is should have only a small effect on the simulation accuracy.

Many other simulations were done to examine the effect of some changes in the reactor, the operating conditions, or underlying assumptions. The cases and related discussion are shown below, the tables of detailed results are in Section 5 - Appendix A.

Table 3 Double power for half the time (inp9191m.inp) end K_{eff}=0.95741 vs. 0.95428 in the base case. In Table 3 the power is doubled and the time of each step is halved so each step represents the same burn as the previous table. Step by step the results differ from the previous table by very small amounts (except for time), showing that changing the reactor power has little effect on the performance of the reactor relative to the burnup state.

Table 2. The burnup results of the basic SMR configuration with SCALE (file inp9194p.inp)

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00847	2.9999	0	97.0001	2.9999	0
1	0.99924	3.0071	0.2919	96.6166	2.7231	1
2	0.99374	3.0157	0.8756	95.8520	2.2797	3
3	0.99296	3.0247	1.7512	94.7085	1.7784	6
4	0.99297	3.0386	2.9188	93.1955	1.3003	10
5	0.99155	3.0691	4.3781	91.3200	0.8943	15
6	0.98852	3.1263	8.7561	85.9165	0.3124	30
7	0.98825	3.1781	13.1343	80.7664	0.1173	45
8	0.98958	3.1882	17.5129	75.8486	0.0496	60
9	0.98848	3.1289	21.8913	71.1470	0.0267	75
10	0.98085	3.0205	26.2697	66.6104	0.0202	90
11	0.97144	2.8528	30.6479	62.2318	0.0195	105
12	0.95428	2.6634	35.0258	57.9648	0.0206	120

Table 4 Fuel density of 15 g/cc not 17 (inp9189m.inp), end K_{eff}=0.92801 vs. 0.95428. Scale does not simulate the change in the fuel density with burnup even though this becomes significant with high burnup such as is simulated in these tables. At the average fuel temperature of 1300 C. uranium has a density of around 17 g/cc so this value is specified for the burn simulations in most cases. In this case the density was specified as 15 g/cc. but the geometry was unchanged so the simulated fuel mass is lower and the simulated total power is lower because the power is specified as 8 MW/metric fuel ton. The results show a lower simulated K_{eff}, but the other columns show little change. Since the actual change in fuel density only occurs after there is significant burnup these results suggest that the change in fuel density with burnup does not cause a significant error in the burn results.

Table 5 More neutrons (inp9210k.inp) end K_{eff}=0.95481 vs. 0.95428. This simulation was run with 4 times as many simulated neutrons but gives essentially the same simulation results, suggesting that the results are not a fluke caused by too few neutrons in the simulations.

Table 6 Fuel wall thickness 2 cm instead of 1 cm and 1 cm less fuel radius (inp9198p.inp), end K_{eff}=0.93939 vs. 0.95428. For this simulation the fuel tube inner radius was decreased by one cm and the fuel tube wall thickness was increased by one cm so the outer diameter of the fuel tube and all other dimensions in the reactor were unchanged. This decreased the fuel mass and increased the fuel tube mass, increasing the ratio of the mass of other materials to fuel mass. As expected, this had a negative impact on the burn results. The conversion ratio was slightly decreased with the final fuel fissile content reduced from 2.7137 to 2.5117 (just under 7.5% change in relative value) and the ending K_{eff} was reduced to 0.93939, still within the range of the control method.

Table 7 One meter more space between the core and fuel return circuit (inp9187m.inp) end K_{eff}=0.95270 vs. 0.95428. The fuel in the MUBR circulates up through the fuel tubes, across through a transfer tube to the heat exchanger, down through the heat exchanger and the MHD fuel pump, and across to the bottom of the fuel tubes via another transfer tube. For this simulation the length of the horizontal transfer tubes was increased by one meter, increasing the separation between the fuel tubes in the reactor core and the heat exchanger and fuel pump. There may be some neutron transmission between the core and the heat exchanger/fuel pump region. By increasing the separation this simulation helps to gauge the importance of this neutron exchange. The results show very small changes in the conversion ratio (ending fissile concentration decreased from 2.7137% to 2.6928%, or less than 1%) and the ending K_{eff} reduced from 0.95428 to 0.95270. This suggests that the neutron transfer between the two vertical fuel flows is not a significant factor in the burn results, so the length of the tubing between the reactor core and the heat exchanger can vary greatly with minimal effects on the reactor neutronics.

Table 8 Fission product removal reference half life 15 days, not 5 days (inp9186n.inp) end K_{eff}=0.94514 vs. 0.95428. Table 8 shows the effect of reducing the simulated rate of fission product removal by a factor of three. The results show very small changes in the conversion ratio (ending fissile concentration increased from 2.6634% to 2.6840%,) but the ending K_{eff} reduced from 0.95428 to 0.94514. This suggests that even a significant error in the simulated rates of fission product removal has no practical effect on the simulated fuel life.

Table 9 Fission product removal reference half life 50 days, not 5 days (inp9188n.inp) end K_{eff}=0.93085 vs. 0.95428. Table 9 shows the effect of reducing the simulated rate of fission product removal by a factor of ten (from the base case). The results show bigger changes in the conversion ratio (ending fissile concentration increased from 2.6634% to 2.7222% but the ending K_{eff} reduced from 0.95428 to 0.93085, near the limit of the control method. The increase in fissile content may be because the higher concentration of fission products absorbs more thermal neutrons which decreases the rate of thermal fission and increases the ratio of fast fission to thermal fission. This suggests that large changes in the simulated rates of fission product removal have a significant practical effect on the simulated fuel life.

Table 10 No Fission product removal (inp9191n.inp) end K_{eff}=0.84796 vs. 0.95428. Table 10 shows the effect of reducing the simulated rate of fission product removal to zero. The results show the ending K_{eff} reduced from 0.95428 to 0.84796 and K_{eff} reduces to near the limit of the control method after only 30 years instead of 120 years. This suggests that no fission product removal greatly reduces the viability of the MUBR concept.

Table 11 No fission product removal using MCNP (inp5297d.txt) end K_{eff}=84888 vs. 94528. Table 11 shows the results of the same simulation as Table 10 but using MCNP instead of SCALE. Use of MCNP is possible in this case where there is no fission product removal simulated whereas all of the other cases had fission product removal and could not be done with MCNP. This simulation was done with fewer simulated neutrons to reduce the simulation run time. There are some differences in the values between Table 10 and Table 11 but these are small compared to the changes in the values between steps. This comparison between the MCNP results and the SCALE results supports the conclusion that both analysis methods agree on the long term results of the simulation, that without some removal of fission products the fuel will only last around 30 years.

Table 12 K_{eff} by step by table, a summary of K_{eff} results from Tables 2 to 11. Looking at the ending simulated K_{eff} shows that it is outside of the control range only in the MCNP and SCALE simulations with no fission product removal. There are also two cases where the ending simulated K_{eff} is close to the limit of the MUBR control range. One is Table 4 where the simulated fuel density was reduced from 17 g/cc to 15 g/cc which does not occur in the physical fuel until well into the burnup. This indicates that the actual effect of the change in fuel density does not seriously impact the validity of the MUBR simulations. The other is Table 9 where the simulated rate of fission product is reduced by a factor of 10 which shows that while some rate of fission product removal is required the simulation results show that the MUBR can operate even if the actual rate of fission product removal is much less than expected.

