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Neutronic simulation, analyses, and performance evaluation of a very small Micro Modular Reactor (MMR) called MARVEL versus different advanced nuclear fuel types

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HIGHLIGHTS

- Developing simulation and analyses of a very high-tech and state-of-the-art MMR.
- Performing accurate neutronic simulation in detail using MCNP6.
- Providing a comparison study between four different advanced fuel types.
- Coming up with more innovations towards industrial or practical applications.
- Enhancing both neutronic and safety parameters practically and coherently.

ABSTRACT

Some prominent and well-known companies or institutes recently developed a new Micro Modular Reactors (MMR) generation. Such a new design mainly includes the monolith core structure, stirring engines, and very tiny or compact designs. MARVEL is one of the most miniature and innovative designs as well. Undoubtedly, such a compact and innovative design has a lot of strategic or multi-purpose applications. Despite their advantages and applications, each new design should be accurately analyzed and checked against different aspects. In this paper, MARVEL, one of the most advanced emerging technologies, is selected for neutronic simulations and in-depth analyses using MCNP. Moreover, the reactor core design is analyzed and evaluated by loading four advanced nuclear fuel types. They are UZRH, UO₂, Uranium Nitride, and MOX. Parameters include the effective multiplication factor, neutron flux distribution, power peaking factor, reactor kinetics, fuel burn up, and consumption. Results are quite promising in utilizing such advanced nuclear fuels for such an advanced compact reactor. Finally, each type may have its advantages and disadvantages as well.

KEYWORDS

MARVEL
MMR
Neutronic Analyses
MCNP
Nuclear Fuel

HISTORY

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1 Introduction

Today, a new generation of nuclear reactors has become fascinating worldwide called Small Modular Reactors (SMRs) and Micro Modular Reactors (MMRs). They are especially proposed to be used for electricity production in remote areas, secondary non-electrical heating or even irradiating applications, and also advanced utilization for deep sea-bed explorations to deep space applications (Liu et al., 2020; McClure and Poston, 2018; McClure et al., 2020; McFall-Johnsen, 2021; Testoni et al., 2021; Zhang et al., 2020).

Although many different conceptual designs have been

introduced worldwide, a few successive action plans have been selected and continued. Furthermore, it seems that some MMRs are more robust, flexible, and practical to be deployed quickly for advanced strategic applications in remote areas (Hedayat, 2020, 2022; IAEA, 2020, 2018, 2024; Zohuri, 2019, 2020).

There are at least 12 advanced conceptual designs for MMRs worldwide. They mainly include Energy Well (Centrum vzskumu e s.r.o., Czech Republic), MoveLuX (Toshiba Energy Systems Solutions Corporation, Japan), ELENA (NRC "Kurchatov Institute", Russian Federation), UNITHERM (NIKIET, Russian Federation), AMR

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(STL Nuclear (Pty) Ltd, South Africa), LFR-TL-30 (NEWCLEO Ltd., United Kingdom), UBattery (Urenco, United Kingdom), Aurora Powerhouse Product Line (Oklo Inc., United States of America), HOLOS-QUAD (HolosGen LLC, United States of America), MARVEL microreactor (Idaho National Laboratory, United States of America), MMR[®] (Ultra Safe Nuclear Corporation, United States of America), and Westinghouse eVinci[™] Micro Reactor (Westinghouse Electric Company LLC, USA).

The MARVEL microreactor is chosen as one of the most advanced and practical case studies for further neutronic analysis in this paper. Monte Carlo simulations are considered for neutronic analyses and safety-related parameters by using MCNP6 (DB, 2008). Some new advanced nuclear fuels may also affect the safety and performance of the reactor as well. They are also studied, compared, and discussed in detail. Results are pretty promising for the future of the last generation of MMRs and advanced nuclear fuels.

2 Materials and Methods

2.1 An introduction to MARVEL

Undoubtedly, the MARVEL microreactor (IAEA, 2018; INL, 2021) is one of the most advanced and strategic MMRs worldwide. MARVEL is a sodium-potassium-cooled microreactor that will generate 85 kilowatts of thermal energy (DOE, 2023). It is also easy to relocate and install. There are some test facilities to improve the Technological Readiness Levels (TRL) in practice. This design's non-active prototype (i.e., a full-scale non-active test loop) was recently built and successfully tested. The final prototype will be built and operated at Idaho National Laboratory (INL) in the United States of America (USA) by 2025 (DOE, 2023). Then, it will be connected to a microgrid. It can supply the electrical energy of a field hospital with 180 beds. Moreover, it is also well-established to utilize co-generation applications in advance. Therefore, MARVEL is supposed to be one of the most compact, integrated, and advanced land-based MMRs ever designed and tested successfully (Fig. 1). The nuclear core of MARVEL (Fig. 2) consists of the Uranium-Zirconium-Hydride fuel with 19.75% enrichment. The reactor core includes 36 fuel pins moderated by Hydrogen in UZRH. The reactor also utilizes four Helium Stirling engines and pumps to produce the electricity.

2.2 Methodology

At first, we must set our aims, methods, aspects, and goals for such a conceptual analysis. Neutronic and safety-related parameters are the most critical aspects of a nuclear reactor design and safety analysis. The MCNP6 (DB, 2008) is chosen for realistic simulation of neutron behavior and tracking in the reactor core based on the Monte Carlo simulation and sampling. For more detail, the MCNP6 code (DB, 2008) is one of the most prominent and practical simulating tools for transporting nuclear particles based on Monte Carlo simulations.



Figure 1: 3D view of MARVEL.

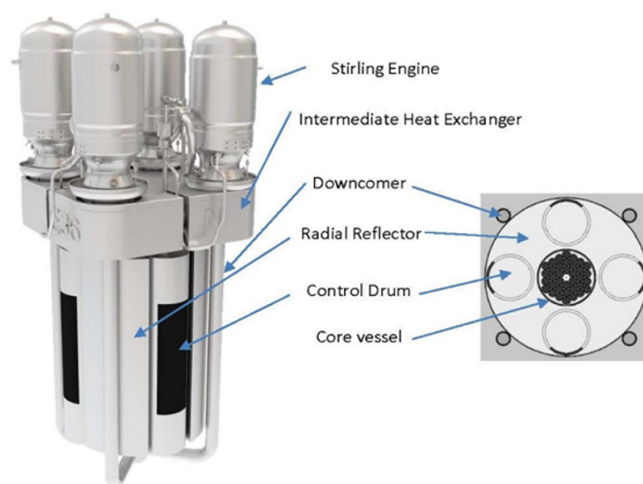


Figure 2: Main components of MARVEL.

