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Review of Neutron Physics Analyses for Accident-Tolerant Fuel System

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Abstract

After the 2011 Fukushima nuclear accident, scientists and engineers have been developing accident-tolerant fuel (ATF) concepts to make nuclear reactors safer. To make light-water reactor fuel-cladding systems more resistant to accidents, researchers are proposing solutions to the problem of Zr-based cladding oxidation. Researchers propose improving the accident tolerance of current fuel cladding systems by enhancing Zr-based alloys, coating Zircaloy, or using new cladding/fuel materials. Neutron physics is essential for evaluating the feasibility of these solutions. It will be used to assess not only their safety but also their economic viability. This review summarises the nuclear industry's efforts to investigate the reactor physics impact of accident-tolerant fuel (ATF) concepts on current and future reactor designs. Important reactor physics parameters that affect safety and economics include reactivity coefficients, cycle length, neutron spectrum, excess reactivity, and control rod worth. This work can be a reference for the nuclear community, especially reactor physicists when studying the implementation of the ATF concept.

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1. Introduction

The nuclear industry has a long history of responding to accidents like Three Mile Island and Fukushima by significantly improving safety. For example, after the TMI accident 1979, the Nuclear Regulatory Commission (NRC) implemented several regulatory changes designed to enhance reactor safety. These changes included enhanced training and protocols for plant operators, upgraded equipment and instrumentation, and new regulations targeting the design and operational factors that contributed to the accident. In addition, the nuclear industry has strongly emphasized developing a safety culture that values continuous learning and improvement.

After the Fukushima accident in 2011, the nuclear industry comprehensively reviewed its safety practices and systems. Working closely with the regulators, the sector identified and implemented several new safety measures. These measures included:

- Strengthening emergency response capabilities
- Improving protection against earthquakes and flooding
- Enhancing backup power and cooling systems
- Adopting passive hydrogen recombiners and filtered containment ventilation systems (FCVS)[1,2]

Although no system is perfect against the forces of nature, the nuclear industry has learned essential lessons from the Fukushima accident. It has made significant progress in improving the safety of nuclear power plants through a combination of regulatory changes, improved practices and procedures, and the adoption of innovative technologies.

The Fukushima-Daichi accident has prompted a search for alternatives to using zirconium-based cladding for nuclear fuel rods. Accidents like Fukushima can be divided into three phases: lead-up, mid-phase, and late-phase. In the lead-up phase, triggered by a station blackout (SBO) or a pipe break,

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the decay heat and stored energy in the fuel cause the coolant in the core to boil. This leads to an increase in temperature in the depressurized core. When temperatures reach 800 degrees Celsius, the core begins to degrade physically and chemically. This marks the start of the mid-phase. Potential fuel rod rupture can occur within the 700-1200 degrees Celsius temperature range. Additionally, under these conditions, zirconium-based cladding undergoes exothermic oxidation that generates heat and hydrogen. This heat production can be greater than the decay heat, driving the temperatures in the core higher to reach around 1500 degrees Celsius. This is known as the late phase, which leads to severe damage to the core.

A comparison of ATF cladding with zirconium cladding has been made for the extreme event when the Emergency Core Cooling System (ECCS) is unavailable.[3] Other studies have shown that ATF claddings can sustain station blackout accidents much longer than Zr-alloy cladding.[4] Better coping times are also reported in different studies.[5] The study shows that ATF cladding can delay the balloon and burst significantly compared to zirconium cladding. The cladding temperatures also showed a significant difference compared to the zirconium system.

Accident-tolerant fuel (ATF) is expected to improve the reaction kinetics of cladding materials against steam, fuel, and cladding properties and retention of fission products.[6] The nuclear community has agreed that ATF should perform as well as or better than current options.[7–10] ATF aims to eliminate hydrogen production from high-temperature steam and cladding reactions, which can lead to beyond-design-basis accidents (BDBAs) in the core. The ATF concept is critical because 96% of the world's 442 operating power reactors are water-cooled.[11] ATF development is a top priority because most current reactors are vulnerable to Fukushima-like accidents.

There are three main ATF options, which can be used alone or in combination:

1. Coating zirconium-based cladding with a suitable material to protect it from reacting with high-temperature steam.
2. Replacing zirconium-based cladding with FeCrAl, stainless steel, or silicon carbide materials. Tables 1 and 2 list the compositions and relevant thermal properties of some materials being considered to replace zircaloy.
3. Replacing uranium dioxide fuel with other materials with better thermal properties and neutron physics advantages, such as larger heavy metal (HM) density (UN, U_3Si_2, *etc.*). Fig. 1 compares the neutron capture cross-sections of materials currently used or proposed as potential replacements.

Nuclear reactor design is a complex optimization problem that involves many interacting factors, such as neutron physics, thermal hydraulics, mechanical and material considerations, and economics. A change to these factors can directly or indirectly affect the others. Fig. 2 shows a simplified flowchart of the nuclear reactor design process. It also illustrates how neutron physics design with a specific cycle length can directly impact plant economics and thermal hydraulics design.

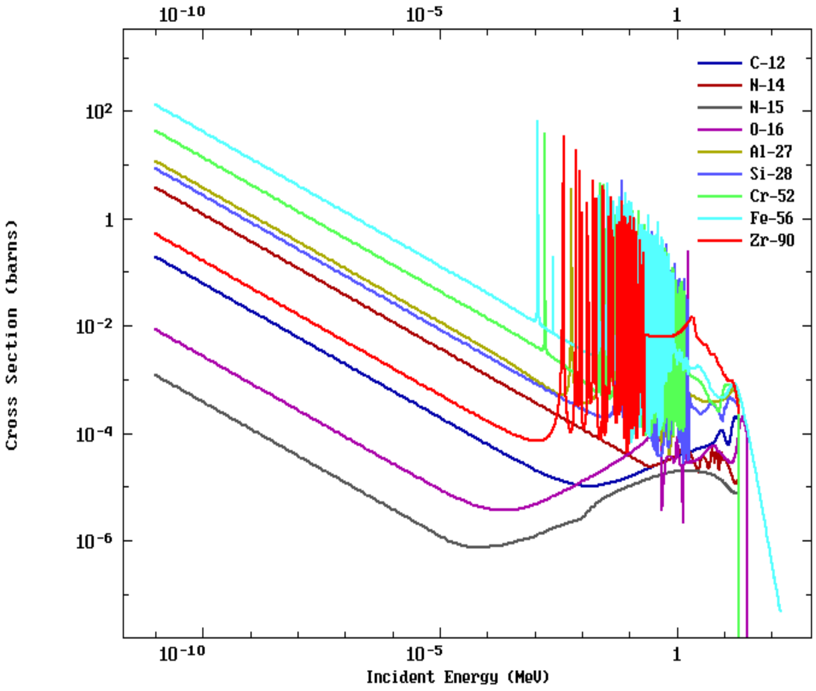


Fig. 1 Neutron capture cross-section comparison of different ATF materials.

Table 1. Atom percent (at%) composition for candidate cladding materials.

	Zircaloy	SS304	SS310	FeCrAl	APMT	SiC
Fe	0.24	70.44	51.72	70.2	65.84	
Cr	0.17	20.04	26.66	20.11	21.89	
Al				9.69	9.57	
Zr	98.43				0.06	
Ni		7.84	18.27			
Sn	1.15					
Mn		0.70	1.9			
Mo		0.16	0.07		1.54	
Y					0.07	
Si		0.82	1.37		0.99	50.0
Hf					0.05	
C						50.0

Table 2. Properties of candidate cladding materials.

Properties/Material	Zircaloy	SS304	SS310	FeCrAl	APMT	SiC
Density (g/cm^3)	6.57	7.90	8.03	7.10	7.30	2.58
Absorption Cross-section (barns)	0.20	2.860	3.210	4.43	2.470	0.086
Thermal expansion (µm/m-K)	6	17.3	14.4	15	12	4
Thermal conductivity (W/m-K)	14.5-14.2	16.2	16.2	15.81	11-21	126.43
Specific heat (kJ/kg -K)	0.285-0.368	0.49-0.53	0.502	0.6697	0.48-0.71	1.313
Melting point (K)	2122	1425	1425	1773	1500	3003

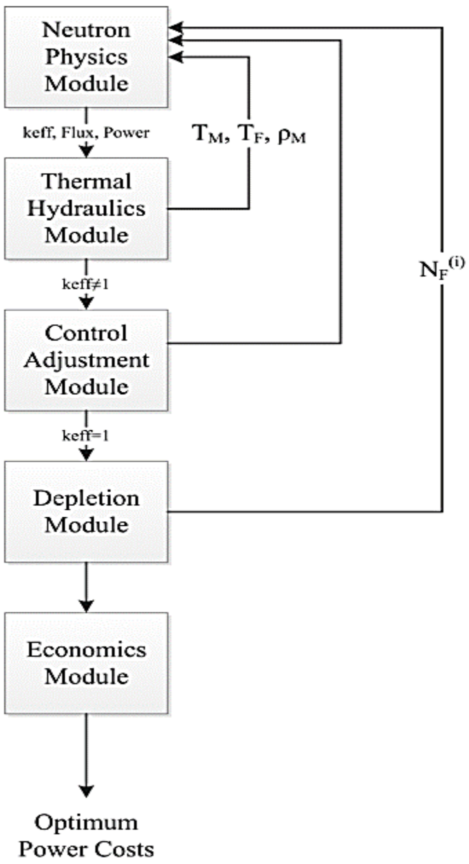


Fig. 2 Nuclear reactor design flow chart. Reproduced with the permission from [12], Copyright 1991 John Wiley and Sons.

Reactor physics analyses use mathematical and computational methods to study the behavior of nuclear reactors. These analyses aim to understand the underlying physical processes in a reactor, such as neutron transport and interactions with fuel, cladding, coolant/moderator, and control materials. They also focus on the heat deposition from fission reactions in the core, the effect of neutron interactions on fluid flow and coolant/moderator properties, and optimizing reactor design and operation. The goal is to improve reactor efficiency by maximizing fuel residence time in the core and achieving higher specific power output. All materials (nuclides and elements) are characterized by their nuclear reaction cross-section. Different materials will behave differently when placed in the reactor core from a neutronic standpoint.[13] The replacement of stainless steel with hafnium-separated zirconium was made due to a relatively smaller absorption cross-section.[14] So now, when the nuclear industry is working on replacing the Zirconium-based cladding, the decision is heavily influenced by the neutronic properties of the candidate materials.[15,16] This work aims to review/summarise the research efforts carried out until now in studying ATF's neutron physics aspects. Fig. 3 gives an overall view of the options being studied under ATF.