Table 13 % Fissile content by step by table, a summary of fissile content from Tables 2 to 11. Looking at the simulated ending fissile content shows some differences between the cases with a range from 2.512 to 3.138 or 0.626% which seems insignificant when compared to the simulated fuel burnup of 35%. If the case of the SCALE simulation with no fission product removal is ignored, then the range is 2.512 to 2.730 or 0.218, which is quite small. The simulation with no fission product removal can be ignored because no MUBR will be constructed with no fission product removal.

Table 14 Higher level of liquid moderator in the control cavities (inp9194m.inp) end K_{eff}=.95275 vs. 94528. This simulation was done with the level of liquid moderator in the control cavities set much higher than in the other simulations (48.5% liquid instead of 42.7% liquid in the base case) in order to evaluate the concern that errors in the simulated control method would change the simulated conversion ratio. The results show that there was only a small change in the ending simulated K_{eff} and ending simulated fissile content (from 0.95901 to 0.95734 and from 2.7137 to 2.6338). This suggests that an error in the simulate K_{eff} does not imply a change in the simulation results large enough to invalidate the results.

Table 15 burnup with fuel that is 50% UNF (inp9196p.inp) end K_{eff}=.94156 vs. 94528. This simulation is of the MUBR fueled by a mixture of 50% LWR UNF and 50% standard LWR LEU (4.95% U-235). The results show that there was only a small change in the ending simulated K_{eff} and ending simulated fissile content (from 0.95901 to 0.94670 and from 2.7137 to 2.3243). The ending K_{eff} is well within the control range which suggests that this fuel option is viable.

The actual tables for the additional simulation results discussed above are in Section 5: Appendix A, additional results. They are there for anyone interested in the details, not just this summary.

4. Conclusions and future work

The many simulations of the MUBR concept reactor suggest that the MUBR achieves a true breed and burn fuel cycle which eliminates the need to refuel the MUBR, produces over 6 times the power per ton of fuel compared to LWR and around 11 times as much power per ton of mined uranium compared to LWR because it uses uranium enriched to 3% instead of 4.95%. It produces 6 times as much power per ton of Used Nuclear Fuel (UNF) produced as LWR and no UNF is removed from the reactor for the life of the reactor compared to LWR which requires removal of many tons of UNF every two years.

The SCALE simulations of various variations of the MUBR configuration or the operating conditions show that the design is robust, the only condition which greatly reduces the fuel life is complete absence of fission product removal.

The MUBR is currently a reactor concept. The only design for it is in the SCALE and MCNP input files used to perform neutronic analysis. This is certainly not an engineering design and leaves out most of the engineering details, especially for the parts outside the reactor core where the materials do not have much effect on the neutronics. The same specifications also allow MUBR6gen to perform an elementary analysis of fluid flow in the fuel circuit. Engineering design cannot be developed at this time because there are too many unknowns. As various unknowns are resolved, the design details will stabilize, and neutronics analysis will need to be updated to analyze and optimize the design. The MUBR advantages described below suggest that the effort needed to create an engineering design and a prototype MUBR are well justified.

4.1 MUBR advantages

4.1.1 The fuel cycle is breed and burn

The primary advantage of the MUBR is that it is a true breed-and-burn mixed spectrum reactor, so most of its power comes from fission of U-238 either directly by fast fission or indirectly (by conversion to Pu-239). The plentiful U-238 in the fuel is converted to fissile Pu-239 in the same location where fission occurs; thus, the fuel lasts for longer than the reactor life, and the reactor is never shut down for refueling or fuel manipulation. The conversion ratio is very close to 1.000; therefore, the fuel reactivity changes very slowly with time, which allows reasonably accurate simulations over long periods of time. In addition, the MUBR uses fuel with a lower enrichment than the Light Water Reactor (LWR); therefore, there is no requirement for High-Assay Low-Enriched Uranium (HALEU) or other special fuels.

Because of the high temperature of molten uranium fuel, the MUBR uses high-temperature turbine generators, which are more efficient than other generators and produce more electricity from the same amount of heat energy. In addition, because high-temperature heat energy can be stored in cheap ordinary salt, it allows the MUBR complex to store energy when the price of electricity is low due to high production from solar or wind systems, and then sell at a higher price when the sun is not shining or the wind is not blowing. It also allows the MUBR in off-grid environments to change the reactor power output gradually while the generated power changes rapidly, so it can follow even sudden changes in demand.

4.1.2 Reactor operations benefits

The MUBR provides heat energy to a heat user complex through the flow of Heat Transfer Fluid (HTF). The heat user complex may include electricity generators, Thermal Energy Storage (TES), and/or various types of process heat users. Energy storage allows a fast response to changes in net energy demand so the MUBR power can be slowly and stably changed in response to changes in the energy demand and available capacity of the TES. This allows MUBR control to be almost entirely automatic. The MUBR power is proportional to the flow of the moderator coolant; therefore, the pumps for the moderator coolant, fuel, and fuel coolant can all be driven by the same variable power supply. This means that the operating temperatures of the fuel and fuel coolant maintain nearly the same temperature profile, independent of the power level, so there is little thermal shock even with frequent load following. The fuel life is much longer than the reactor life; therefore, the reactor is never shut down for refueling or fuel manipulation. Because there is no refueling, there is no need for a facility to store and safeguard a new batch of UNF on site every so often. There are few expenses for MUBR which increase with the rate of heat delivered; therefore, the marginal cost of heat production is near zero.

4.1.3 Energy security

With MUBRs, the fuel supply problem is almost completely eliminated. Once a MUBR is installed, it does not require any additional fuel for the life of the reactor. It does not need to be shut down periodically for refueling and requires little maintenance of the nuclear portion of the power system. Concerns about the cost, availability, and transportation of fuels are eliminated. Because there is no need for a fuel resupply, there is no price shock, and the cost of electricity generation is mostly the payoff of the initial capital cost of the system and its initial fuel. Similar to all electrical systems, an MUBR can be damaged or destroyed by a military or terrorist attack; however, this is not the usual energy security issue. With fossil fuel generation, weather events or military or terrorist attacks on power stations are possible, but weather events or attacks on the highly dispersed and hard-to-protect fuel transportation network can also shut down power generation for fossil fuel generator types.