It is also able to do the criticality and perturbation calculations required for the reactor analysis as well. Even though it does not solve an explicit transport equation directly, it can obtain answers reliably by simulating the stochastic behavior of nuclear particles instead. MCNP simulates and analyzes the particle tracks and destinations for a determined statistical sample. After that, the average behavior of particles in the chosen physical system can be inferred from the statistical point of view.

In this paper, the neutron multiplication factor, reactor excess reactivity, radial and axial neutron flux distribution, Power Peaking Factor (PPF), neutron generation time, effective delayed neutron fraction, core cycle length, and burnup effects are chosen for further parametric analysis in detail. Finally, the primary goal of this research paper is to discuss the pros and cons of utilizing different advanced nuclear fuels for such an innovative advanced nuclear reactor.

2.3 The reactor core modeling

In this research, the MCNP6 (DB, 2008) is used to simulate and analyze the MARVEL core (Table 1) in detail. It

Table 1: Main specifications of the MARVEL reactor.

Major Systems	Parameters	Value/Type	Units
Core	Thermal Power	100	kWth
	Core Life	2	years
	Fuel Type	(UZRH)	-
	Fuel Uranium Enrichment	<19.75	%U-235
	Maximum Uranium in Core	<30	kg U
	Number of Fuel Pins	36	-
	Neutron Moderator	Hydrogen in UZRH	-
coolant	Peak Cladding Temperature	550	°C
	Heat-Transfer Method (Sodium-Potassium Eutectic [NaK])	Liquid-Phase Natural Circulation 120	- kg
Reactivity Control	Safety-related control drums	Four/ex-core	-
	Defense-in-depth shutdown rod	One/Central	-
Reflector & shield	Neutron Reflector Material	Beryllium Oxide, Beryllium Metal	-
	Neutron Absorber Material	Boron Carbide (B ₄ C)	-
	Neutron Radiation Shield Material	Boron Carbide	-
	-	Borated Polyethylene	-
	Gamma Radiation Shield	Stainless Steel	-
-	-	Concrete	-

mainly has 36 cylindrical fuel rods. They are 98.5 cm and 36 in length and diameter, respectively. Next, the Control drums and rods should also be taken into account. In practice, these control rods can control or adjust the suitable nuclear fission rates or the reactor power. The control rods are 91 cm and 18.4 cm in length and diameter, respectively. They are mainly composed of B₄C absorber material. Each drum also has 15 neutron reflector rods inside. In other words, the drums can act as an absorber or reflector, depending on the rotating position.

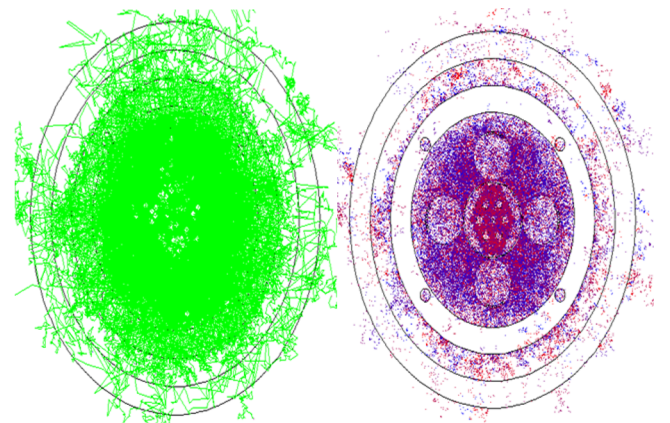
The beryllium oxide rods surround the reactor core. They shape the reactor reflector that enhances the neutron economy inside the active part of the reactor core. They are 90 cm and 5 cm in length and diameter, respectively. Finally, the reactor core and reflector are capsulated within the reactor shields and the vessel. They are composed of different parts and layers. The first layer is filled with air and has a length of 120 cm and a diameter of 80 cm. The next layer is made of stainless steel and has a length of 180 cm and a diameter of 100 cm. The third layer contains B₄C and has a height of 245 cm and a diameter of 120 cm. Finally, the last layer is made of stainless steel and has a height of 345 cm and a diameter of 274 cm, respectively. It is worth mentioning that this simulation includes all parts and materials as well. Moreover, four different fuel types are chosen for further simulations and analyses. They include UZRH, UO₂, Uranium Nitride, and MOX fuels. Table 2 presents the isotopic fractions for the mentioned fuels.

2.4 Statistics

For such a Monte Carlo simulation in a complex system, the number of particles should be considered deliberately. The more particles simulated, the more accurate the results, but the longer the calculation takes. The number of particles used in a simulation is usually chosen based on the desired level of statistical uncertainty and available computers.

Table 2: Fuel isotopes (UZRH, UO₂, Uranium Nitride, MOX).

Fuel Type	Fuel Isotopes	Ratio
UZRH	92234	0.000057
	92235	0.007204
	92238	0.992739
	40000	0.97838
	1001	0.021262
UO ₂	8016	0.118533
	92234	0.000057
	92235	0.007204
Uranium Nitride	92238	0.992739
	7014	0.055590
	92234	0.000057
MOX	92235	0.007204
	92238	0.992739
	92236	0.000122
	92238	0.992739
	94238	0.001030
	94239	0.022532
	94240	0.010751

**Figure 3:** Particle tracking in MARVEL core using MCNP6.

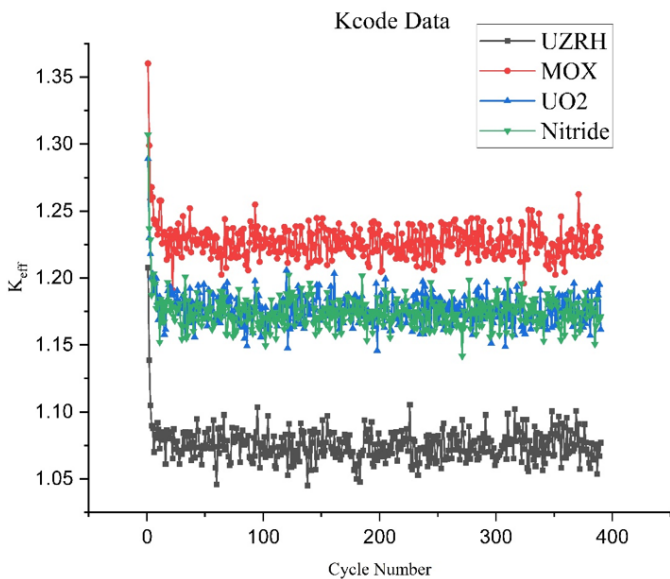


Figure 4: K_{eff} for all four fuels (UZRH, UO_2 , Uranium Nitride, MOX) versus active cycles.