Reactor physics analyses are essential for designing and evaluating the safety of nuclear power plants, exploring new

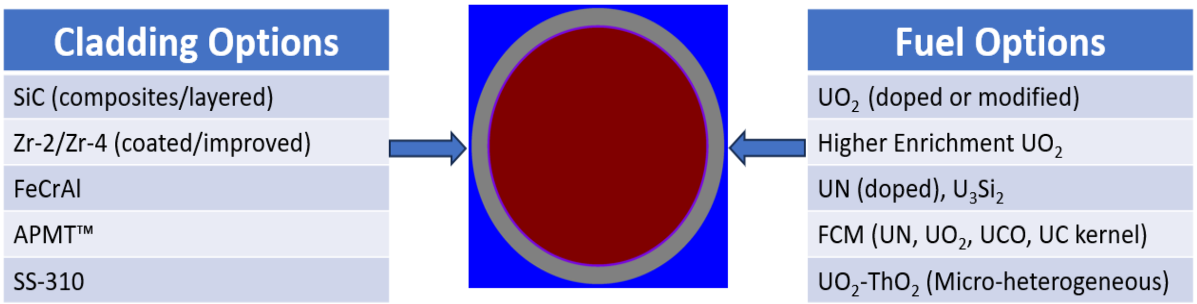
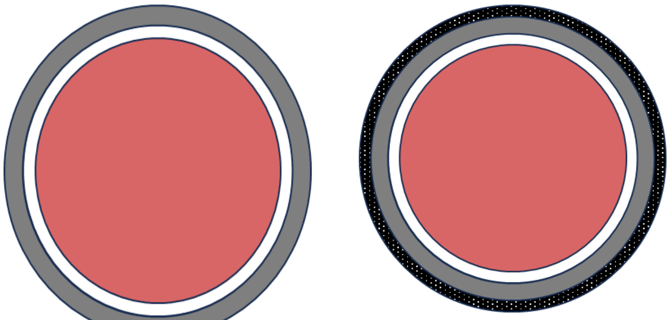


Fig. 3 Cladding and Fuel options being studied for ATF.

reactor concepts, and assessing nuclear waste and disposal options. In the context of reactor physics analyses, particular attention is paid to investigating the effects of specific materials on core criticality and cycle length, which are directly relevant to the current topic. Reactor physicists and fuel managers are especially interested in how implementing ATF concepts will affect criticality, cycle length, neutron energy spectrum, inherent safety (characterized by reactivity coefficients), and other related factors. This section will summarize the findings of studies on the reactor physics aspects of ATF. For better organization, information is divided into relevant subsections.

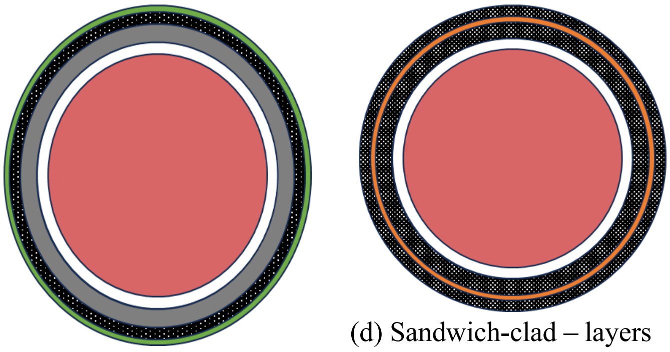
2. ATF Fuel/Clad Systems

The search for new cladding material for nuclear reactors is a lengthy endeavor, which may span over two decades.[17] This protracted timeline encompasses the novel material's development, qualification, and rigorous testing. In the aftermath of the Fukushima accident, the development of advanced cladding materials has gained momentum. The primary objective of this pursuit is to postpone the onset and diminish the extent of Zr-alloy oxidation heat, thereby alleviating the strain on the Emergency Core Cooling System (ECCS). Various cladding material concepts are under consideration, including the coating of existing Zr-based cladding and the development of entirely new materials to replace the current cladding with alternatives such as FeCrAl, SS304, and SiC. To supplant the Zr-based cladding employed in modern PWRs, candidate materials must demonstrate comparable or superior performance in terms of safety and economics. From a neutron physics and economic standpoint, the proposed materials should possess a smaller neutron absorption cross-section than Zircaloy. This characteristic would ensure no penalty in terms of the need for an increase in uranium-235 enrichment. Additionally, candidate cladding should be capable of withstanding a prolonged residence time in the reactor core. Another crucial requirement is the ability of the proposed material to facilitate power uprates in typical LWRs, aided by enhanced thermal conductivity. Fig. 4 illustrates schematics of various SiC cladding concepts: monolithic, duplex, triplex, and sandwich.



(a) Monolithic-clad – layers are fuel, gap, and SiC clad

(b) Duplex-clad – layers are fuel, gap, SiC clad, and SiC/SiC Fiber



(c) Triplex-clad – layers are fuel, gap, SiC clad, SiC/SiC fiber and coating

(d) Sandwich-clad – layers are fuel, gap, and metallic liner sandwiched between SiC/SiC fiber layers

Fig. 4 Different SiC cladding concepts.

3. LWR Systems

An economic study was conducted to assess the impact of iron-alloy-clad fuel bundles on electricity production costs based on their neutronic characteristics.[18] A single-fuel pin model based on a PWR fuel bundle was employed to analyze burnup using various cladding materials: zircaloy, FeCrAl, and SS310. The TRITON module from the SCALE 6.1 package, utilizing the ENDF/B-VII cross-section library (238 energy group structure), was employed for the analysis. The FeCrAl[19] cladding exhibited a smaller penalty than the SS-310 cladding due to the absence of nickel, which has a thermal absorption cross-section approximately twice that of iron.

Overall, the penalty associated with iron-based alloys is substantial due to their higher absorption cross-section compared to Zircaloy, reducing cycle length. To mitigate this penalty, options such as increased enrichment, decreased clad thickness (owing to a higher strength than Zircaloy), and increased HM mass in the core can be explored. Maintaining the geometry of the fuel rod and fuel assembly, the decreased cladding thickness and increased HM options can be implemented simultaneously. This is achieved by reducing the cladding thickness for a constant gap thickness and occupying the now available space by increasing the dimensions of the fuel pellet. In the past, another approach involved increasing the fuel rod's active height, which this study did not cover. A decreased cladding thickness was analyzed to increase the core HM loading by approximately 13% compared to the reference case. For the conservative case of using the same thickness as Zircaloy, an enrichment increase of less than 1% was calculated based on the EOC multiplication factor value for iron-alloy-based cladding materials. The economic analysis of the iron-alloy-based claddings predicted a 4-10% increase in the total electricity production cost.

A more comprehensive study was conducted by George *et al.* to examine the neutronic behavior of SS304, SS310, FeCrAl, APMT alloys, and SiC-based materials.[20] A straightforward comparison of thermal neutron absorption cross-sections revealed that only SiC (0.086 barns) could provide superior neutron economy compared to Zircaloy (0.20 barns). All other investigated materials exhibited a considerably higher microscopic thermal absorption cross-section than the reference cladding material, in addition to their higher physical density. For materials with higher absorption cross-sections, increased fuel pellet radius by decreasing clad thickness or increased enrichment was explored. The SCALE 6.1 code system was utilized to perform the studies with the ENDF/B-VII.0 cross-section library, which has a 238-group structure. Using an analytical approach, single-pin depletion studies were carried out to approximate a multi-batch core loading configuration. Burnup studies for iron-based alloys, including SS304, SS310, FeCrAl, and APMT, revealed a substantial decrease in EFPDs. The variation in multiplication factor diminished with increasing burnup due to a harder spectrum of the candidate iron-based cladding materials and, consequently, increased production of Pu-239. Maintaining the reference geometry, reactivity penalties for the iron-based alloys are approximately 4%. Significantly thinner cladding or an increase of roughly 1.6% enrichment is required to enhance cycle reactivity using iron-based alloys. Conversely, SiC produced a slight positive difference in the infinite multiplication factor. SS310 and FeCrAl exhibited slightly more negative Moderator Temperature Coefficient (MTC), while SiC showed slightly less negative values than the reference, demonstrating a decreasing trend with increasing burnup. Economic analysis revealed that SS310 and FeCrAl would lead to a 15-36% increase in fuel pellet cost, while SiC clad-based fuel cost would remain the same or even slightly lower.

A screening study was conducted by Todosow *et al.* to evaluate the feasibility of replacing the UO_2/Zircaloy system with advanced fuel and cladding.[21] This investigation considered UN (100% N-15) composite fuels (80-10% volume fraction) with a secondary phase (U_3Si_5, U_3Si_2, UB_2, and UB_4). Along with the Zircaloy reference case, SS, APMT, and FeCrAl claddings were also examined. The SCALE package served as the computational reactor physics tool. For SS304, thinner cladding with increased diameter fuel could lead to a substantial net increase in cycle length. Even using the composite fuel with Zircaloy cladding demonstrated a significant increase in cycle length. The study revealed that the soluble boron coefficient becomes less negative than the reference fuel-clad system. UN-U_3Si_2 and UN-U_3Si_2-UB_4 with Zircaloy were investigated in a 3D equilibrium core configuration using the PARCS code system. The addition of UB_4 offers the additional advantage of acting as a burnable absorber, potentially replacing the IFBA-coated rod configuration employed in the reference case. It can also eliminate the need for B-10 enrichment, as in the IFBA concept. A comparison of the two fuel options with the reference case demonstrated superior performance for composite fuel in terms of radial assembly peaking factors. Due to the higher thermal conductivity of composite fuel, a significant reduction in fuel temperature is also achieved at EOC for equilibrium core configuration.