4.1.4 Energy economics benefits

It is very difficult to assess the cost of the MUBR at this point because the cost depends on the engineering design, but much of the work required to develop an engineering design has not yet been done. A recent technical report by the Idaho National Laboratory (INL)[11] is an extensive study of the expected costs of new reactors built in the next few decades. They looked at the expected costs of early examples (~2030) and then the Nth of a kind (around 2050). Using moderate assumptions, they suggest that the original capital cost for a 300 MegaWatt electrical Small Modular Reactor (SMR) (including the initial fuel) will decrease from $8,000 per KiloWatt electrical (KWe) in 2030 to $4,000 per KWe in 2050 (expressed in 2022 $). They suggest that these results are true independent of the technology (LWR, High Temperature Gas Reactor, Sodium Fast Reactor, MSR). There is nothing in their analysis to suggest any different result for the MUBR. After the reactor is operating, they suggest that the operating and maintenance cost per MegaWatt hour (MWh) will be around $11 for fuel, $15 for fixed costs, and $2.60 for variable costs for a total of $28.60 per MWh. For the MUBR there is no fuel cost because the original fuel will last for longer than the life of the reactor. For the MUBR the operating and maintenance cost per MWh will be around $15 for fixed costs and $2.60 for variable costs for a total of $17.60 per MWh, a reduction of over 38% compared to other SMR designs. This gives the MUBR a significant operating cost advantage over the other proposed SMR designs.

4.1.5 uranium supply benefits

In the basic MUBR fuel cycle, approximately 5.4 tons of uranium is mined and enriched to provide one ton of LEU (3% U-235). The fuel is placed in an MUBR, where it remains for the life of the reactor (at least 60 years). During that time around 175 kg per ton of the fuel is split to provide around 4,200 GWh thermal of power, but the fuel is not "used." The reactor may have reached the end of its life, but the fuel can be transferred to a new MUBR and used again until the burnup is around 350 kg per fuel ton. Of the original six tons of uranium mined per fuel ton, around 6.5% is used (split) compared to around 0.5 to 0.6% for light water reactors. The mined uranium required per MWh thermal is reduced by a factor of over 10.

4.1.6 UNF mitigation

Several important factors are involved in the mitigation of UNF by MUBRs. MUBR fuel produces over 6 times as much power per ton of fuel as LWR, so there is only one sixth of the amount of UNF produced per MWh. The MUBR is never refueled, so no UNF is removed from the reactor for on-site or off-site storage during the 60 to 120 year life of the fuel. The eventual MUBR UNF has fissile plutonium content (Pu-239 and Pu-241), which is approximately seven times greater than that of the LWR UNF, so it is much more valuable as a resource for making new fuel. Many of the fission products are separated from the fuel during MUBR operation, so the extraction of valuable elements and isotopes from the eventual UNF is greatly simplified. These factors significantly reduce the present and future liability aspects of MUBR UNF and increase the eventual value of the much smaller amount of MUBR UNF produced.

In addition, the MUBR can be built with its initial fuel being a mix of standard LEU (4.95% U-235) and LWR UNF which has had the fuel pellets removed from the fuel rods and the metal oxides reduced to metal (the simulation for this in the results section used 50% LEU (4.95% U-235) and 50% LWR UNF). This takes some of the existing or newly produced LWR UNF out of storage and sequesters it in an MUBR reactor for 60 to 120 years while using it to produce much larger amounts of energy than originally produced and converts it into more valuable MUBR UNF. Because MUBR reactors use so much less fuel per MWH they cannot dispose of most of the existing LWR UNF but may be able to use most of the future UNF production.

4.1.7 Grid reliability and stability with intermittent power sources such as wind and solar

A complete MUBR system has two major islands: a nuclear island and a generation island. The nuclear islands are highly regulated and relatively compact. The generation island is connected to the nuclear island by pipes that carry HTF in a loop between the two islands. The generation island has turbines and generators but may also have TES and/or the ability to sell thermal energy directly as industrial process heat or for other heat needs. It may also have the ability to power turbines with liquid fuel to supplement the output at times of high price. The distance between the islands may be large enough to satisfy the requirements of nuclear regulators to keep the generation complex far enough away so it is not nuclear regulated, or the regions may both be in a single box in the case of a microreactor. Because of the high temperature of the MUBR fuel, the HTF is also very hot, and the thermal energy storage can be performed in cheap ordinary salt (NaCl) or other materials.

An electrical power grid may have many sources of power such as natural gas, coal, nuclear, hydroelectric, wind, and solar; many users such as towns or cities, industrial sites, data centers, etc.; and may have energy storage with pumped hydroelectric or batteries. The grid should have sufficient generation and storage capacities to meet the maximum demand. Because the average demand is less than the maximum demand, this implies that the grid must have excess generation and storage capacity, some of which is used much less than 100% of the time. In the MUBR system with TES, the system may have excess generation capacity with relatively cheap turbines and generators while having a lower capacity for high-cost nuclear power.

A reasonable amount of TES cannot store enough energy to handle all catastrophic situations, but it is sufficient to handle somewhat more than the average daily net demand fluctuations. The MUBR marginal cost of energy production is almost zero; therefore, any operation of the nuclear plant at less than full power is a loss of income. The ability to cheaply store energy when there is a large amount of solar or wind power and sell it when there is little solar or wind power is a significant income opportunity and contributes to grid reliability and stability.

4.1.8 Nuclear nonproliferation advantages

MUBR operates by converting U-238 to Pu-239 and splitting most of its production. LWRs also converted U-238 to Pu-239. The difference is that, with LWRs, some of the used fuel is removed from the operating reactor every two years and stored somewhere (usually on site) where it is much more accessible. In the MUBR, the fuel is kept inside the reactor in a circulating molten form, which is very difficult to access. The concentration of Pu-239 in an MUBR is higher (approximately 3%) than in LWRs, but it is burned as fast as it is created when the initial U-235 has been replaced by Pu-239. When the MUBR fuel reaches its end of life, the amount of plutonium is vastly less per MWh of power produced than for the LWR. The MUBR UNF plutonium will be around one fissile atom (Pu-229 and Pu-241) for each non fissile atom (Pu-238, Pu-240, and Pu-242) compared to the LWR UNF with around two fissile plutonium atoms for each non fissile plutonium atom; therefore, the MUBR UNF will be much less desirable for weapons use.

4.1.9 Climate change
Climate change and environmental pollution are generally recognized as the result of high fossil fuel use for many purposes, including electricity generation, transportation, heating, and industrial process heat production. National and international efforts to reduce fossil fuel consumption are often in the form of incentives or legislative mandates to either increase the use of “green” energy sources such as solar, wind, nuclear, etc., or to increase the efficiency of fossil fuel users by increasing vehicle miles per gallon or insulation of buildings, etc. These efforts are accelerating the use of green energy and may lead to the development of the MUBR concept.

If the MUBR goes into production and reaches a production rate of several MUBRs per year, then the potential cost benefits of mass production might reduce the Levelized Cost OF Electricity (LCOE) to below that of natural gas. At that point, the decision to order an MUBR will be mostly based on simple economics, not incentives or mandates, and the MUBR demand could greatly increase, which could further reduce costs. Eventually, the decrease in demand for fossil fuels could decrease their asking price, which would affect the competitive advantage of the MUBR. However, this would probably not occur until there is a significant decrease in fossil fuel consumption, which is the desired goal.