Table 3: K_{eff} (UZRH, UO_2 , Uranium Nitride, MOX).

Fuel type	K_{eff}	Multiplication coefficient in the presence of control elements
UZRH	1.06593	0.00042
UO_2	1.11937	0.00046
Uranium Nitride	1.11589	0.00042
MOX	1.17863	0.00048

Table 4: Neutron flux (UZRH, UO_2 , Uranium Nitride, MOX).

Fuel type	Tally(F4)	Flux ($n \cdot cm^{-2} \cdot s^{-1}$)
UZRH	3.9405E-01	4.7820E+11
UO_2	4.4390E-01	4.7475E+10
Uranium Nitride	4.4116E-01	2.8252E+10
MOX	4.8148E-01	3.9807E+11

Users usually adjust those cycle numbers according to the desired statistical and physical uncertainties, the problem complexity, available hardware, and the economy of time. In this research, 50000, 50, and 400 are considered and set for the particle numbers, inactive and active cycles, respectively. Then, it is also necessary to check the particle tracking (Fig. 3) in the core. In other words, the neutron population should grow and spread all over the active part of the reactor core before the beginning of the active cycles.

3 Results and discussion

3.1 Multiplication Factor

MCNP can calculate this parameter using the KCODE card, an initial guess, the neutron population number, and cycle (i.e., inactive and active) numbers. Figure 4 presents the variation of the K_{eff} versus active cycles. Table 3 also

presents the final results of K_{eff} for the four different fuels. According to the results, MOX fuel has the highest multiplication factor, while UZRH fuel has the lowest value. The MOX fuel has Pu with a higher fission cross-section, while the UZRH fuel is a strong solid moderator. Therefore, results are meaningful due to the physics of the problem.

3.2 Neutron Flux

In MCNP calculations, sampled neutrons are simulated and tracked. Then, the code counts them through the chosen surface or volume. Tally F4 is usually more suitable and practical for neutron flux estimation. Results are raw data and need to be normalized. The neutron flux can be normalized using the following equation:

$$S = \frac{P[W] \times \nu \left[\frac{\text{neutron}}{\text{fission}} \right]}{(1.6022 \times 10^{-13} \left[\frac{J}{MeV} \right]) \times \omega_f \left[\frac{MeV}{\text{fission}} \right]} \quad (1)$$

where P is the power, ν is the number of neutrons emitted per fission, and ω_f is the average energy released for each fission. Table 4 presents the neutron flux for the chosen fuels in MARVEL. UZRH and MOX fuel sequentially have the maximum neutron fluxes over the reactor core. UZRH has the strongest moderator that can produce and keep more thermal neutrons. MOX fuel has the most influential fission neutron cross section and delayed neutrons. Therefore, they can be more effective and influential in the active part of the reactor core.

Next, Figs. 5 to 8 present the radial neutron flux distribution over the MARVEL core for different nuclear fuels. These figures show similar shapes and functionality except for the maximum values at centers. Additionally, results indicate that UZRH and MOX are more suitable for the economy of neutrons and fuels in the determined core structure.

Figures 9 and 10 present the axial distribution of thermal and fast neutron fluxes along the reactor core for four fuels. Results indicate that axial neutron fluxes are more sensitive to the fuel type. Results show that the most fluctuation and peaks belong to the MOX and UO_2 fuels regarding the thermal and fast neutron fluxes, respectively. In other words, UZRH and Nitride fuels are better for axial power flattening. Next, Fig. 11 depicts normalized neutron density over the reactor core for chosen fuels (UZRH, UO_2 , Uranium, MOX). Even though they all represent a thermal spectrum, we need something more separable and characteristic. Figure 12 shows the neutron spectrum magnified over the thermal range. As expected, UZRH has the softest neutron spectrum, while MOX has the hardest neutron spectrum instead.

3.3 Power distribution

Power distribution and Power Peaking Factors (PPF) significantly influence the thermal-hydraulic and safety of the reactor core. Besides, axial power distribution is essential when calculating thermal-hydraulic parameters. It can be calculated as follows:

$$P(z) = \frac{\phi(z) \times V(z) \times \sigma f(z)}{\Sigma f(z) \times \nu} \quad (2)$$

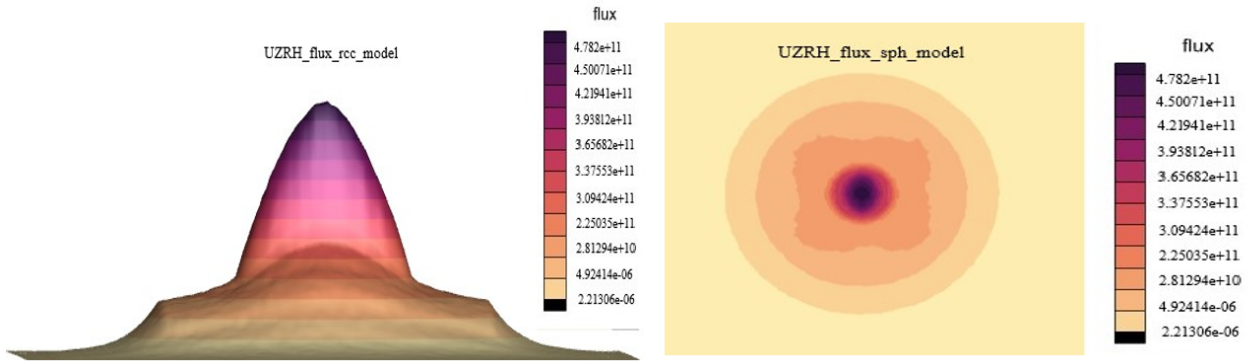


Figure 5: Radial neutron flux distribution at the critical state for UZRH fuel.