Brown *et al*.[22] conducted an extended version of this study. A comparison of the absorption rate per unit lethargy of constituent materials of ATF cladding materials revealed that molybdenum exhibits the highest parasitic absorption. In contrast, aluminum exhibits the least, as illustrated in Fig. 5. Detailed transmutation studies of APMT cladding with UO_2 fuel demonstrated hydrogen production, which could affect cladding properties. MTC and Fuel Temperature Coefficient (FTC) were investigated for the composite fuel UN-U_3Si_5 with iron-based cladding, revealing a relatively more negative value than the reference case. The effectiveness of Ag-In-Cd control rods was observed to decrease. A variation in phase content of the UN (50-80%) and secondary U_3Si_5 (40-10%) phase was studied. The APMT cladding requires a minimum of 80 vol% of UN to achieve comparable EFPDs to the reference case. Increasing the vol% of UN renders the soluble

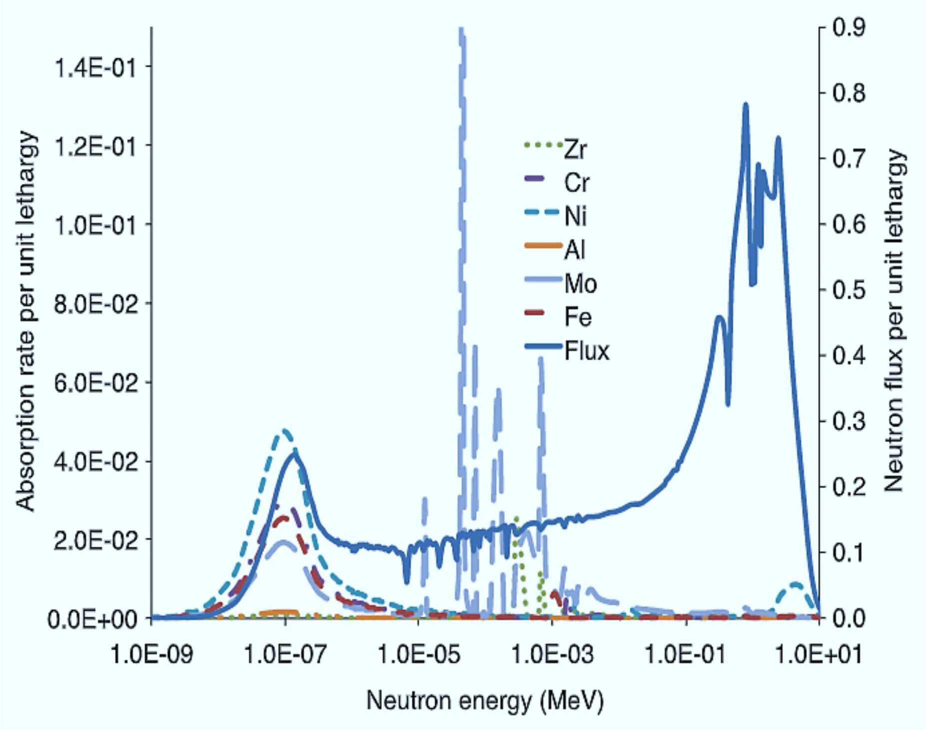


Fig. 5 Comparison of ATF cladding materials absorption. Reproduced with the permission from [22], Copyright 2015 Elsevier B.V.

boron less effective. Regarding fuel utilization, UN-U_3Si_5-UB_2 with APMT cladding performs very similarly to the reference case. All other studied combinations compromise discharge burnup. To prevent hydrogen generation in the fuel via the N-14(n,p) reaction and parasitic absorption of neutrons, nitrogen enriched in N-15 is recommended. A significant penalty of >100 EFPDs was observed without N-15 enrichment. The critical boron concentration required to control excess reactivity for composite fuel with APMT cladding is also higher than the reference fuel-clad system.

FeCrAl cladding offers superior oxidation resistance compared to Zircaloy but significantly diminishes neutron economy. Researchers have explored various strategies to mitigate this potential drawback. Studies have employed thinner cladding for FeCrAl material due to its enhanced strength and a slight increase in enrichment. Alternatively, higher HM density fuels, such as U_3Si_2, can be utilized with FeCrAl clad to counter the parasitic neutron absorption.[23] TRITON and KENO-VI from the SCALE 6.1 package and RMC were employed to perform the analysis using ENDF/B-VII.0. The Westinghouse fuel assembly served as the reference calculation geometry. U_3Si_2-FeCrAl systems with varying clad thickness values were analyzed. The initial multiplication factor for all U_3Si_2-FeCrAl cases is lower than the reference calculation. However, the cycle length in terms of EFPDs for thinner cladding cases is higher due to increased plutonium production. The analysis also demonstrated that thinner cladding cases could provide the same cycle length while using reduced enrichment. The reactivity coefficients, *i.e.*, MTC, FTC, and Void Reactivity Coefficient (VRC), for U_3Si_2-FeCrAl, exhibited values similar to those of the reference case. The reference system control rods were less effective in the U_3Si_2-FeCrAl system. Compared to reference fuel pellets, a significantly larger self-shielding effect is observed in the U_3Si_2 pellets.

SiC cladding does not suffer from parasitic neutron absorption like iron-based claddings. Tan *et al*.[24] investigated the use of SiC cladding in a PWR environment and assessed its neutronic impact. The DRAGON code was employed to perform assembly-level analysis using the collision probability method, while the generalized Stamm'ler method was utilized to address resonance self-shielding effects. A generic PWR assembly without burnable poison (BP) was considered for the analysis. The UO_2-Zr system served as the reference case, while SiC with UO_2 and UO_2/BeO cases were explored. To compensate for the decrease in HM loading due to the presence of BeO, the U-235 enrichment was increased compared to the reference and UO_2-SiC cases. BeO inclusion aims to enhance the thermal conductivity of monolithic UO_2 fuel, as SiC has only 33% of the thermal conductivity of irradiated Zircaloy. Due to strength limitations, the researchers suggested a slightly thicker SiC cladding, which could exacerbate the temperature drop from the fuel centerline to the coolant in a UO_2 system. Consequently, only thinner SiC cladding was analyzed for UO_2 fuel, while UO_2+BeO was used for thicker cladding. At the beginning of cycle (BOC), the UO_2-SiC case exhibited a more thermal spectrum than the UO_2+BeO-SiC system. This distinction vanishes at the end of cycle (EOC), with both SiC cases displaying a very similar neutron spectrum that is softer than the reference case. The SiC cladding-based system would produce less Pu-239 due to the softer neutron spectrum. The SiC cladding system's multiplication factor is higher than the reference system's. A similar BP inclusion effect and behavior were proposed for the three studied systems. Due to a softer spectrum than Zircaloy, a flatter flux profile is observed inside the fuel pellet, reducing self-shielding. The MTC, FTC, and total temperature coefficient of reactivity exhibit similar trends and values for the studied options.

Cladding coatings have shown minimal impact on neutronic properties, as studied by Younker *et al*.[25,26] The Serpent code was employed for the studies using ENDF/B-VII cross-sections, with the reference geometry for PWR being the AP1000 fuel assembly and GE14 for BWR. The authors investigated different thickness values of various candidate coating materials, including Ti_3AlC_2, Ti_2AlC, Nb_2AlC, and TiAlN. Parameters such as FTC, MTC, and VRC were analyzed. Ti_3AlC_2 had the most significant impact on reactivity, requiring the highest additional enrichment. Nb2AlC required the smallest enrichment increase for an equal cycle length among the remaining candidate materials. Alternative claddings such as SiC, FeCrAl, Titanium-Zirconium-Molybdenum alloy (TZM), and Alloy 33 were also explored for PWR and BWR environments. Iron-based materials exhibited a substantial penalty in EFPDs due to their higher neutron absorption, necessitating a significantly reduced cladding thickness or increased U-235 enrichment to avoid the reactivity penalty. Conversely, SiC can have a small but positive effect on cycle length and exhibits values similar to reactivity coefficients in the reference system. TZM and Alloy 33 demonstrate more negative values for the studied reactivity coefficients.

As seen earlier,[24] adding beryllium oxide (BeO) to uranium dioxide (UO_2) fuel can improve its thermal conductivity. Studies[27,28] have shown that adding 5-10% BeO to UO_2 fuel has a small effect on neutron physics characteristics, causing a 2-3 EFPD change. BeO also softens the neutron spectrum, which increases the control rod's worth. However, BeO has a higher (n, α) cross-section than UO_2, which leads to higher alpha and helium production. Studies comparing different ATF cladding combinations with UO_2-BeO fuel showed that FeCrAl cladding has a higher neutronic penalty than SiC cladding. These analyses used the RMC code with the ENDF/B-VII.0 library for a typical PWR fuel assembly.

Mustafa[29] conducted a study to investigate the neutronic properties of various cladding materials (SiC, FeCrAl, and SS310) in an advanced PWR assembly using the MCNPX code[30] and ENDF/B-VII.1 data. Initial investigations, using both MCNPX and WIMSD-5[31] codes, were performed for pin-cell geometry, while fuel assembly calculations were solely conducted using MCNPX. The findings align, indicating that SiC cladding can extend cycle length and discharge burnup compared to the reference case. Conversely, FeCrAl and SS310 claddings lead to reduced cycle length and discharge burnup due to their larger absorption cross-sections. Iron-based claddings exhibited increased Pu-239 production, which was attributed to the harder neutron spectrum associated with them, caused by increased absorption of thermal neutrons in the cladding. Additionally, a slight reduction in enrichment could achieve the same burnup value with SiC cladding due to its favorable neutron economy. At BOL, FTC, MTC, and VRC for SiC are slightly less negative than the reference and iron-based claddings. However, this trend reverses early in the cycle when SiC shows more negative values than other options. Regarding PPF, iron-based claddings exhibit slightly higher values than SiC and Zr-based cladding, with SiC cladding showing similar PPF values compared to the reference. Thermal neutron fraction is high at BOL, dips at MOL, and then increases with increasing burnup for all studied claddings, with SiC showing the most significant value of thermal neutrons. The impact of different cladding materials on fission fragments, such as Ru-106 and Xe-135, was also examined. Ru-106 shows minimal variation with a change of cladding material. On the other hand, Xe-135 shows a slight increase for iron-based claddings due to a smaller thermal neutron fraction. The study also revealed that ATF claddings would lead to a slightly smaller delayed neutron fraction than the reference case.

Rivai *et al.* explored the potential of SiC cladding for replacing Zircaloy in a typical LWR.[32] using the SRAC code system.[33] Their findings revealed a slightly more thermalized neutron energy spectrum when using SiC compared to the reference case. However, a slight burnup penalty was observed, as evidenced by a marginally smaller infinite multiplication factor for SiC compared to the reference case. SiC cladding exhibited a slightly larger variation in the neutron energy spectrum across different fuel temperatures compared to the reference case. These calculations were performed for a simplified single fuel pin model where an equivalent-radius cylinder represented the square pitch.