4.2 MUBR future development
New information is needed on two aspects of MUBR design before investors and others will invest significant money and effort in the MUBR development process. The two initial questions are as follows:

- The proposed material for the fuel tubes is silicon carbide. Experimental work is required to determine whether molten uranium will dissolve or degrade silicon carbide at temperatures from 1150 °C to 1550 ° C. Additional work is required to determine how the high neutron flux changes the results and how fast any degradation occurs.

- As the MUBR operates, neutron interactions continuously change some of the fuel (initially mostly uranium) into fission products and actinides. The MUBR depends on removing some fission products from the circulating molten uranium fuel, mostly by the evaporation of some of the fission product elements that have boiling points below or near the maximum fuel temperature. Experimentation is required to determine the actual rates at which fission products and actinides are separated from fuel by evaporation. Some fission products and actinides may be insoluble in molten uranium and float on top of dense molten uranium metal fuel as dross. Experiments are required to determine the rate at which this occurs.

If it appears that silicon carbide is not a suitable material for the MUBR fuel tubes, other materials can be tried. Once there is favorable information on the issues of fuel tube material and fission product removal rates, development of the MUBR can proceed much the same as any other advanced reactor concept. The development of the MUBR product can be completed by an existing or new nuclear engineering corporation and the usual product development steps undertaken. These include funding, public relations, politics, regulation and licensing, engineering, marketing, financial analysis and product development.

Much of the engineering work and analysis will focus on the changing properties of the fuel and the two groups of separated fission products: condensed evaporated fission products (and actinides) (which may be separated into subgroups based on their condensation temperature) and floating dross. The properties of these three material groups change as a function of time, burnup, and temperature. The properties of interest include mass, composition, melting point, viscosity, density, specific heat, thermal conductivity, radioactivity, and decay heat production. In addition, as each group includes different elements, stable chemical compounds may form, which may affect the material properties.

More traditional engineering analyses will include thermal hydraulic analysis of all fluid circuits under conditions of constant power, slowly changing power for load following, and rapid changes due to power failure or other abnormal conditions; and analysis of stress, temperature, thermal expansion, and heat flow in all solid components under the same power conditions as for the thermal hydraulic analysis. Safety analysis must be performed for all components and systems, and these days must include direct attack by terrorist groups and foreign military forces, as well as the effects of nuclear weapons explosions at different powers and distances from the reactor. Procedures need to be developed for the eventual decommissioning of MUBRs at the end of the reactor life and the recovery of valuable fuel and other fluids. Large fuel tubes are the MUBR components that are most likely to fail. The tubes are vertically oriented and are surrounded by an inert pressure gas; therefore, the pressure outside the tubes is higher than the fuel pressure inside the tubes. This reduces the stress on the tubes and slows the propagation of small cracks in the tubes, which means that small leaks allow the pressure gas to flow into the fuel circuit rather than the fuel flowing out of the tubes. Experimentation and analysis of the fuel tube failure modes and rates are needed, so instrumentation can be included in the MUBR which can detect degradation of the tubes before there is a catastrophic tube failure. Studies on the effects of the power level on the fuel tube life are also needed.

Financial analysis will include the cost and availability of all proposed materials, cost of fabrication and construction, expected demand, competition, and cost of money.

Above all, the MUBR analysis and design process are iterative. Results from marketing, financial, engineering, neutronics, and safety analyses will lead to suggested design changes and require repetition of all analysis steps. At some point a prototype reactor design will be built and operated for some period of time. Lessons learned from the construction and operation of the prototype will be applied to the proposed design for the first commercial reactor.

5. Appendix A: Supplemental results for section 3

Table 3. Double power for half the time with the same burn per step (inp9191m.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00847	2.9999	0	97.0001	2.9999	0
1	0.99737	3.0047	0.2919	96.6143	2.7228	0.5
2	0.99109	3.0158	0.8757	95.8473	2.2796	1.5
3	0.98902	3.0271	1.7513	94.7039	1.7791	3
4	0.98978	3.0466	2.9188	93.1886	1.3021	5
5	0.98999	3.0830	4.3782	91.3154	0.8972	7.5
6	0.98788	3.1580	8.7562	85.9258	0.3151	15
7	0.98991	3.2232	13.1345	80.8012	0.1179	22.5
8	0.99315	3.2315	17.5131	75.9183	0.0471	30
9	0.99240	3.1678	21.8917	71.2469	0.0213	37.5
10	0.98499	3.0481	26.2701	66.7405	0.0124	45
11	0.97332	2.8840	30.6482	62.3736	0.0096	52.5
12	0.95741	2.6810	35.0620	58.1252	0.0090	60

In Table 3 the power is doubled and the time of each step is halved so each step represents the same burn as the previous table. Step by step the results differ from the previous table by very small amounts (except for time), showing that changing the reactor power has little effect on the performance of the reactor relative to the burnup state.

Table 4. Fuel density of 15 g/cc not 17 (SCALE doesn't do density change) (inp9189m.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00818	3.0000	0	96.9999	3.0000	0
1	0.99848	2.9967	0.2919	96.6233	2.7211	1
2	0.99163	2.9864	0.8756	95.8779	2.2731	3
3	0.98808	2.9664	1.7511	94.7611	1.7656	6
4	0.98371	2.9524	2.9184	93.2729	1.2815	10
5	0.98126	2.9504	4.3775	91.4292	0.8714	15
6	0.97158	2.9568	8.7545	86.0770	0.2919	30
7	0.96569	2.9952	13.1319	80.9514	0.1053	45
8	0.96603	2.9981	17.5092	76.0522	0.0436	60
9	0.96321	2.9465	21.8867	71.3559	0.0242	75

Table 4. *Continued*

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
10	0.95551	2.8384	26.2638	66.8361	0.0193	90
11	0.94390	2.6894	30.6409	62.4532	0.0194	105
12	0.92801	2.5084	35.0176	58.1889	0.0207	120

Scale does not simulate the change in the fuel density with burnup even though this becomes significant with high burnup such as is simulated in these tables. At the average fuel temperature of 1300 C. uranium has a density of around 17 g/cc so this value is specified for the burn simulations in most cases. In this case the density was specified as 15 g/cc. but the geometry was unchanged so the simulated fuel mass is lower and the simulated total power is lower because the power is specified as 8 MW/metric fuel ton. The results show a lower simulated K_{eff}, but the other columns show little change. Since the actual change in fuel density only occurs after there is significant burnup these results suggest that the change in fuel density with burnup does not cause a significant error in the burn results.

Table 5. More neutrons (inp9210k.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-233%	Years
0	1.00823	2.9999	0	97.0001	2.9999	0
1	0.99786	3.0079	0.2919	96.6143	2.7233	1
2	0.99333	3.0182	0.8757	95.8497	2.2799	3
3	0.99267	3.0270	1.7513	94.7062	1.7790	6
4	0.99222	3.0421	2.9188	93.1909	1.3011	10
5	0.99207	3.0705	4.3782	91.3177	0.8951	15
6	0.98721	3.1304	8.7561	85.9119	0.3128	30
7	0.98754	3.1836	13.1344	80.7617	0.1175	45
8	0.98994	3.1924	17.5128	75.8440	0.0497	60
9	0.98878	3.1344	21.8913	71.1400	0.0268	75
10	0.98177	3.0179	26.2697	66.6104	0.0202	90
11	0.96957	2.8597	30.6479	62.2225	0.0196	105
12	0.95481	2.6662	350257	57.9579	0.0206	120

This simulation was run with 4 times as many simulated neutrons but gives essentially the same simulation results, suggesting that the results are not a fluke caused by too few neutrons in the simulations.