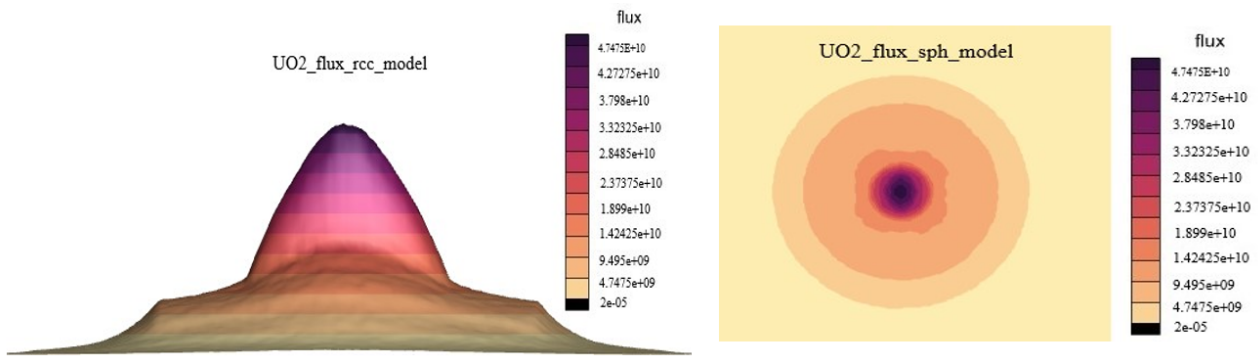


Figure 6: Radial neutron flux distribution at the critical state for UO₂ fuel.

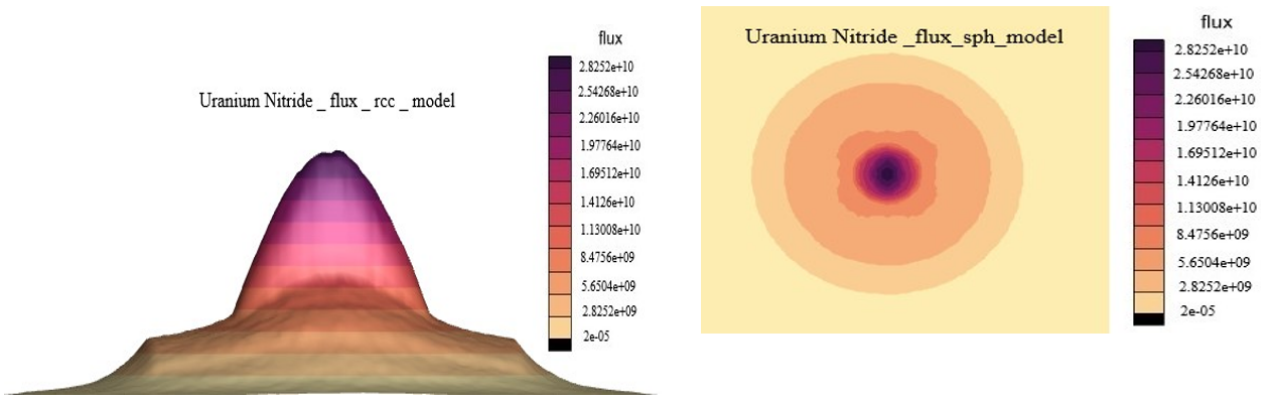


Figure 7: Radial neutron flux distribution at the critical state for Uranium Nitride fuel.

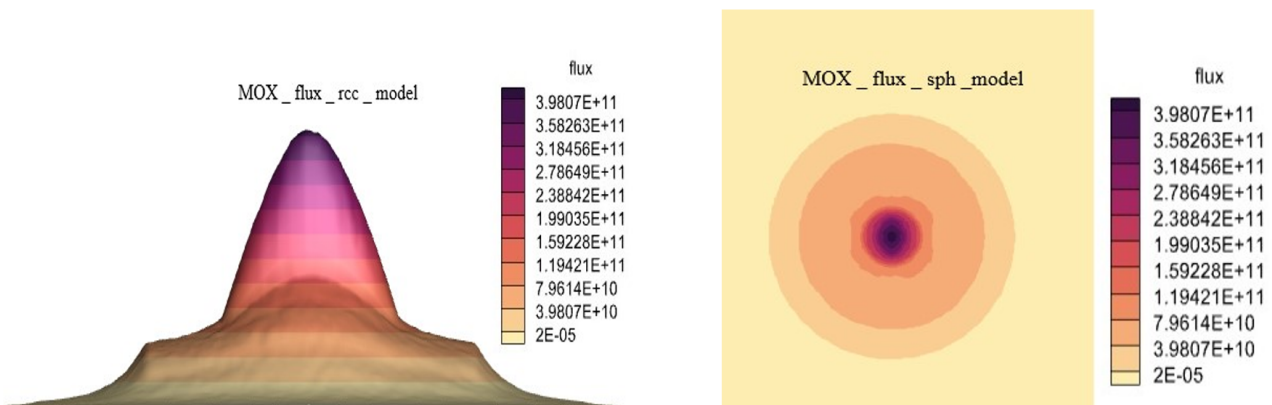


Figure 8: Radial neutron flux distribution at the critical state for Uranium Nitride fuel.

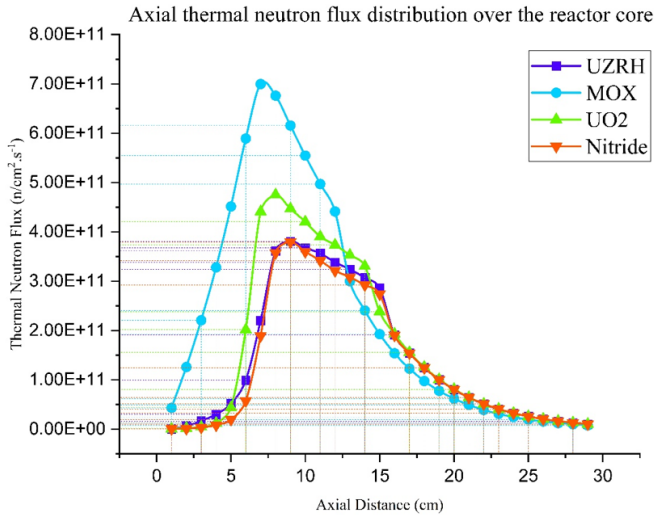


Figure 9: Axial thermal neutron flux distribution over the reactor core.