Researchers have conducted studies on PWRs to investigate the feasibility of replacing current UO_2 fuel with composite UN fuel.[34,35] Brown *et al*.[34] evaluated UO_2, UN, and UN composite fuels in PWRs. They recommended using UN with 80% theoretical density in composite fuels to accommodate fission gases. Few-group parameters from Serpent and TRITON were compared and found to agree. The study indicated that higher U-238 content, due to the higher physical density of the fuel, would lead to increased self-shielding. Different options, such as U_3Si_5, U_3Si_2, ZrO_2, and UB_4, were examined as composite fuel matrices with UN. The safety reactivity coefficients were found to be within an acceptable range. Equilibrium core studies also demonstrated satisfactory performance. The issue of higher excess reactivity at BoC, which necessitates higher Critical Boron Concentration (CBC), is addressed using enriched soluble boron with nitride/composite fuels.

UN and UC fuels pose the challenge of reaction with water.[36] To address this issue, composite fuels employ an oxidation layer to prevent accidental interactions between the fuel material and water. Another solution, similar to FCM, involves coating fuel pellets with layers of suitable materials to create additional barriers against water-fuel reactions. To compare the neutron characteristics of coated and composite UN fuel in PWRs,[37] the BEAVRS[37] core served as the system of interest. OpenMC code was employed to conduct the analysis. N-15 enrichment was necessary to mitigate the neutronic penalty caused by parasitic absorption in N-14. Theoretical densities of 80% and 95% were considered two bounds for UN fuels. Higher theoretical density fuels offer the advantage of longer cycle lengths, but increased HM content and subsequent spectral hardening make control measures slightly less effective. Cases with 80% theoretical density exhibit similar or somewhat worse performance than the reference case regarding EFPDs.

In the context of BWR applications, studies have been conducted to investigate the behavior of accident-tolerant SiC/SiC channel boxes[38] using the Virtual Environment for Reactor Applications (VERA)[39] developed by the Consortium for Advanced Simulation of Light Water Reactors (CASL). Neutronics results obtained from VERA using MPACT are compared with those from PARCS/PATHS and Serpent codes, demonstrating good agreement. SiC/SiC refers to a silicon carbide fiber-reinforced silicon carbide matrix composite employed as the material for fabricating BWR channel boxes. Multiphysics studies were carried out to assess the three-dimensional temperature and flux distributions within the water box. These studies revealed that potential distortions could arise due to fast flux and temperature gradients along the radial and axial directions of the channel box. Channel boxes closest to control blades are predominantly susceptible to such gradients, leading to non-uniform irradiation swelling rates. This underscores the importance of developing high-fidelity tools for design and safety analysis.

High-density fuel materials offer the possibility of extending the residence time of fuel within the reactor core.[40] In the case of a BWR, replacing conventional UO_2 fuel with nitride fuel resulted in a 40% cycle length extension. However, this came at the expense of a harder neutron spectrum, leading to decreased BP performance. Additionally, nitride fuel slightly decreased total void worth and DRC. Due to the significant parasitic absorption associated with N-14, enrichment with N-15 is preferred. UN doped with 10% ZrO_2 demonstrated a comparable U-235 burnup to the reference fuel.

4. APR-1400

The United Arab Emirates (UAE), a newcomer to nuclear power, has four APR-1400 reactor units. A study evaluated the neutronic impact of alternative cladding materials in APR-1400 fuel assemblies.[41] The Serpent code was employed for the analysis. Due to their higher neutron absorption cross-sections, iron-based cladding materials introduced a neutronic penalty, reducing cycle lengths. These clad materials also caused a decrease in thermal neutron flux due to their larger neutron absorption cross-sections. The hardening neutron spectrum led to a higher production of Pu-239 compared to the reference case. Additionally, an increase in the amount of hydrogen and helium produced in the cladding materials was observed, attributed to the presence of nickel, compared to the reference case. SiC cladding exhibited neutronic performance that was very similar to the reference case. Linear heat generation rates (W/cm) were also analyzed for the cladding materials, and the results showed comparable performance to that of the reference case.

To achieve the cycle length of the current APR-1400 system, a sensitivity study was conducted to evaluate the impact of various design parameters, including fuel enrichment, cladding thickness, and pellet diameter.[42] Due to its superior strength characteristics, FeCrAl cladding thickness can be reduced to offset the neutronic penalty caused by its higher absorption cross-section. Conversely, SiC cladding requires a slightly larger thickness for safe operation. Similarly, enrichment variations were also investigated to obtain an equivalent cycle length to the reference system. Different combinations of these parameters were explored in various fuel assemblies. It was observed that a reduced thickness and slightly increased enrichment in FeCrAl cladding could achieve a cycle length equal to the reference system. For SiC, even with increased clad thickness while maintaining the outer clad radius constant, the pellet radius and enrichment can be decreased to obtain the reference cycle length. Overall spectral hardening is observed with FeCrAl, while the SiC option provides a similar or slightly more thermal neutron spectrum than the reference case. This leads to larger and smaller Pu-239 production in FeCrAl and SiC, respectively.

Another concept used for protecting Zircaloy from oxidation is to use swaging technology to apply thin tubes of SS316 on the inside and the outside of the cladding. A thickness value of 10 μm is considered for the inside tube, while a 30 μm thickness value is used for the outside SS316 tube. The problem of poor neutron economy hampers iron-based claddings due to their higher neutron absorption cross-section. Using the swaging technique can help reduce the neutron economy penalty and offer the benefit of accelerated induction in nuclear power plants as an ATF option. Khalefih *et al*.[43] investigated the neutron physics impact of using this concept of ATF cladding for APR1400. The Monte Carlo-based code Serpent[44] and in-house nodal code KANT[45] were used to perform the analysis. Using a single fuel assembly model, assessment for the penalty of using iron-based (SS316) tubes showed an impact of -700 pcm at BOL, which decreased to a value of -400 pcm at EOL. Aside from coating Zircaloy cladding, this technique is also viable for low neutron physics impact.

A neutron physics comparison of SiC and FeCrAl claddings was conducted for APR-1400 reactors.[46] SiC cladding offers the advantage of a lower neutron absorption cross-section compared to the reference Zr-alloy cladding. The Serpent code (version 2.31) was employed to perform the analysis. Pin-cell calculations with Zr-alloy (reference), SiC, and FeCrAl demonstrated that SiC can slightly outperform the reference system. Conversely, FeCrAl exhibited a significant reactivity penalty due to the higher absorption cross-sections of its constituents. SiC exhibits a neutron spectrum similar to the reference system, while FeCrAl shows spectral hardening, leading to higher Pu-239 production. When assembly-level calculations were performed, this Pu-239 was observed to offset some of the negative impact of the higher absorption cross-section in the longer run. 2D full-core calculations also showed similar trends, *i.e.*, SiC showed better performance than the reference system, while FeCrAl showed a severe neutron penalty.

The neutronic performance of SiC "sandwich" cladding was evaluated for APR-1400 reactors under normal operating conditions.[47] The term "sandwich" refers to a Niobium or Tantalum liner placed between two composite layers of SiC to ensure leak tightness. The Serpent code was used with the ENDF-B/VII.1 library to conduct the study. Pin, assembly, and core-level calculations were performed. Pin and assembly-level calculations revealed that Niobium outperformed Tantalum in neutronic performance due to Tantalum's significantly higher neutron absorption cross-section. Tantalum would lead to a much harder neutron spectrum, resulting in higher Pu-239 production, similar to Iron-based cladding materials. SiC sandwich cladding exhibited performance comparable to the reference case of Zr-alloy. Based on these findings, core calculations were only performed with Nb liner, and the results were similar to those of the reference core. As a result, it was concluded that SiC sandwich cladding with Nb liner could be a viable neutronic alternative to current Zr-alloy cladding.

Considering the aspects of technological readiness and ease of implementation in existing nuclear reactors, Cr-coated Zr-cladding was investigated for its impact on the neutronic characteristics of the APR-1400 nuclear reactor.[48] The Serpent code, using the ENDF/B-VII.1 library, was employed to conduct assembly-level and 2D whole-core studies. The eight different types of fuel assemblies present in the APR-1400 core were analyzed for various coating thickness values. As anticipated, increasing the coating thickness resulted in a more significant reactivity penalty for the examined fuel assemblies. Pu-239 quantity, burnup, and other neutronic parameters, such as neutron flux and PPF, exhibited similar trends for the investigated cases. The deviation from the reference case intensified only with increasing coating thickness. Whole-core 2D analyses also demonstrated the same behavior, with a reactivity difference as slight as approximately 200 pcm for a 10 μm Cr-coating. This indicates that such coated cladding options can be incorporated into the current reactor design without causing any substantial alterations to neutron physics behavior.

Alrwashdeh *et al*.[49] investigated the possibility of using higher enrichment fuel to extend the cycle length of the APR-1400 reactor from the current 18 months to 24 months. They examined homogeneous (single enrichment per assembly) and heterogeneous (two enrichments per assembly) fuel arrangements. To control excess reactivity at BOL, the researchers investigated the use of IFBA coatings on the surface of fuel pellets. The Serpent code was used to evaluate the neutronic performance of increased enrichment cases in conjunction with ATF materials such as FeCrAl, APMT, SS304, SS310, and SiC. The iron-based cladding option showed a neutronic penalty, making SiC the most suitable choice for replacing Zr-alloy cladding. FTC and MTC were examined and found within safe limits throughout the cycle. IFBA coating or SiC did not affect the PPF compared to the reference design.