Table 6. Fuel wall thickness 2 cm instead of 1 cm and 1 cm less fuel radius (inp9198p.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00828	2.9999	0	97.0001	2.9999	0
1	0.99893	3.0007	0.2916	96.6236	2.7254	1
2	0.99229	2.9944	0.8748	95.8729	2.2869	3
3	0.99097	2.9787	1.7496	94.7550	1.7897	6
4	0.98841	2.9661	2.9160	93.2653	1.3131	10
5	0.98736	2.9632	4.3740	91.4223	0.9059	15
6	0.98049	2.9574	8.7474	86.0792	0.3160	30
7	0.97461	2.9832	13.1212	80.9593	0.1177	45
8	0.97669	2.9731	17.4951	76.0671	0.0491	60
9	0.97419	2.9084	21.8692	71.3794	0.0260	75

Table 6. *Continued*

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
10	0.96668	2.7945	26.2430	66.8614	0.0196	90
11	0.95455	2.6415	30.6166	62.4805	0.0189	105
12	0.93939	2.4575	34.9896	58.2205	0.0200	120

For this simulation the fuel tube inner radius was decreased by one cm and the fuel tube wall thickness was increased by one cm so the outer diameter of the fuel tube and all other dimensions in the reactor were unchanged. This decreased the fuel mass and increased the fuel tube mass, increasing the ratio of the mass of other materials to fuel mass. As expected, this had a negative impact on the burn results. The conversion ratio was slightly decreased with the final fuel fissile content reduced from 2.7137 to 2.5117 (just under 7.5% change in relative value) and the ending K_{eff} was reduced from 0.95901 to 0.94547, still within the range of the control method.

Table 7. One meter more space between the core and fuel return circuit (inp9187m.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00859	2.9999	0	97.0001	2.9999	0
1	0.99881	3.0068	0.2919	96.6166	2.7231	1
2	0.99466	3.0135	0.8757	95.8453	2.2789	3
3	0.99393	3.0185	1.7513	94.7132	1.7772	6
4	0.99324	3.0288	2.9188	93.2048	1.2982	10
5	0.99251	3.0520	4.3781	91.3363	0.8918	15
6	0.98729	3.1033	8.7559	85.9398	0.3095	30
7	0.98679	3.1539	13.1340	80.7919	0.1156	45
8	0.98801	3.1597	17.5125	75.8812	0.0486	60
9	0.98727	3.1018	21.8908	71.1795	0.0264	75
10	0.97954	2.9872	26.2690	66.6499	0.0201	90
11	0.96869	2.8259	30.6469	62.2690	0.0196	105
12	0.95270	2.6328	35.0246	58.0067	0.0205	120

The fuel in the MUBR circulates up through the fuel tubes, across through a transfer tube to the heat exchanger, down through the heat exchanger and fuel pump, and across to the bottom of the fuel tubes via another transfer tube. For this simulation the length of the transfer tubes was increased by one meter, increasing the separation between the fuel tubes in the reactor core and the heat exchanger and fuel pump. There may be some neutron transmission between the core and the heat exchanger/fuel pump region. By increasing the separation this simulation helps to gauge the importance of this neutron exchange. The results show very small changes in the conversion ratio (ending fissile concentration decreased from 2.7137% to 2.6928%, or less than 1%) and the ending K_{eff} reduced from 0.95901 to 0.95818. This suggests that the neutron transfer between the two vertical fuel flows is not a significant factor in the burn results.

Table 8. Fission product removal reference half life 15 days, not 5 days (inp9186n.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00847	2.9999	0	97.0001	2.9999	0
1	0.99694	3.0081	0.2919	96.6143	2.7233	1
2	0.99041	3.0201	0.8757	95.8473	2.2804	3
3	0.98862	3.0312	1.7513	94.7016	1.7807	6
4	0.98746	3.0495	2.9188	93.1863	1.3040	10
5	0.98619	3.0829	4.3782	91.3084	0.8991	15
6	0.98124	3.1484	8.7562	85.9003	0.3168	30

Table 8. *Continued*

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
7	0.98103	3.2002	13.1345	80.7548	0.1201	45
8	0.98280	3.2080	17.5130	75.8416	0.0510	60
9	0.98093	3.1524	21.8916	71.1377	0.0272	75
10	0.97350	30344	26.2701	66.6150	0.0202	90
11	0.96043	2.8758	30.6481	62.2318	0.0194	105
12	0.94514	2.6840	35.0259	57.9695	0.0204	120

Table 8 shows the effect of reducing the simulated rate of fission product removal by a factor of three. The results show very small changes in the conversion ratio (ending fissile concentration increased from 2.7137% to 2.7288%,) but the ending K_{eff} reduced from 0.95901 to 0.95109. This suggests that some error in the simulated rates of fission product removal have no practical effect on the simulated fuel life.

Table 9. Fission product removal reference half life 50 days, not 5 days (inp9188n.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00847	2.9999	0	97.0001	2.9999	0
1	0.99621	3.0082	0.2919	96.6143	2.7233	1
2	0.99003	3.0200	0.8757	95.8473	2.2807	3
3	0.98479	3.0338	1.7514	94.7016	1.7813	6
4	0.98225	3.0546	2.9189	93.1839	1.3055	10
5	0.97978	3.0946	4.3783	91.3014	0.9017	15
6	0.97252	3.1768	8.7563	85.8840	0.3212	30
7	0.97094	3.2377	13.1347	80.7315	0.1232	45
8	0.97217	3.2471	17.5132	75.8207	0.0526	60
9	0.96837	3.1905	21.8917	71.1214	0.0277	75
10	0.96038	3.0715	26.2703	66.6011	0.0202	90
11	0.94633	2.9169	30.6486	62.2202	0.0190	105
12	0.93085	2.7222	35.0264	57.9672	0.0199	120

Table 9 shows the effect of reducing the simulated rate of fission product removal by a factor of ten (from the base case). The results show bigger changes in the conversion ratio (ending fissile concentration increased from 2.7137% to 2.7605% but the ending K_{eff} reduced from 0.95901 to 0.93526, near the limit of the control method. The increase in fissile content may be because the higher concentration of fission products absorbs more thermal neutrons which decreases the rate of thermal fission and increases the ratio of fast fission to thermal fission. This suggests that large changes in the simulated rates of fission product removal have significant practical effect on the simulated fuel life.