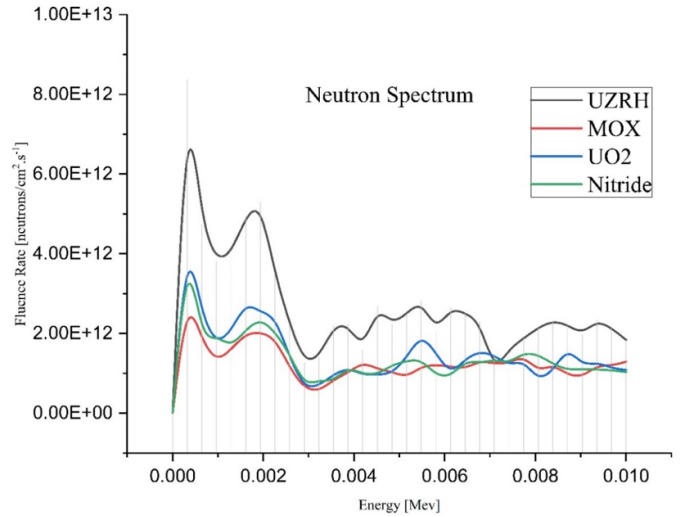


Figure 12: Neutron Spectrum over the reactor core for chosen fuels (UZRH, UO₂, Uranium, MOX).

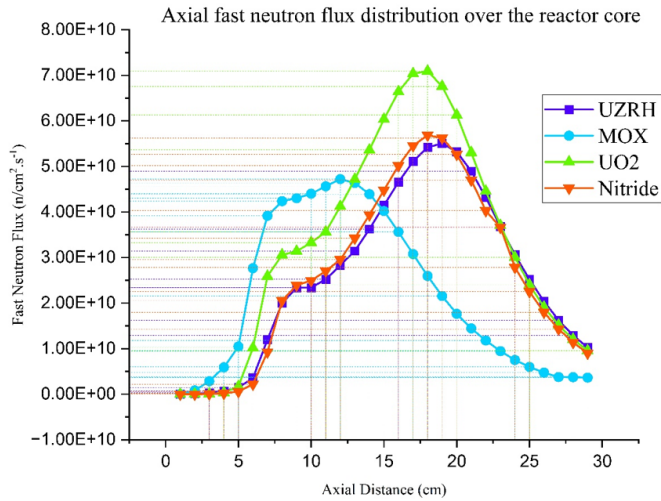


Figure 10: Axial fast neutron flux distribution over the reactor core.

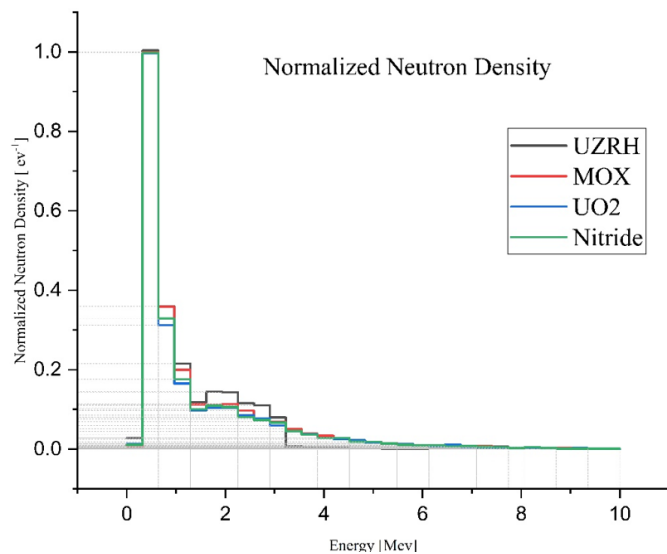


Figure 11: Normalized neutron density over the reactor core for chosen fuels (UZRH, UO₂, Uranium, MOX).

Table 5: PPF (UZRH, UO₂, Uranium Nitride, MOX).

Fuel type	Maximum PPF	Average PPF	PPF Domain
UZRH	1.2050	0.9546	0.7042 - 1.2050
UO ₂	1.2500	1.0096	0.7692 - 1.2500
Uranium Nitride	1.2660	1.0118	0.7576 - 1.2660
MOX	1.2820	1.03785	0.7937 - 1.2820

where $P(z)$ is the power at the z position, $\phi(z)$ is the neutron flux at the z position, $V(z)$ is the volume at the z position, $\sigma f(z)$ is the microscopic fission cross-section at z position. $\Sigma f(z)$ is the macroscopic fission cross-section at position z , and ν is the average number of neutrons produced per fission.

Figures 13 and 14 show the radial and axial PPF distribution over the reactor core, and Fig. 15 presents the PPF distribution at each fuel pin. After all, Table 5 presents the average, maximum, and domains of PPF values for each fuel type. Results indicate that UZRH and MOX have the lowest and highest values, respectively. It means that the power distribution strongly correlates with the fuel type used. On the one hand, UZRH is more suitable for the reactor safety. On the other hand, MOX fuel may be more interactive and practical to be used for fast load following instead.

3.4 Fuel consumption and burnup effects

Fuel burnup measures the amount of energy that can be extracted from a given mass of nuclear fuel. It is typically expressed in Giga-Watt-days per metric ton of Heavy Metals (GWd/tHM). The burnup of nuclear fuel depends on various factors, including the composition of the fuel, the operating conditions, particularly the reactor power, and the core cycle length.

In general, although the fuel burn up and consumption depend on many different factors, they can affect different Operational Limits and Conditions (OLC) at the same

time. Therefore, understanding and analyzing those factors and parameters is very important for coherently improving the performance and safety of the reactor. We seek the best nuclear fuel to maximize released energy while maintaining reactor safety. The first step is to analyze such neutronic and burnup parameters.

This research performs the fuel consumption calculations using the CINDER90 code (i.e. an upgraded version of the CINDER code that is built-in for the MCNP6 code). The appropriate card for calculating fuel consumption is the BURN card. It mainly includes the following options: TIME (i.e. for the selection of appropriate time in days), MAT (i.e. for the selection of the desired materials for fuel consumption), POWER (i.e. to introduce the power in megawatts), PFRAC (i.e. to determine the power fraction or sometimes to set a normalization factor), MATVOL (i.e. to introduce the total volume of cells

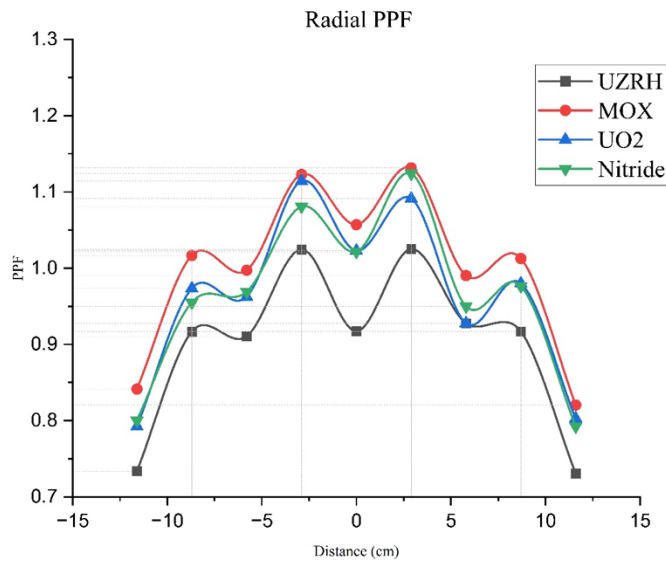


Figure 13: Radial Power Peaking Factor (PPF) for chosen fuels (UZRH, UO₂, Uranium, MOX).