5. SMR and other reactors

The neutronic properties of ATFs have been investigated for potential use in CANDU reactors.[50] Various fuel concepts were analyzed, including composite fuels (UO_2-SiC, UN/U_3Si_2), UN fuel, (U, Zr)N, U-9Mo, and UO_2/UN fuel kernels embedded in a SiC matrix or as bare kernels within the SiC matrix, along with ATF cladding options. The study's primary goal was to identify a combination that would enable the continued use of natural uranium, as is currently practiced in CANDU reactors. The Serpent code, based on the Monte Carlo method, was employed for the analysis. Doppler Broadening Rejection Correction (DBRC) was applied for U-238 in the energy range of 0.4 to 210 eV. The cross-section library was based on ENDF/B-VII.0. Burnup calculations were performed for four regions, one for each ring of the 37-element fuel bundle. The impact of varying additives and combinations is studied for different fuel, clad, and additive materials. Cladding materials like SiC, Zr/FeCrAl, and Zr-4 were studied with various fuel materials for varying burnup values. For FCM fuel, UO_2 and UN kernels in the form of TRISO particles embedded in SiC were examined for different kernel diameters and packing fraction values. Neutronic studies were also conducted for two-layer cladding, consisting of 80% nominal thickness Zircaloy with 20% thickness of FeCrAl. Among the studied combinations, only UO_2 with SiC cladding and UN (99.5% enriched in N-15) options allowed using natural uranium. Increasing the packing fraction can reduce the enrichment requirement to avoid a burnup penalty, although the enrichment value will still be higher than natural enrichment. All other studied options required some uranium enrichment to prevent the burnup penalty and maintain reactor criticality. The improved thermal conductivity of the studied fuel options can lead to lower fuel temperatures. It was observed that adding 10% SiC to UO_2 fuel can significantly reduce fuel temperature. The coolant void reactivity (CVR) is more significant for higher uranium-density fuel materials. Conversely, using additives like SiC to UO2 will decrease the HM density, resulting in a smaller CVR value.

Building upon earlier studies on long-life civil marine reactors,[51–53] Alam *et al*.[54] conducted neutronic analyses to investigate the use of ATF cladding in this type of reactor. They employed the concept of "micro-heterogeneous duplex fuel," utilizing uranium-thorium oxide fuel in a heterogeneous arrangement for assembly-level analysis. A 13 × 13 array in a soluble-boron-free configuration was considered. Various ATF cladding materials, including SiC, FeCrAl, APMT, and SS310, were studied as potential replacements for Zircaloy. The UO_2/Zr system served as the reference case. WIMS-10[55], equipped with the JEFF 2.2[56] library, performed assembly-level calculations for reactivity, burnup, spectrum, and reactivity feedback. SiC emerged as a suitable candidate due to its lower capture cross-section than Zircaloy and other candidate materials. This property potentially allows for a reduction in fuel enrichment or an extension of the cycle length. SiC and Zircaloy exhibited relatively higher fission rates in the outer region of the pellet, attributed to their better neutron economy resulting from their lower neutron absorption cross-sections. The APMT and SS310 claddings displayed more negative MTC and FTC values than the other studied cladding materials due to molybdenum.

Investigations on the effects of accident-tolerant fuel (ATF) cladding on the neutronic performance and parameters of the Autonomous Transportable on-demand reactor module (ATOM)[57] core were carried out by Nguyen *et al*.[58] The ATOM core is designed to be soluble boron free (SBF). Serpent 2, based on the Monte Carlo method and the in-house code COREDAX,[59] was employed for the analyses. The ENDF/B-VII.1 library was used with the Serpent code. Scoping analyses were performed using reflective boundary conditions for a single assembly. Two different ATF cladding concepts were studied, namely Cr/Cr alloy-coated Zircaloy-4 and SS304/FeCrAl cladding. Even with reduced cladding thickness for SS304 and FeCrAl cladding, a significant reactivity penalty was observed. An enrichment value greater than 5% is required to achieve the same cycle length as the reference core. The 30 µm Cr-coated Zr-4 exhibited very similar reactivity to that of Zr-4. PPF was also analyzed for the studied cases, and a monotonic decrease from BOL to EOL was observed, with very similar performance for Cr-coated and reference core systems. The SS304 and FeCrAl showed slightly higher PPF values than the reference case. The neutron spectrum for SS304 and FeCrAl showed a visible hardening, while the Cr-coated neutron spectrum was almost the same as that of the reference case. Based on the neutronic results and relatively higher technology readiness, Cr-coated (Cr15Al-coated Zr-4) was selected for whole core calculations. The self-shielding effect was also studied using pin-cell calculations at both BOL and EOL. The coated cladding showed very similar behavior to the reference system. The core burnup reactivity swing with ATF cladding was smaller than that of the reference case. There is a small negative impact on the cycle length when using Cr-coated cladding. The axial power peaking factor didn't show any change.

Due to the diverse designs of nuclear reactors employed worldwide, a wealth of information exists regarding the applicability of the ATF concept in reactor designs. For the VVER-1000 reactor, Cr-coated Zr-alloy, FeCrAl, and SiC claddings were investigated as replacements for conventional

Zircaloy cladding.[60] A comprehensive core analysis was performed using DRAGON[61] and PARCS codes. The results revealed that iron-based claddings necessitate increased enrichment to achieve the same cycle length. In contrast, SiC cladding leads to a longer cycle length even with identical geometry and enrichment. The FeCrAl cladding exhibited a decrease in integral control rod worth compared to other cladding materials. The PPF values were similar for the studied cladding materials. SiC and FeCrAl claddings resulted in a less negative MTC than the reference case and Cr-coated claddings. The FTC and boric acid reactivity coefficient showed similar performance for the studied materials. The CBC requirement for the SiC cladding with identical geometry and enrichment was higher than the reference case.

A summary of the main findings from the ATF fuel/clad system is given here.

- Composite fuels can lead to longer cycle lengths but may lead to relatively smaller utilization of HM as compared to UO_2.
- While iron-based claddings allow for thinner claddings due to their superior strength, this benefit may not fully offset the need for increased uranium enrichment to achieve the same cycle length.
- Silicon carbide cladding may need to be slightly thicker than the current Zr-based cladding, but it offers similar or marginally better neutronic performance.
- Different coatings are being analyzed to protect the Zr-alloy + steam reaction. They provide a medium-term solution for ATF. Some cladding materials can provide similar neutronic performance as the reference fuel/clad system.
- Swaging technology for using thin tubes inside and outside of normal Zr-based claddings is also being studied and provides a viable/comparable neutronic solution.

6. Fully Ceramic Microencapsulated (FCM) fuel concept

Water-based thermal nuclear reactors worldwide use uranium dioxide (UO_2) fuel in the form of monolithic blocks. Enriched UO_2 powder is sintered into ceramic pellets. These pellets are then loaded into Zirconium-based clad. Utilities and regulators have high confidence in this fuel option due to the fabrication and operational experience gained over the past 6-7 decades. However, significant changes have been proposed after the Fukushima accident to avoid BDBAs. Major UO2/Zr system changes are being considered to prevent hydrogen production. Improved fuel design can lead to economic and safety benefits. Areas identified to achieve these goals include nuclear fuel composition, cladding integrity, and reducing pellet-clad interaction (PCI) to allow for increased linear heat generation rate (LHGR) and fuel burn-up.

Fully Ceramic Microencapsulated (FCM) fuel is a new concept that was first proposed in 2012.[62] It consists of Tri-Structural Isotropic (TRISO) microspheres embedded in a compact matrix of mostly Silicon Carbide (SiC) and shaped like a fuel pellet. The TRISO fuel concept dates back to the 1960s and was initially used with uranium dioxide (UO_2) fuel kernels in High-Temperature Gas Cooled Reactors (HTGRs). UO_2 kernel is still being explored as a design option in some concepts like prismatic high-temperature reactors.[63,64] In 2002, the Department of Energy (DOE) began considering different fuel kernel materials, such as uranium oxycarbide (UCO).[65] TRISO-embedded FCM pellets are placed in a clad similar to UO_2 fuel pellets. The main difference between FCM fuel and the coated TRISO fuel concept envisioned for Next Generation Nuclear Plants (NGNPs) is the size and shape of the ceramic matrix in which the TRISO microspheres are placed.[66] The FCM fuel concept is also known as particle-based accident-tolerant (PBAT) fuel in Korea and inert matrix dispersion pellets (IMDP) fuel in China.[67] It was developed based on the experience of using TRISO fuel in HTGRs and offers the benefits of higher burnups, defense in depth by providing additional barriers against radioactivity release, and better safety margins in normal and accidental conditions. FCM fuel is considered a promising medium-term concept to replace current UO_2 fuel pellets.

The potential advantages offered by FCM fuel in an LWR[62] include:

- Improved thermal properties: The FCM fuel form can reduce peak fuel temperatures compared to the current UO_2 fuel.
- Enhanced proliferation resistance: The FCM fuel form makes extracting plutonium from spent fuel more difficult, strengthening non-proliferation efforts.
- Better fission product retention: The suspension of particles within a ceramic or metal alloy matrix creates additional barriers to releasing fission products during normal operation and even in postulated accidents.
- Higher fuel burn-up: Reduced fuel-cladding interaction and fuel pellet swelling allow for higher fuel burn-up, extending the lifespan of the fuel.
- Excellent mechanical stability: The FCM fuel form exhibits high mechanical stability, ensuring its integrity under various operating conditions.

Some shortcomings of using the FCM fuel[68]:

- Smaller heavy metal loading demands much higher enrichment (>5% and <20%) to obtain the same cycle length as normal uranium dioxide LEU fuel, impacting the economics and resource utilization.

- MTC and FTC are less during the cycle's first half than the conventional UO_2 fuel.
- Due to the heterogeneous arrangement of FCM fuel with UO_2 fuel, power peaking factors can be higher which needs to be controlled using burnable poison re-configuration.

Leveraging HTGR fuel manufacturing technology: The FCM fuel form can utilize existing HTGR fuel manufacturing technology, facilitating its adoption and deployment.

It is necessary to adjust the fuel and core design accordingly to counterbalance the reduced density of fissile material due to the inclusion of extra barriers in the fuel. To increase the fissile loading and avoid the cycle length (EFPDs) penalty, the combination of uranium enrichment up to the practical upper limit of LEU (< 20% of U-235), utilization of high-density fuel kernel (UN, UC, UCO, *etc*.), increasing kernel-to-particle volume fraction, higher TRISO-packing fraction, and enlarging fuel pin diameter are the proposed solutions.

In the United States, this fuel concept has been a topic of active R&D, since the early 2010s, focused on fabrication[69] and fabrication.[62,70,71] It is well-established that the neutronic aspect directly impacts the economics of the whole power plant operation.[9] Basic scoping studies can utilize the results from very simple single-pin-cell calculations to draw basic conclusions about the economic viability of the concept.[62] Studies have been performed to integrate the FCM with the current LWR fleet on assemblies containing both UO_2 fuel rods and FCM rods.[69,72,73] The FCM option has been analyzed in three distinct scenarios for LWR.