Table 10. No Fission product removal (inp9191n.inp).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00803	2.9999	0	97.0001	2.9999	0
1	0.98206	3.0132	0.2908	96.6096	2.7247	1
2	0.97511	3.0344	0.8722	95.8334	2.2848	3
3	0.96824	3.0638	1.7443	94.6737	1.7899	6
4	0.96206	3.1072	2.9071	93.1374	1.3193	10
5	0.95712	3.1676	4.3605	91.2410	0.9201	15

Table 10. *Continued*

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
6	0.94158	3.3043	8.7196	85.7910	0.3422	30
7	0.93020	3.4170	13.0796	80.6107	0.1390	45
8	0.92173	3.4737	17.4396	75.6790	0.0623	60
9	0.91075	3.4590	21.7994	70.9634	0.0326	75
10	0.89448	3.3792	26.1586	66.4361	0.0218	90
11	0.87341	3.2543	30.5174	62.0575	0.0188	105
12	0.84796	3.0999	34.8752	57.7998	0.0190	120

Table 10 shows the effect of reducing the simulated rate of fission product removal to zero. The results show the ending K_{eff} reduced from 0.95901 to 0.85460 and K_{eff} reduces to near the limit of the control method after only 30 years instead of 120 years. This suggests that no fission product removal greatly reduces the viability of the MUBR concept.

Table 11. No fission product removal using MCNP (inp5297d.txt).

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00462	3.0000	0	97.0000	3.0000	0
1	0.97830	3.0060	0.2902	96.3400	2.7160	1
2	0.97895	3.0145	0.8707	95.0800	2.2630	3
3	0.97180	3.0199	1.7410	93.2600	1.7520	6
4	0.96788	3.0406	2.9020	90.9200	1.2730	10
5	0.96501	3.0680	4.3540	88.1100	0.8708	15
6	0.96350	3.1557	8.7070	81.1400	0.3008	30
7	0.95845	3.1900	13.0600	74.5600	0.1148	45
8	0.94427	3.1236	17.4100	68.4300	0.0473	60
9	0.92658	3.0114	21.7700	62.6900	0.0235	75
10	0.90554	2.8519	26.1200	57.3500	0.0160	90
11	0.87798	2.6534	30.4800	52.3500	0.0140	105
12	0.84888	2.4468	34.8300	47.4600	0.0143	120

Table 11 shows the results of the same simulation as Table 10 but using MCNP instead of SCALE. Use of MCNP is possible in this case where there is no fission product removal simulated whereas all of the other cases had fission product removal and could not be done with MCNP. This simulation was done with fewer simulated neutrons to reduce the simulation run time. There are some differences in the values between Table 10 and Table 11 but these are small compared to the changes in the values between steps. This comparison between the MCNP results and the SCALE results supports the conclusion that both analysis methods agree on the long term results of the simulation, that without some removal of fission products the fuel will only last around 30 years.

Another way of looking at the data is to view the changes in the most significant variables over time in each of the different simulations.

Table 12. K_{eff} by step by table.

Step/ case	Tbl 2 base	Tbl 3 2xPwr	Tbl 4 - dens	Tbl 5 + neut	Tbl 6 + wall	Tbl 7 + gap	Tbl 8 - FPR	Tbl 9 - FPR	T 10 0	T 11
0	1.0085	1.0085	1.0082	1.0082	1.0083	1.0086	1.0085	1.008	1.008	1.005
1	.99924	.99737	.99848	.99768	.99893	.99881	.99694	.9962	.9821	.9783

Table 12. *Continued*

Step/ case	Tbl 2 base	Tbl 3 2xPwr	Tbl 4 - dens	Tbl 5 + neut	Tbl 6 + wall	Tbl 7 + gap	Tbl 8 - FPR	Tbl 9 - FPR	T 10 0	T 11
2	.99374	.99109	.99163	.99333	.99229	.99466	.99041	.9900	.9751	.9790
3	.99296	.98902	.98808	.99267	.99097	.99393	.98862	.9848	.9682	.9718
4	.99297	.98978	.98371	.99222	.98841	.99324	.98746	.9823	.9620	.9679
5	.99155	.99999	.98126	.99207	.98736	.99251	.98619	.9798	.9571	.9650
6	.98852	.98788	.97158	.98721	.98049	.98729	.98124	.9725	.9416	.9635
7	.98825	.98991	.96569	.98754	.97461	.98679	.98103	.9709	.9302	.9584
8	.98958	.99315	.96603	.98994	.97669	.98801	.98280	.9722	.9217	.9443
9	.98848	.99240	.96321	.98878	.97419	.98727	.98093	.9684	.9108	.9266
10	.98085	.98499	.95551	.98177	.96668	.97954	.97350	.9604	.8945	.9055
11	.97144	.97332	.94390	.96957	.95455	.96869	.96043	.9463	.8734	.8780
12	.95428	.95741	.92801	.95481	.93939	.95270	.94514	.9309	.8480	.8489

Looking at the ending simulated K_{eff} shows that it is outside of the control range only in the MCNP and SCALE simulations with no fission product removal. There are also two cases where the ending simulated K_{eff} is close to the limit of the MUBR control range. One is Table 4 where the simulated fuel density was reduced from 17 g/cc to 15 g/cc which does not occur in the physical fuel until well into the burnup. This indicates that the actual effect of the change in fuel density does not seriously impact the validity of the MUBR simulations. The other is Table 9 where the simulated rate of fission product is reduced by a factor of 10 which shows that while some rate of fission product removal is required the simulation results show that the MUBR can operate even if the actual rate of fission product removal is much less than expected.

Table 13. % Fissile content by step by table.

Step/ case	Tbl 2 base	Tbl 3 2xPwr	Tbl 4 - dens	Tbl 5 + neut	Tbl 6 + wall	Tbl 7 + gap	Tbl 8 - FPR	Tbl 9 - FPR	T 10 0 FPR	T 11 MCNP
0	3.000	3.000	3.000	3.000	3.000	3.000	3.000	3.000	3.000	3.000
1	3.007	3.005	2.997	3.008	3.001	3.007	3.008	3.008	3.013	3.006
2	3.016	3.016	2.986	3.018	2.994	3.014	3.020	3.020	3.034	3.015
3	3.025	3.027	2.966	3.027	2.979	3.019	3.031	3.034	3.064	3.020
4	3.039	3.047	2.952	3.042	2.966	3.029	3.050	3.055	3.107	3.041
5	3.069	3.083	2.950	3.071	2.963	3.052	3.083	3.095	3.168	3.068
6	3.126	3.158	2.957	3.130	2.957	3.103	3.148	3.177	3.304	3.156
7	3.178	3.223	2.995	3.184	2.983	3.154	3.200	3.238	3.417	3.190
8	3.188	3.232	2.998	3.192	2.973	3.160	3.208	3.247	3.474	3.124
9	3.129	3.168	2.946	3.134	2.908	3.102	3.152	3.191	3.459	3.011
10	3.021	3.048	2.838	3.018	2.794	2.987	3.034	3.072	3.372	2.852
11	2.853	2.884	2.689	2.860	2.642	2.826	2.876	2.917	3.254	2.653
12	2.663	2.681	2.508	2.666	2.458	2.633	2.684	2.722	3.100	2.447

Looking at the simulated ending fissile content shows some differences between the cases with a range from 2.512 to 3.138 or 0.626% which seems insignificant when compared to the simulated fuel burnup of 35%. If the case of the SCALE simulation with no fission product removal is ignored, then the range is 2.512 to 2.730 or 0.218, which is quite small. The simulation with no fission product removal can be ignored because no MUBR will be constructed with no fission product removal.