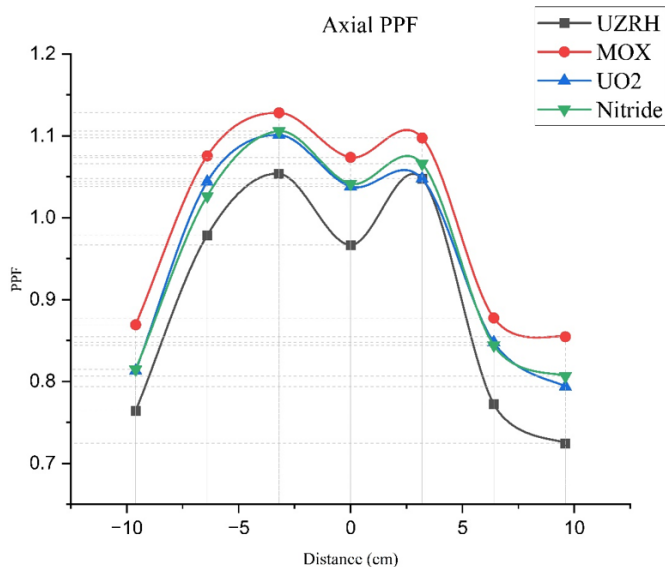


Figure 14: Axial Power Peaking Factor (PPF) for chosen fuels (UZRH, UO₂, Uranium, MOX).

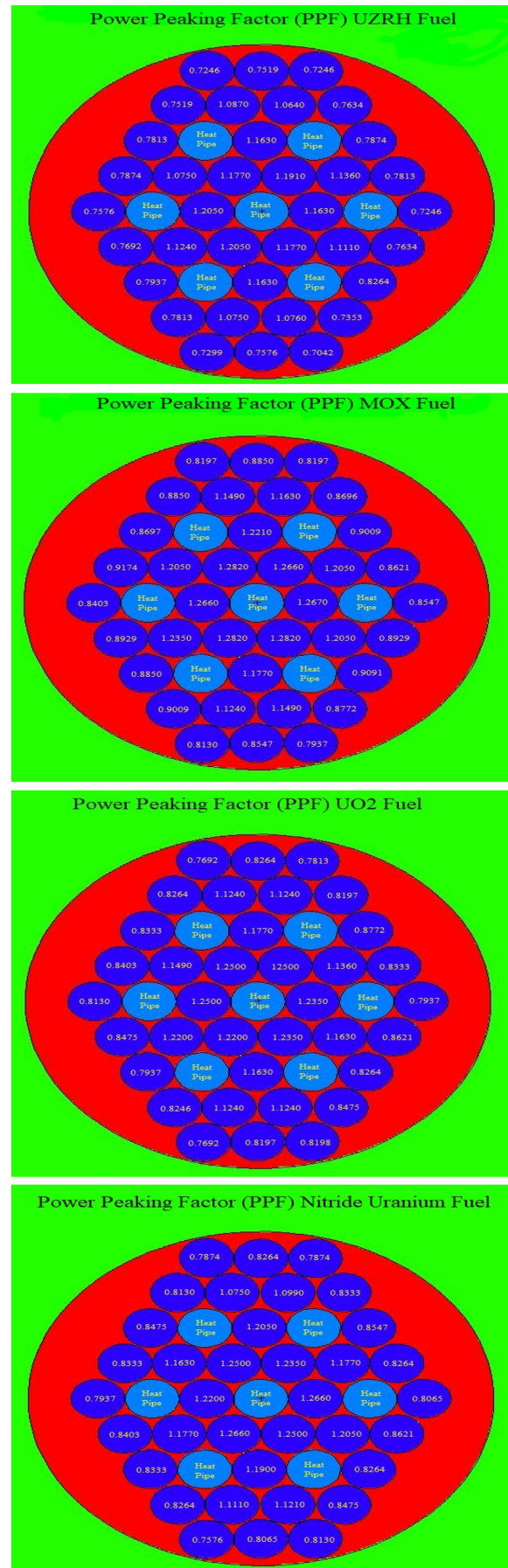


Figure 15: Radial Power Peaking Factor (PPF) distribution over the fuel pins for chosen fuels (UZRH, UO₂, Uranium, MOX).

Table 6: Fuel burnup rate (GWD/MTH) (UZRH, UO₂, Uranium Nitride, MOX).

Fuel Type	TIME (days)	BURN UP (GWd/tHM)	Initial Mass (g)	Consumption Percentage (g)
UZRH	30	5/46E-02	25000	-
	100	2/37E-01	-	-
	170	4/03E-01	-	-
	240	5/69E-01	-	-
	310	5/83E-01	-	%43.23
UO ₂	30	2/38E-03	25000	-
	100	1/90E-02	-	-
	170	3/35E-02	-	-
	240	4/73E-02	-	-
	310	4/86E-02	-	%14.54
Uranium Nitride	30	2/20E-03	25000	-
	100	1/95E-02	-	-
	170	3/27E-02	-	-
	240	4/69E-02	-	-
	310	4/77E-02	-	%15.1
MOX	30	5/68E-02	25000	-
	100	2/69E-01	-	-
	170	4/24E-01	-	-
	240	5/89E-01	-	-
	310	6/01E-01	-	%45.1

filled by fuel materials in cubic centimeters), OMIT (i.e., to set the isotopes that can be ignored because they are necessary for the considered problem), BOPT (i.e., as a control card for printing the output, for example, the printing according to atomic or mass fractions), AFMIN (i.e., to introduce the atomic fraction of isotopes that is considered in the transfer calculation). It is also worth mentioning that the fuel consumption unit in the MCNP6 code is the Gigawatt-Day per Metric Ton of heavy elements.