1. burning of TRU in LWRs
2. the accident-tolerant capability of FCM
3. FCM in the small modular reactor (SMR) concept.

7. TRU burning in LWRS

Two options studied for LWR to recycle plutonium in LWRs are Combined NonFertile and UO_2 (CONFU) and CORAIL mixed oxide (MOX) concepts.[74,75] The CONFU concept uses TRU in an inert matrix to make fuel pins. These TRU-bearing fuel pins are loaded in conjunction with standard UO_2 fuel pins.[74] The second concept is CORAIL which mixes MOX fuel pins with standard UO_2 fuel pins.[75] in the same assembly. These studies[69,72,73] utilize the CORAIL concept. CE System 80 fuel bundles and Westinghouse fuel bundles were studied in which FCM fuel was used for the destruction of transuranic (TRU) obtained from spent fuel.[69] Monte Carlo-based McCARD[76] with ENDF-B/VII[77] continuous energy library was used to perform the analysis. Snead *et al*.[69] explored the sensitivity of kernel diameter, packing fraction, and uranium enrichment for UO_2 fuel pins in assembly-level calculations. As expected, increasing the uranium enrichment gave longer cycle lengths (EFPDs). For a three-batch core, the authors reported a discharge burnup value of 732 MWD/kgHM for TRU fuel compared to a value of ~77 MWD/kgHM for the co-residing UO_2 fuel. Local power peaking was observed in different fuels, and due to different depletion rates, Gadolinium-based burnable poison BISO particles are suggested to counter this. The study also pointed out the need for more comprehensive calculations on the full-core scale to analyze the neutronic safety issue that can arise due to the absence of U-238.

Considering the better structural integrity and reduced amount of fissile material content in FCM fuel as compared to the UO_2 option, the option of recycling Transuranic waste using the current fleet of PWRs is carried out by Gentry *et al*..[72]. Different assembly layout configurations are tested and core loading patterns are analysed to study the feasibility of Transuranic-based FCM in PWRs. Westinghouse core was used to perform the analysis. The study aimed to obtain a balance (production = destruction) of transuranic isotopes at EOC. SCALE 6.1 is used for 2D transport lattice physics calculations along with depletion studies. 3D calculations are carried out using NESTLE[78] code based on nodal diffusion theory using the SCALE 6.1 code system. The single-pin, quarter assembly, and quarter core calculations employ the RPT method, benchmarked against deterministic and Monte Carlo (KENO)[79] double heterogeneity models. After assessing the validity of the RPT modeling for the case at hand, the lattice-based comparison is made for the reference case against the proposed assembly (FCM-based rods at the periphery). Both instances show very comparable behavior, in the presence and absence of a burnable absorber, with slight differences showing near the EOC. A net balance for transuranic is achievable. Core calculations use two different loading patterns producing radial power peaking factors > 1.4. The core level comparison showed that the net balance of transuranic is obtained.

An extended form of the work by Gentry *et al*.[72] carried out additional studies for lattice and an equilibrium core analysis [73] Lattice studies showed that FCM rods could provide better fuel utilization than UO_2 fuel rods. The studies concluded a net destruction of Np and Pu isotopes, with net production of Am and Cm in some of the designs. This is a potential concern for decay heat. DRC and MTC show slightly less negative values as compared to the UO_2 lattice. Boron worth for both lattices showed a very comparable value at BOL but the FCM lattice showed more negative worth towards EOL. The delayed neutron fraction of FCM lattice compared to that of UO_2 was less at BOL but quite similar at

EOL. The CBC requirement for FCM fuel showed a relatively higher value as compared to the reference UO_2-based core, which will require adjustment/alteration of BP. The inlet coolant temperature perturbation studies by the NESTLE core simulator at BOC conditions showed safe behavior hence the conclusion that higher CBC can be used without the worry of rendering the MTC positive.

The transuranic-loaded FCM fuel in LWRs using open-source lattice calculation code DRAGON-4[80] was studied by Pope *et al*.[81] A comparison of TRU-only FCM fuel pin-cells is made against UO2 and MOX fuel. FCM exhibits a reduction in EFPDs for different burnup values but displays far superior fuel utilization, *i.e.* (>450 GWd/tonne). Soluble boron worth is more negative as compared to MOX but less as compared to UO_2 at BOC. With increasing burnup, however, this value becomes more negative for FCM. Coefficients of reactivity show a significant magnitude change during core life. The reason for this is the shift in neutron spectrum, which remains almost similar for UO_2 and MOX options but, becomes more thermal for FCM at EOC values. From the experience of full-core uranium-free Inert Matrix Fuel (IMF) cores,[82] it is known that resonant absorbers should be used as burnable absorbers. Erbia (Er_2O_3) is investigated with different loadings in unit-cell and assembly-level calculations. Different heterogeneous configurations are studied based on the number and pin position of FCM fuel pins alongside LEU fuel pins. Compared to UO_2 and MOX assemblies, numerical pin power peaking factor values are generally higher for FCM-based configurations. Burnup studies revealed that the obtained burnup value for the studied FCM cases is slightly smaller than UO_2 and MOX reference cases. The FCM assembly showed safe behavior displaying negative values for MTC, DRC, and VRC. Fast fluence for FCM pins in an assembly showed quite high values. It was recommended that further investigation be done to reduce fast fluence.

8. FCM as replacement option for UO_2

A combined study carried out by Oak Ridge National Laboratory (ORNL) and Korea Atomic Energy Research Institute (KAERI) was carried out to see the feasibility of FCM as a full replacement in LWRs.[70] Neutronic studies are carried out which act as input for subsequent thermal hydraulics and safety analysis. Optimized Power Reactor of 1000 MW (OPR-1000) and 1200 MW Westinghouse reactor fuel assemblies are used to perform neutronic analysis. The DeCART2D/MASTER code system was used by KOREA while the SERPENT[44] code was utilized by Westinghouse. Various cladding materials such as SS304 and SiC (Triplex and Monolithic) are studied with UN kernel FCM. DeCART[83] solves the 2D MOC equation using the HELIOS[84] library. As a function of temperature and burnup, tabular cross-sections are used in the MASTER code. MASTER[85] is based on the two-group diffusion theory using the nodal method for both steady-state and transient calculations of the 3D core. Along with studies using FCM fuel in the original geometry, analyses were carried out to suggest alternate (reduced) assembly lattices, with some geometrical/mechanical constraints. The 12 × 12 configuration and the reference 16 × 16 design for OPR-1000 are studied. Assembly-level studies are conducted on kernel diameter cladding material's sensitivity and packing fraction. Enrichment of FCM for a 12 × 12 lattice is found to be less than that used in a 16 × 16 lattice. A burnup analysis showed that FCM-based assembly's multiplication factor drops faster than solid UO2-based ones due to lower HM content despite higher U-235 enrichment. This reduced amount of U-238 in FCM renders the MTC, FTC, and chemical shim less negative. Boron worth for SiC and SS clad-based FCM shows a more pronounced change with increasing burnup as compared to reference. Owing to higher enrichment in FCM, Gadolinium oxide, and Erbium oxide were studied as burnable poison options in BISO-coated particle form to suppress the initial excess reactivity. Natural erbia performed better as it suppressed the excess reactivity for longer and showed less residual poison, owing to its smaller absorption cross-section than gadolinia. Fast (> 0.18MeV) neutron fluence was studied, and it was seen that FCM options had shown slightly higher values than that of the reference calculation. Studies presented for the Westinghouse 1200 MW concept, using SERPENT Monte Carlo code, showed similar behavior, i.e., less negative values for MTC and FTC for FCM as compared to UO_2 assemblies. Westinghouse also conducted the VRC in which FCM 13 × 13 performed better than the 17 × 17 UO_2 assembly.

One of the main challenges regarding the neutronic modeling of FCM is its double heterogeneous (DH) nature. Only a few deterministic codes can currently handle the DH nature. Developing an input deck to model each TRISO particle explicitly is highly time-consuming and computationally expensive. Although helper functions are available in codes like OpenMC[86] to carry this task, still the computational cost is prohibitively high for explicit TRISO modeling in whole core systems. Researchers have proposed various methods to cater to this effect instead of conventional volume-weighted homogenization (VWH). To correctly account for the double heterogeneity effect in the homogenization of TRISO-based fuels, the Reactivity-Equivalent Physical Transformation (RPT) method[87–89] is applied to generate the homogenized cross-sections. The essence of the technique is explained in Fig. 6. A slight variation of the method termed a modified RPT is also shown in the same figure to highlight the difference. Various studies conducted have proposed a modified shape of the RPT method, namely Ring Reactivity Physical Transformation (RRPT)[90], for TRISO particles that might contain strong neutron poisons. The RRPT method exhibited better results than the RPT method when studying the B_4C, Gd_2O_3, Er_2O_3, Ag, Hf, Dy_2O_3, Eu_2O_3, *etc*. in TRISO particles for excess reactivity management.

The assembly FCM option, *i.e*., all pins of a 17 × 17 Westinghouse AP1000 assembly are replaced with FCM fuel, was analyzed by Brown *et al*.[10] This work studied the effect of double heterogeneity modeling methodologies on homogenized cross-section, cycle length, and reactivity coefficients. SERPENT and TRITON[92] codes were used for the lattice-based studies. Different modeling options were used by the codes and compared. TRITON calculations were done using both the DH option, which uses explicit modeling of FCM particles to carry out self-shielding calculations using the CENTRM module, and the RPT method.[87] SERPENT modeling was done using a repeated hexagonal lattice of TRISO particles in the ceramic matrix, a repeated square lattice of TRISO particles, and an implicit dispersion modeling option native to SERPENT code. A comparison of these modeling options matched well with each other. Different burnup zones were considered for a single assembly owing to thermal flux peaking in the guide tubes. Both codes utilized different numbers of nuclides for modeling. This is known to affect the results, as demonstrated by studies.[93,94] The higher thermal absorption cross-section of N-14 showed severe penalty for UN-kernel FCM. Isotopic enrichment to N-15 was proposed as a solution for this. Another important result deduced from the burnup studies was the application of non-linear reactivity models to quantify the cycle length for FCM-based assembly. FCM assemblies, when in a driven mode, displayed concave non-linear behavior as compared to the linear behavior exhibited by UO_2 fuel assemblies. Analysis was performed and a quadratic reactivity model was found to be more suited for the cycle length estimation with FCM fuel. Analyses revealed that even with an optimistic fuel loading, using a higher packing fraction and larger kernel diameter of high HM density fuel (UN), a reduction of 50 EFPD was witnessed. Using FCM in the same lattice as UO_2 makes the neutron spectrum softer as less amount of HM is present with more moderating materials (water + carbon of TRISO particles). Similar to the case of UO_2, soluble boron concentration plays an important role in how the MTC will behave.