Table 14. **Higher level of liquid moderator in the control cavities (inp9194m.inp).**

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.01860	2.9999	0	97.0001	2.9999	0
1	1.00862	3.0015	0.2908	96.6212	2.7221	1
2	1.00316	2.9999	0.8723	95.8659	2.2766	3
3	1.00068	2.9936	1.7445	94.7387	1.7723	6
4	0.99883	2.9907	2.9074	93.2420	1.2911	10
5	0.99624	3.0019	4.3609	91.3851	0.8830	15
6	0.98911	3.0345	8.7208	86.0072	0.3025	30
7	0.98747	3.0761	13.0812	80.8733	0.1115	45
8	0.98894	3.0794	17.4416	75.9695	0.0466	60
9	0.98579	3.0304	21.8019	71.2632	0.0253	75
10	0.97958	2.9200	26.1617	66.7359	0.0195	90
11	0.96790	2.7640	30.5210	62.3527	0.0192	105
12	0.95275	2.5748	34.8793	58.0903	0.0203	120

This simulation was done with the level of liquid moderator in the control cavities set much higher than in the other simulations (48.5% liquid instead of 42.7% liquid in the base case) in order to evaluate the concern that errors in the simulated control method would change the simulated conversion ratio. The results show that there was only a small change in the ending simulated K_{eff} and ending simulated fissile content (from 0.95901 to 0.95734 and from 2.7137 to 2.6338). This suggests that an error in the simulate K_{eff} does not imply a change in the simulation results large enough to invalidate the results.

Table 15. **burnup with fuel that is 50% UNF (inp9196p.inp).**

Step	K_{eff}	Fissile %	Burn %	U-238%	U-235%	Years
0	1.00794	3.0808	0	93.9431	2.7276	0
1	1.02216	3.0954	0.2919	93.5963	2.5083	1
2	1.02352	3.0588	0.8757	92.9073	2.1347	3
3	1.01997	3.0054	1.7513	91.8691	1.6905	6
4	1.01671	2.9454	2.9187	90.4887	1.2494	10
5	1.01079	2.8933	4.3780	88.7662	0.8634	15
6	0.99605	2.8136	8.7554	83.7040	0.2972	30
7	0.98776	2.7972	13.1330	78.8188	0.1088	45
8	0.98246	2.7800	17.5105	74.1195	0.0457	60
9	0.98013	2.7066	21.8880	69.6110	0.0255	75
10	0.96912	2.6028	26.2654	65.2448	0.0202	90
11	0.95727	2.4585	30.6423	61.0095	0.0199	105
12	0.94158	2.2861	35.0190	56.8822	0.0208	120

This simulation is of the MUBR fueled by a mixture of 50% LWR UNF and 50% standard LWR LEU (4.95% U-235). The results show that there was only a small change in the ending simulated K_{eff} and ending simulated fissile content (from 0.95901 to 0.94670 and from 2.7137 to 2.3243). The ending K_{eff} is well within the control range which suggests that this fuel option is viable.

Ethical approval and consent

Ethical approval and consent were not required.

Data availability

Underlying data

The MCNP and SCALE input files used to produce the above results are available and can be run by anyone with recent versions of MCNP and SCALE. The four input files used to show the control range in Table 1 are inp9282a.txt and inp9284a.txt for MCNP and inp9283a.inp and inp9285a.inp for SCALE. The SCALE input file used to run the burnup in Table 2 was inp9194p.inp, Tbl 3 was inp9191m.inp, T4 was inp9189m.inp, T5 was inp9210k.inp, T6 was inp9198p.inp, T7 was inp9187m.inp, T8 was inp9186n.inp, T9 was inp9188n.inp, T10 was inp9191n.inp, T11 was inp5297d.txt, T14 was inp9194m.inp and T15 was inp9196p.inp, Readers interested in these code inputs can request them by contacting the corresponding author of this paper. The files can be provided to personnel at national labs, universities, or other institutions which are not developing competing nuclear reactor design and who sign a non-disclosure agreement.

Software availability

The software "MUBR6gen" described in 2 is proprietary and is not needed to reproduce the results. It generates the MCNP or SCALE input files for MUBR models and executes MCNP or SCALE to perform the computational simulations which generate the results used in the document. The input files contain some proprietary details about the MUBR design.

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None.

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Open Peer Review

Current Peer Review Status: ? ?

Version 1

Reviewer Report 20 November 2024

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Güven Tunç
[1] Gazi University, Ankara, Turkey
[2] Gazi University, Ankara, Turkey

The manuscript introduces an innovative and potentially transformative reactor concept. The detailed analysis using SCALE and MCNP simulations provides a strong foundation for theoretical understanding. Below, I outline specific comments and recommendations for revision.

1. Authors should enhance the clarity of Figures 1–6 by ensuring consistent labeling and providing detailed captions.

2. Authors should include additional figures or diagrams illustrating the proposed fission product removal mechanisms.

3. The authors should add a graph showing the change in the reactor conversion rate over time.

4. Authors should produce graphs or tables showing the rate of change in uranium, plutonium, and minor actinide isotopes in the reactor over time.

5. Authors should explicitly state assumptions made in your SCALE and MCNP simulations.

Is the work clearly and accurately presented and does it cite the current literature?
Yes

Is the study design appropriate and does the work have academic merit?
Yes

Are sufficient details of methods and analysis provided to allow replication by others?
Yes

If applicable, is the statistical analysis and its interpretation appropriate?

Partly

Are all the source data underlying the results available to ensure full reproducibility?
Yes

Are the conclusions drawn adequately supported by the results?
Yes

Competing Interests: No competing interests were disclosed.

Reviewer Expertise: Nuclear and thermal energy engineering

I confirm that I have read this submission and believe that I have an appropriate level of expertise to confirm that it is of an acceptable scientific standard, however I have significant reservations, as outlined above.