Table 6 presents the burnup for the chosen fuels (i.e., UZRH, UO₂, Uranium Nitride, and MOX). For each fuel type, the table provides the burnup rate (in GWd/tHM) at five different time steps (i.e., 30, 100, 170, 240, and 310 days). The table also represents the primary mass of each fuel type (in grams) and the final fuel consumption percentage after the 310 days of burnup.

Results show that the burnup rates for all four fuel types generally increase with time. As the fuel is exposed to the neutron flux in the reactor, it undergoes fission rates and produces thermal energy. The longer the fuel is in the reactor, the more fissions occur. Therefore, higher burnup rates can release higher energy, but high burnup can also affect radiation damage and restrict the maximum discharge limit.

Results also indicate that the burnup rate varies significantly between fuel types. MOX and UZRH fuels have the highest burnup rate, reaching almost 0.6 and 0.58 GWd/tHM after 310 days, respectively. UO₂ and uranium nitride have much lower burnup rates, reaching only around 0.047 GWd/tHM after 310 days.

As a result, the consumption percentage of each fuel type also varies significantly. Similarly, MOX and UZRH have the highest burnup, with over 45% and 43% of the fuel consumed after 310 days, respectively. UO₂ and uranium nitride have much lower consumption rates, with only around 15% of the fuel consumed after 310 days.

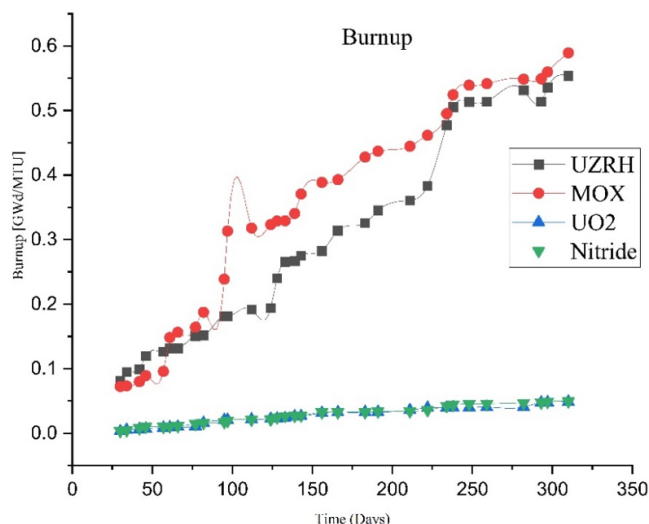


Figure 16: Burnup chart for MOX fuel.

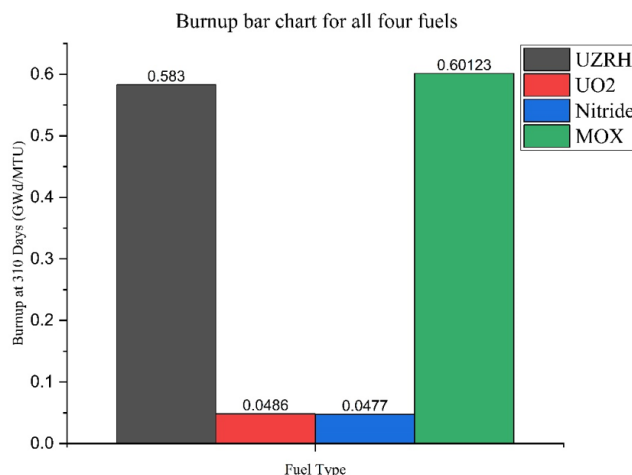


Figure 17: Burnup bar chart for all four fuels (UZRH, UO₂, Uranium Nitride, MOX).

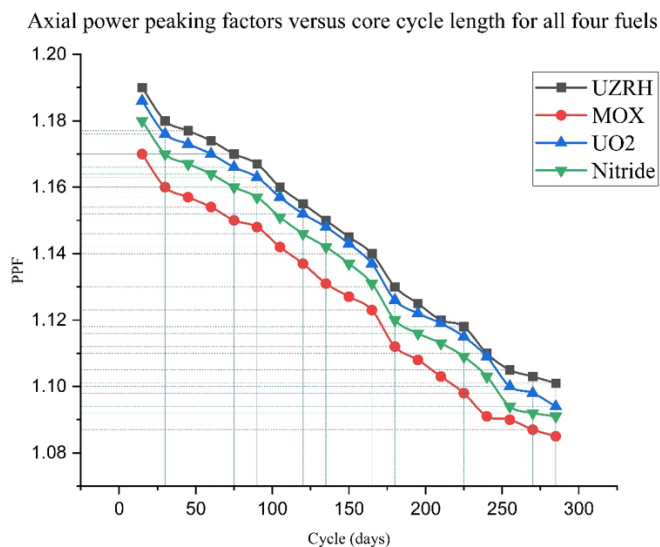


Figure 18: Axial power peaking factors versus core cycle length for all four fuels (UZRH, UO_2 , Uranium Nitride, MOX).

Table 7: Kinetics (UZRH, UO_2 , Uranium Nitride, MOX).

Fuel type	Delayed Neutron Fraction	Neutron Generation Time (μs)
UZRH	0.1029	65.9
UO_2	0.0257	119
Uranium Nitride	0.0305	116
MOX	0.1040	179

Figure 16 presents burnup effects on shape individually for all four chosen fuels (UZRH, UO_2 , Uranium Nitride, MOX). Next, Fig. 17 presents a good comparison among them. Concerning the capacity and capability of energy production, the best activity goes to the MOX and UZRH in an orderly manner. Furthermore, Fig. 18 presents the variation of axial PPF all over the reactor core for four different nuclear fuels. Finally, concerning energy performance and nuclear safety, MOX fuel is the best choice for fuel burnup.

3.5 Kinetic Parameters

In addition to general neutronic parameters (i.e., considering the static behavior of neutrons), some other characteristics are directly correlated with the reactor's dynamic system. They contribute to fast responses, especially in terms of nuclear safety. They mainly include the Delayed Neutron Fraction (DNF) and neutron generation time. Undoubtedly, the fuel composition or reactor core design can affect those parameters. Those dynamic parameters can influence the nuclear safety feature of the reactor. This factor is imperative during a Reactivity-Induced Accident (RIA). The second kinetic parameter is the neutron generation time. This parameter represents the average time needed for a neutron to be released from one fission and then produce another fission. This parameter is important because it affects the overall behavior and stability of the reactor as well.