The continuation study[95] based on lattice physics analyses[10] was done which carried out the whole core analysis for FCM along with Reactivity Insertion Accident (RIA) analysis. As shown in the lattice physics analyses by Brown *et al*.[10] the DRC and MTC are less negative for the FCM-based core. Hence, RIA becomes a point of concern and needs to be analyzed to demonstrate the safety of the fuel concept. PARCS code[96] is used for the whole core analysis. Material temperature-dependent properties for UN and SiC are

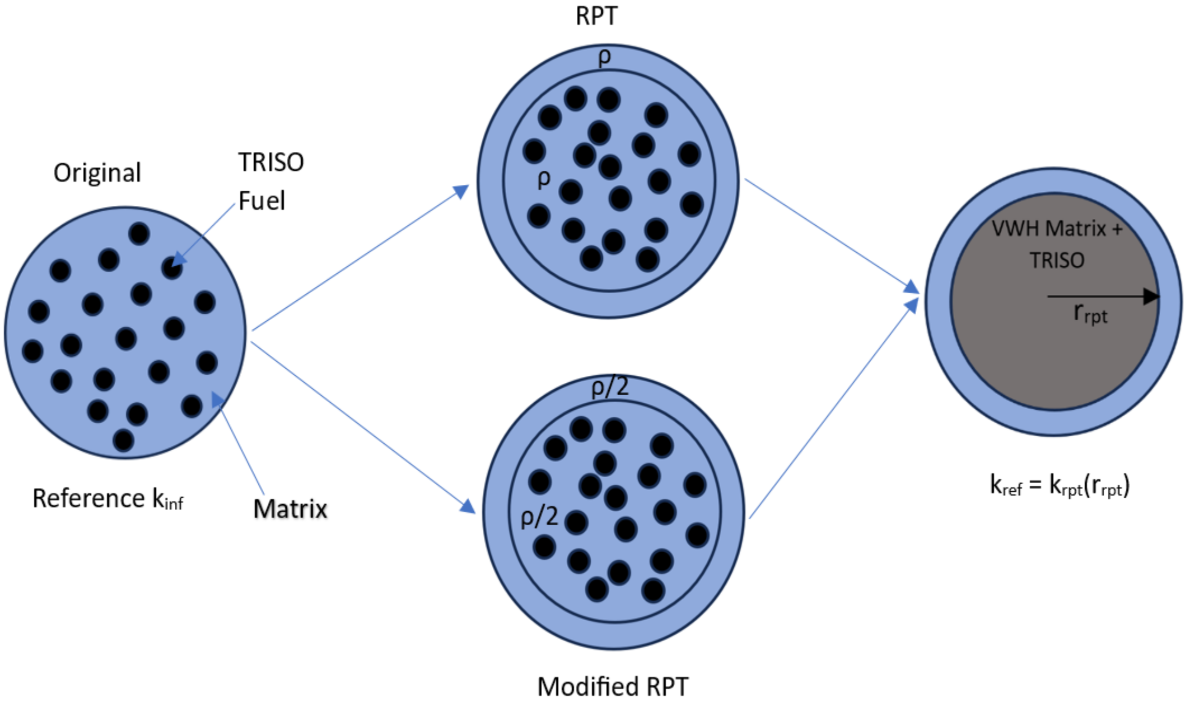


Fig. 6 Schematic for applying reactivity equivalent physical transformation (RPT) method [88], and modified RPT for TRISO [91].

incorporated in PARCS. Doppler Broadening Rejection Correction (DBRC) for heavy nuclides having scattering resonances in the epithermal energy range, used by SERENT, seems to impact the order of ~100 pcm with MCNP5 results. After benchmarking, the SERPENT was used to generate homogenized cross-sections for whole-core calculations using PARCS. Westinghouse AP1000 core was selected as a reference for performing the studies. For fuel temperature branching calculation, it was analyzed that only changing the temperature of the fuel is the better option than changing the fuel's temperature plus TRISO layers along with the ceramic matrix. This is because the second option will over-estimate because of carbon feedback from the TRISO layer. Volume fraction-based homogenization for the thermal properties of the fuel pin is utilized, like the approach used for Pebble Bed Reactors. For slow transients, this approach is acceptable. For RIA, authors suggest that this approach will underestimate the fuel temperature and hence Doppler feedback as due to the fast nature of the transient the energy deposition is almost adiabatic. Hence this approach will not provide conservative predictions. IFBA, WABA, and mixing different BP materials such as B, Gd, and Eu in fuel kernel are studied. "Extreme shift" in the radial power peaking factors was seen from BoC to EoC in the FCM core due to the placement of fresh high enriched fuel assemblies at the periphery and twice burnt fuel assemblies at the center. This leads to a "complete inversion" of radial power distribution. Hence standard shuffling scheme used with the UO_2 core cannot be used with the FCM option and needs a new core shuffling scheme. The RIA studies showed that although the FCM core has less negative temperature coefficients of reactivity, very similar behavior to that of the reference UO_2 core is obtained regarding average energy deposition.

A neutronic study[97] was carried out based on only using FCM fuel pins in a Westinghouse fuel assembly rather than mixing FCM pins with UO_2 pins[72,73]. A more homogeneous design with only using FCM fuel pin-based assemblies is expected which can give a flatter power distribution. TRITON code included in the SCALE 6.1 package was used for analysis. 2D calculations are performed using ENDF/B-VII cross-section-based 238-group library. Burnup and decay calculations are carried out by SCALE/TRITON [92] coupled with ORIGEN. Explicit DH modeling using a two-step self-shielding methodology was available in SCALE but computationally very expensive for large 2D problems. Hence, a modified RPT was used to simulate the DH problem. The modified RPT is shown in Fig. 4 compared to the originally suggested RPT.

The RPT method was benchmarked against KENO-CE and SCALE-DH calculations and was found to be in reasonable agreement. Pin-cell calculations were carried out using the SCALE-DH option while assembly calculations, utilizing quarter assembly, were performed using the RPT. UN performed best from the three options (UO_2, UCO & UN) studied. Due to the same material fuel pins being considered in this concept, a much flatter power distribution was obtained. When the same calculation was performed in a colour set to determine the power-sharing between burnt and fresh fuel assemblies, fresh fuel assemblies showed relatively higher peaking factor values as compared to the once-burnt fuel assemblies. The reason is that the higher enrichment in the fresh FCM fuel assemblies and smaller uranium mass leads to rapid burnup. To obtain an equivalent cycle length with reference, combinations of different packing fractions and buffer layer thickness were studied for different kernel diameter values. A larger packing fraction with a smaller buffer layer thickness performed the best. The results concluded that a 10% increase in fissile inventory compared to reference UO2 fuel assembly is required to reach the same cycle length. Studies performed for neutron spectra at BOL, MOL, and EOL exhibited a spectrum shift from hard to soft for the FCM option with increasing burnup. This spectral shift is due to higher enrichment of U-235 at BOL which depletes quickly and so leads to a softer spectrum at EOL. To control the excess reactivity at BOL, the inclusion of thorium was studied in FCM. Varying quantities of thorium (10-70 at%) were compared with FCM and reference UO_2 design. Thorium obtained the initial excess reactivity suppression, but the conversion of Th-232 to U-233 was not enough to compensate for the decrease in cycle length. The figure of Merit (FOM) to compare the non-proliferation of FCM as compared to conventional UO_2 was studied. At BOL, FCM is more attractive due to the higher enrichment of U-235. At the EOL, a similar value of FOM is obtained for the two studied options which lie in the region of unattractive materials. The authors also pointed out that FOM doesn't consider the difficulty of separation offered by the FCM fuel option.

9. SMR application

The nuclear community has shown interest in applying the FCM for Small Modular Reactors (SMRs).[98–103] Studies performed by Dai *et al*.[98] utilized the FCM concept with a UO_2 fuel kernel for a small PWR concept. The CASMO-4E/SIMULATE-3 code system[104] is used to analyze. Simple homogenization of FCM is used to perform the analysis. Benchmark calculations are carried out to verify the methodology using DRAGON-5 [61] heterogeneous collision probability-based (CP) calculation. The studies aimed to design a reactor with a longer cycle length of> 5 years. The study concluded that a typical PWR lattice would need enrichments exceeding the LEU limit. Hence a modified lattice of 11 x 11 with 9.3 % enriched UO_2 kernel-based FCM was proposed. A single batch core with 89 fuel assemblies using Pu-240 as a burnable absorber was proposed. The suggested BP configuration exhibited a good excess reactivity control with a small reactivity swing of ~5000 pcm throughout a 6-year cycle length. This quantity of control is higher as this core concept doesn't use a chemical shim for normal operation. Only the cold shutdown state requires a chemical shim of 450 ppm. Calculations showed a sudden increase in the EOC's enthalpy rise factor and radial peaking factor value. An average burnup value of 56.4 GWd/tonne HM is obtained with a cycle length of 6 equivalent full power years (EFPY).

Design analysis of a small PWR core having a cycle length of ~ 4 years was carried out by Bae and Hong.[101] A standard 16 × 16 lattice of ABB/CE design was used. The DeCART2D/MASTER code system of the KAERI was used to analyze with the ENDF/B-VII library. A mixed configuration of FCM pins with ThO_2-UO_2 fuel pins is utilized to obtain the net destruction of TRUs. TRU from PWR spent fuel is loaded in FCM fuel pins. Thorium oxide is used to replace a part of conventional UO_2 fuel to avoid the production of TRU and also supplement the efforts for excess reactivity control. Excess reactivity control is deployed using Gd_2O_3 and Er_2O_3 in the form of BISO particle kernels. Also, ZrB_2 coating as IFBA is applied on ThO_2-UO_2 fuel pins in some fuel assemblies. U-235 enrichment of 17 w/o is needed to meet the goal of multiplication factor with 50% ThO_2. A relatively lower average linear power density is used for the design to increase the safety margin compared to typical LWRs. This will also help in extending the cycle length of the design. The combination of 10 w/o SS + 90 w/o graphite is selected. B_4C is employed as CR material. The authors reported a cycle length goal of 4 years, along with net TRU destruction.