Reviewer Report 08 October 2024

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Andang Widi Harto
[1] Universitas Gadjah Mada, Yogyakarta, Indonesia
[2] Universitas Gadjah Mada, Yogyakarta, Indonesia

This paper describe new nuclear reactor concept, i.e. MUBR which utilize molten low enriched uranium metal fuel and heavy water moderator. This design has capability to breed Pu-239 and directly burn this isotope. This concept is very interesting to develop simple sustainable nuclear reactor. However, there are several comments to improve this paper:
1. The number of references in this paper is small, only 11 references.
2. The authors may have performed several studies and calculations related to MUBR before this paper is written. The results from these studies and calculations can be used as additional references. The explanations in sub sections in section 2 (i.e. sub section 2.1, sub section 2,2, ...until sub section 2.14) need more specific reference citations.
3. It is better the authors add the flowchart diagram of the MUBR6gen software
4. The authors uses SCALE and MCNP. Why two software must be used? It is better if the authors state the cycle number and the number of sample particles utilized in the criticality calculation. What is the average standard deviations of these criticality calculations? Are these standard deviation values in acceptable range? Are the calculation results obtained from SCALE and MCNP have significant differences?
5. It is better if the authors display the reactor geometrical model obtained by SCALE or MCNP input with geometrical dimension information.
6. Below Figure 6, the authors state that : "The results also indicate that fission is caused

approximately 45% by thermal neutrons, approximately 35% by fast neutrons, and approximately 20% by intermediate neutrons" This statement needs more detail explanation. How the authors make additional calculation by using the result from SCALE or MCNP calculation to obtain the values as be stated in this statement
7. Still bellow Figure 6, there is another author statement, i.e. : "These flux diagrams show that the neutron flux in the MUBR is very different from that in any other existing or proposed reactor" The authors compare the MUBR neutron flux spectrum or distribution with neutron flux distribution of other reactors. However, there are no data or graphical diagram which represent the neutron flux spectrum or distribution of some examples of other reactor for comparison in this paper.
8. Also, still below Figure 6, the authors state that "This allows the MUBR to have a high conversion ratio and operate on a true breed-and-burn fuel cycle." What is the calculated value of the MUBR conversion ratio? It better if this value is displayed in this paper to give more information related to breeding capability of MUBR.
9. Why the burn up calculation is only performed by using SCALE, not MCNP?
10. In burn up calculation, is the continuous fission product removal included in the calculation. How to estimate the continuous fission product removal in burn up calculation using SCALE simulation?
11. The last paragraph in Sub Section 4.2 ("MUBR burn analysis") need more detail explanation. The statement that the reactor able to perform true breeder capability for 90 years and then continue with conversion ratio slightly less that 1 for the next 30 years also needs more detail explanation. The explanation must be supported with the calculation result performed in this paper. For example, how the calculated neutron spectrum support this capability. The authors should able to show by using calculated fission reaction based on the calculated neutron spectrum that direct or indirect fission by U-238 dominate the fission reaction in this reactor.
12. More specific references are needed to support the author statement in this paragraph which state that other simulations of MUBR give similar results with the result obtained by simulation performed in this paper
13. The important calculation result should be included in "Conclusion"
14. The "Conclusion" in this paper is too long, longer than Section 4 ("Result"). The Section 4 in this paper is too short because lack of more detail explanation of the calculation result
15. The "Conclusion" in the paper contains very spreading ideas. The "Conclusion" in this paper should be focused on important summary discussed in the section 3 and section 4 ("Method" and "Result"). Many new topics which are not related with this paper appear in the "Conclusion", such as compatibility of material, thermal hydraulics, financial analysis, This is not appropriate.

Is the work clearly and accurately presented and does it cite the current literature?

Partly

Is the study design appropriate and does the work have academic merit?

Yes

Are sufficient details of methods and analysis provided to allow replication by others?

Yes

If applicable, is the statistical analysis and its interpretation appropriate?

Yes

Are all the source data underlying the results available to ensure full reproducibility?

Yes

Are the conclusions drawn adequately supported by the results?
Partly

Competing Interests: No competing interests were disclosed.

Reviewer Expertise: Nuclear engineering

I confirm that I have read this submission and believe that I have an appropriate level of expertise to confirm that it is of an acceptable scientific standard, however I have significant reservations, as outlined above.

Author Response 12 Nov 2024

Neal Mann

Item 3: MUBR6gen software. The software is described in detail in reference 1 - Neutronic Analysis of the Conceptual Molten Uranium Breeder Reactor (MUBR) Using MCNP and SCALE Tools. This article is about the SCALE simulation results and the implications of them.

Item 4: both MCNP and SCALE. MCNP and SCALE can both do criticality calculations but provide slightly different information about the results. Using both provides confidence in the basic results but provides additional information. Table 1 shows the values of the multiplication factor with the standard deviation for both MCNP and SCALE at both extremes of the control method.

Item 6: fission rates by neutron energy. Criticality calculations with MCNP provide the percentage of fission caused by thermal, intermediate, and fast neutrons as part of the output.
Item 8: conversion ratio. Table 2 shows the results of the SCALE burnup simulation through a burnup of 35% of the initial fuel mass. At each step the simulated fissile content (U-235 + Pu-239 + Pu-241) is shown as a percentage of the initial fuel heavy metal mass. At each step through 90 years there is more fissile content than in the original fuel and the mass fissioned is more than 8.8 times the initial fissile mass. Since there is no refueling or fuel manipulation, by definition this is a breeder reactor with a breed and burn fuel cycle.

Item 9: Only SCALE burnup. With the high burnup simulated, the fission products become a significant part of the fuel over time and some of the fission products are significant neutron absorbers. Only SCALE provides a capability to simulate some removal of the fission products, so only SCALE can provide accurate simulation of high burnup with continuous removal of fission products.

Item 10: fission product removal. Continuous rates of removal of elements can be specified in the SCALE input file. The rate is specified as a fraction of the element mass to be removed per unit of time, which is mathematically equivalent to a half life for the element. This allows SCALE to use the same logic for continuous removal of fission products that it uses for continuous removal of isotopes that undergo radioactive decay. The proposed physical removal of fission products is by evaporation from the fuel, so the SCALE input file rate of removal for each element is calculated based on the fuel temperature, the boiling point of the element, and an overall efficiency factor. In the paper's conclusion, the second item of proposed necessary physical experimentation is to determine the actual rates of evaporation of various elements from molten uranium.

Item 11: calculation of results. The results are not calculated based on physics, they are taken from the SCALE output file.

Item 12: other simulation results. Other SCALE burnup simulations were run using MUBR6gen created input files based on different reactor sizes and other reactor parameters. These show similar results and show that the results are robust and can be achieved with various changes in the reactor design or fuel composition. These input files can be made available on the same terms as those cited in the paper.

Item 13: conclusion. Yes, the first sentence of the conclusion should be something like: SCALE burnup simulations show a burnup of 35% of the fuel mass over a 120 year period of continuous reactor operation when the initial fuel fissile content was 2.98 % U-235.

Item 14: conclusion. Conclusions and future work are often included together in the last section of a paper. The title of this paper is "Molten Uranium Breeder Reactor (MUBR) and Its Development Steps", so it is appropriate to discuss the steps needed in the future development of this concept reactor.

Competing Interests: none

Comments on this article

Version 1

Reader Comment 03 Oct 2024

Rhoads Stephenson, Technology, Jet Propulsion Laboratory, Pasadena, USA

I suggest you remove "research and development needs" from Conclusions and put in a separate section. The reactivity control method need a control system analysis to show it can respond quickly enough and is stable.
A detailed thermal analysis should be done of the Fuel start up process. At least two cases: 1. Insertion of Uranium fuel into an evacuated cold primary loop. 2. Remelting of a frozen loop. How will each be done?

Competing Interests: none