Neutron generation time is usually expressed in microseconds and is dominantly influenced by the fuel composition and design. Table 7 presents those kinetic parameters for four different fuels. Results indicate that the MOX fuel has the best kinetic parameters to control the reactivity and mitigate the unwanted transients more safely.

4 Conclusions

In this paper, the MARVEL microreactor is simulated and analyzed in detail according to the neutronic point of view using MCNP6. It is imperative because it is one of the most advanced and futuristic designs among new MMRs. It is supposed to be in operation soon. Moreover, the reactor core has been analyzed to utilize advanced nuclear fuels. They are UZRH, UO_2 , Uranium Nitride, MOX.

The results confirm that K_{eff} differs significantly among chosen nuclear fuels. MOX and UZRH fuels have the highest and lowest values of 1.17863 and 1.06593, respectively. The UO_2 and Uranium Nitride fuel types have similar K_{eff} values of about 1.11937 and 1.11589, respectively.

The influence of the control rod also varies slightly between the fuel types, with the MOX fuel type having the highest value at 0.00048 and the UZRH fuel type having the lowest value at 0.00042, while drums are rotated in the maximum reactivity functionality at BOC for fresh core due to conservative issues.

Neutron flux distributions are also represented for the chosen fuels as well. The results confirmed that the neutron flux varies significantly among the different fuel types. The highest neutron flux was observed in the UZRH fuel of about $4.7820E+11$. Next, the MOX fuel has the second rank with $3.9807E+11$ ($\text{n}\cdot\text{s}^{-1}\cdot\text{cm}^{-2}$), while the UO_2 and Uranium Nitride fuel types have lower neutron fluxes of about $4.7475E+10$ and $2.8252E+10$, respectively. These results can also be used to optimize the core configuration design and operation according to the neutron flux distribution.

This paper also presents a detailed analysis of fuel burnup and consumption as a function of time. The burnup rate and the fuel consumption are given in GWD/MTU and percentage, respectively. The results show that the burnup rate and consumption vary significantly among the different fuel types and over time. The MOX fuel type has the highest consumption of about 45.1% after 310 days. Next, UZRH has the highest burnup rate and consumption of about 43.23% after 310 days. The UO_2 and Uranium Nitride fuel types have lower burnup rates and consumption, with 14.54% and 15.1% of the primary mass consumed after 310 days, respectively.

After that, the paper also presents the main kinetic parameters for the four chosen nuclear fuels. They are the delayed neutron fraction (DNF) and prompt neutron time (PNT). The DNF measures the delayed neutrons emitted by fission products after a particular time, while the PNT measures the time for most prompt neutrons to be emitted after a fission event. The results show that the DNF varies significantly among the different fuel types. The highest

DNF was observed in the MOX fuel type at about 0.1040, while the lowest was in the UO₂ fuel type at about 0.0257. The UZRH and Uranium Nitride fuel types have intermediate DNF values of about 0.1029 and 0.0305, respectively. The PNT also varies slightly among different types. The UZRH fuel type has the lowest PNT value of about 65.9 s, while the MOX fuel has the highest PNT value of about 179 μ s. These kinetic parameters are essential for the dynamic simulation and nuclear safety analyses.

Shortly and briefly, results show that MOX fuel can enhance the design's neutronic performance and safety. It is also worth mentioning that UZRH may have some advantages. Mostly, the neutron spectrum of the UZRH is the softest neutron spectrum toward the thermal energy over the reactor core. In other words, optimizing material distribution and object planning by artificial intelligence techniques could easily change the MOX to be the best option, dominantly and practically, for such a high-tech design in the future.

Conflict of Interest

The authors declare no potential conflict of interest regarding the publication of this work.

References

- DB, P. (2008). MCNPX TM User's Manual Version 2.6.0. *Los Alamos, NM: Los Alamos National Laboratory*.
- DOE (2023). MARVEL Microreactor Reaches Final Design Step, Office of Nuclear Energy, <https://www.energy.gov/ne/articles/marvel-microreactor-reaches-final-design-step>. *MARVEL*.
- Hedayat, A. (2020). A review of advanced SMRs particularly iPWRs regarding safety features, economy issues, innovative concepts, and multi-purpose deployment. *Radiation Physics and Engineering*, 1(4):29–53.
- Hedayat, A. (2022). The role of advanced nuclear reactors in non-electrical and strategic applications, producing sustainable energy supplies and reducing the greenhouse gasses. *Kerntechnik*, 87(5):579–596.
- IAEA (2018). Advances in small modular reactor technology developments. a supplement to: IAEA advanced reactors information system (ARIS).
- IAEA (2020). Advanced Reactors Information System, ARIS International Atomic Energy Agency (IAEA).
- IAEA (2024). Advanced Reactors Information System, ARIS International Atomic Energy Agency (IAEA).
- INL (2021). Final Environmental Assessment For The Micro Reactor Applications Research; Validation; and Evaluation (MARVEL) Project at Idaho National Laboratory.
- Liu, X., Zhang, R., Liang, Y., et al. (2020). Core thermal-hydraulic evaluation of a heat pipe cooled nuclear reactor. *Annals of Nuclear Energy*, 142:107412.
- McClure, P. R. and Poston, D. I. (2018). Kilopower-Powering a NASA Mission to Mars, *Frontiers in Science Public Lecture Series*. Technical report, Los Alamos National Laboratory (LANL), Los Alamos, NM (United States).
- McClure, P. R., Poston, D. I., Gibson, M. A., et al. (2020). Kilopower project: the KRUSTY fission power experiment and potential missions. *Nuclear Technology*, 206(sup1):S1–S12.
- McFall-Johnsen, M. (2021). NASA's attempt to burrow into Mars met 2 insurmountable obstacles: cement-like soil and an unexpected energy shortage, *BUSINESS INSIDER*.
- Testoni, R., Bersano, A., and Segantin, S. (2021). Review of nuclear microreactors: Status, potentialities and challenges. *Progress in Nuclear Energy*, 138:103822.
- Zhang, W., Zhang, D., Wang, C., et al. (2020). Conceptual design and analysis of a megawatt power level heat pipe cooled space reactor power system. *Annals of Nuclear Energy*, 144:107576.
- Zohuri, B. (2019). *Small modular reactors as renewable energy sources*. Springer.
- Zohuri, B. (2020). *Nuclear micro reactors*. Springer.

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