Qasim Awan *et al*.[99] performed assembly-level calculations using a UC fuel kernel with SS-304 and FeCrAl cladding. The 4.95% enriched UO_2 fuel assembly of the 3-loop Westinghouse PWR assembly was considered as a reference. Alternate lattice configurations of 15 × 15 and 13 × 13 were proposed. MVP[105] code, using JENDL-3.3[106] cross-section, with explicitly detailed geometry modeling is used to accurately capture the double heterogeneity effects by using the Statistical Geometry Model (STGM) capability of the code. Assembly modification constraints ensured that hydraulic diameter and fissile loading should be the same as that of the reference fuel assembly. Other studies have concluded that a 10% higher fissile loading is required to reach the same value of EFPDs[97] which was confirmed later in this study. A simple heat conduction equation using the solid block supposition exhibited a significantly smaller centreline temperature for the FCM option as compared to the reference fuel assembly. Still, to perform conservative analysis temperatures for the FCM and the reference fuel assemblies were considered equal in subsequent analysis. The modified assembly configurations for FCM showed higher control rod worth than reference fuel assembly. The FCM assembly calculations for the proposed lattice configuration showed a slight reduction in EFPDs with FeCrAl clad. For the SS clad, a much larger penalty for EFPDs was seen. Regarding fuel utilization, relatively much higher values for the FCM fuel assemblies are obtained. Assembly level spectrum calculations at BOL, MOL, and EOL for reference fuel show minimal variation. For the modified lattice FCM assemblies, SS-clad assemblies showed a harder spectrum due to more parasitic absorption of thermal neutrons by Fe. A comparison of Pu isotope concentration showed that FCM-based assemblies produce a significantly smaller amount. This was expected due to smaller U-238 loading in FCM assemblies. Natural Erbia and Gadolinia in the form of QUADRISO are incorporated in the modified lattices to control the excess reactivity. The Erbia option for BP performed better than Gadolinia although a small penalty in cycle length EFPDs was seen due to residue poison at EOC. Enrichment for Erbia was proposed by the studies to counter the problem.

The designed fuel assembly[99] was utilized in proposing a small modular IPWR concept.[100] The assembly design was slightly modified by reducing the clad thickness based on better strength for FeCrAl than the Zr-4 option. This study was performed using the SERPENT code. By lowering the clad thickness for FeCrAl, an increase of 140 EFPDs was seen using assembly-level calculations. The design goals were to develop a small modular reactor with a longer cycle length with a single batch core. To keep the core peaking factors in an acceptable range while obtaining a cycle length of greater than 4 EFPYs, two different enrichment assemblies were used. Following previous assembly-level studies,[99] 80 w/o enriched Erbia was used in QUADRISO particle form for excess reactivity control. A relatively flat power profile was obtained at BOC using different BP configurations. The core discharged fuel was deemed proliferation resistant as it had > 20% Pu-240 present. FTC, MTC, and VRC all remained negative for whole core calculations throughout the cycle length. RCCA rods with HfB_2 enriched up to 99.0% in B-10 were used as control material.

Hakim *et al*.[102] studied the design of a new fuel assembly using UN as fuel kernel material for FCM fuel. SCALE and

SERPENT codes were used for the calculations using the ENDF/B-VII library. Assembly design parameters were proposed based on sensitivity studies. Then, the reactivity coefficients were studied for the whole temperature range, i.e., 300 - 1200 K, and exhibited a negative trend in the complete operating region. Some numerical differences between FTC, MTC, and VRC were exhibited by the two codes. This was to be expected as both codes use different solution methodologies, *i.e.*, deterministic vs. probabilistic with energy variation of cross-section difference along with placement of TRISO particles in SiC compact. The Burnup performance of the selected fuel assembly with varying N-15 enrichment was compared to the reference case. Increasing enrichment of N-15 positively impacted the cycle length, as expected, due to decreasing parasitic absorption of neutrons in N-14. A higher enrichment of U-235 was deemed necessary to reach the same burnup as the reference fuel assembly, due to the smaller HM loading in the FCM fuel concept.

The utilization of FCM in the SMART[107] reactor was studied by Al-Zahrani *et al*.[103] Fuel kernel choices studied were UCO and UN. Higher enrichment and packing fraction values were used to compensate for the smaller HM loading. The IFBA BP design from the reference core was used with FCM. Similar values of average burnup were seen for FCM-loaded cores. Although the cycle length in terms of EFPDs was seen to decrease. Core radial and axial power peaking factors showed better performance as compared to that of the reference core for the FCM-based designs. Relatively less negative values for FTC and MTC were exhibited by FCM-loaded core with both fuel kernels.

The main outcomes from the neutronic studies discussed are summarized here.

- RPT method for catering for the DH effect while some authors used explicit double heterogeneous treatment when using the codes with the specific option available.
- Less negative value for MTC, FTC, and VRC
- A more negative value of soluble shim worth was obtained with FCM
- UN is the kernel of choice with (99.5%) N-15 enrichment
- Higher enrichment is required for FCM to achieve equal fuel cycle length (EFPDs) – the rule of thumb 10% increase in fissile mass[97]
- A fuel temperature difference of 300 compared to 1500 is expected[108], although some authors have used the conventional approach of using the same temperature as UO_2 for FCM fuel kernels.
- Studies suggest that different fuel temperatures for fuel kernel and matrix should be used as the thermal scattering of matrix material will affect the FTC result.
- A change in neutron spectrum from hard to soft is seen for FCM-based cores, which affects the numerical values and trend reactivity coefficients.

10. Conclusion

In the wake of the Fukushima severe accident, the development of the accident-tolerant fuel concept fuel system is actively under development. The fact that most power reactors in operation globally are of the LWR type and, therefore, vulnerable to similar accidents makes addressing this issue even more pressing. The reactor physics analyses are an essential part of nuclear reactor design which can impact the safety and economics of the power production operation. Hence, before incorporating these new fuel / cladding systems in the reactor core reactor physics analyses are essential. New fuel concepts like fully ceramic microencapsulated (FCM) are required to treat the double heterogeneity (DH) effect correctly. Researchers have adopted approaches like the explicit inclusion of DH models in the codes or new and innovative ways to homogenize the TRISO and matrix for correct and reliable analyses. Other special needs for using FCM are higher enrichment requirements to obtain the same/similar cycle length due to smaller HM loading. Higher enrichment will lead to higher excess reactivity at BOC, which will require redesigning the BP/CBC combination. Neutronic benefits provided by the use of FCM fuel are better fuel utilization and the option of using transuranic fuel from LWR burnt fuel for incineration in operating LWRs. FCM fuel also shows a larger rate of reactivity change during burnup compared to conventional UO_2 fuel due to smaller fertile material loading. The development of other fuel materials such as U3Si2, UN, and UC are actively under development. These materials bring higher HM density and better thermal conductivity to the table. Material problems are associated with these fuels, which require additives to be mixed with them. Successful development, manufacturing, and licensing are the stages these materials must go through before they can be used for power production in reactors. Other small to medium-term solutions can be the addition of additives like BeO to UO_2 fuel to increase the thermal conductivity. On one hand, when it solves the problem of low thermal conductivity, it will reduce HM loading. Higher enrichment needs to be used to counter this problem. As for the cladding materials, iron-based claddings have manufacturing experience and associated infrastructure to fulfill the needs and maintain the supply chain. Due to their better strength characteristics, a smaller thickness can be used as compared to zircaloy. This can help offset the reactivity penalty caused by higher parasitic absorption of neutrons in these claddings. The higher neutron absorption for iron-based claddings also makes the neutron spectrum a little harder as compared to the current fuel-clad system. This behavior increases the rim effect in the fuel pellet. The silicon carbide cladding options provide better neutron economy as compared to zircaloy but may require a slightly larger thickness of the clad. In all of the neutron physics aspects, the closest match to the current clad is provided by silicon carbide. Other more innovative options, depending upon the ease and practicability of manufacturing, like a thin layer of coating material or thin tube for the inner and outer surface of zircaloy clad are also inline candidates for short to medium time frame deployment. They will provide the option of using current facilities for the zircaloy-clad manufacturing with new developments taking place to apply the protective layer. Due to the small amount of material introduction, neutronic characteristics are not perturbed much. Not to mention, these can prove to be the easiest of the lot to be licensed.

Despite the significant progress made by nuclear scientists and engineers, gaps still need to be addressed. One of these is in the area of the neutronics, where there is a scarcity of experimental data for Fuel Compact Matrix (FCM). This lack of data impedes the validation of computational code results. The double heterogeneous nature of FCM further complicates matters, as it limits the availability of relevant models in computational codes. Only a select few deterministic codes can handle it. On the other hand, the explicit modeling of each TRISO fuel in FCM is both computationally intensive and challenging in the Monte Carlo codes. This has led to the need for the RPT-like equivalence theories for analysis, which in turn require actual solutions from Monte Carlo reactor physics codes. Another contemplation is proposing a new fuel assembly design for optimal standalone utilization of FCM in Light Water Reactors (LWR). The licensing process for a new fuel assembly is quite rigorous. Furthermore, literature on transitioning from conventional UO_2 fuel to an FCM-based core is sparse. While fresh core analyses are available, studies on transition and equilibrium cycle configurations are hard to come by. Comprehensive multi-physics examinations of novel materials, considering the intertwined influences of neutronics, thermal hydraulics, chemistry, and material characteristics, are crucial in assessing suggested ATF materials' viability, advantages, and disadvantages. However, the existing literature does not sufficiently cover this aspect of ATF analysis. This demands combined effort and comprehensive studies by the nuclear community to introduce the ATF concepts and make nuclear power even safer.

Data Availability

Acknowledgment

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Conflict of Interest

There is no conflict of interest.

Supporting Information

Not applicable.

Abbreviations

APMT	Advanced Powder Metallurgical Technology
ATF	Accident Tolerant Fuel
BA	Burnable Absorber
BDBA	Beyond Design Basis Accident
BISO	Bistructural Isotropic
BOC	Beginning of Cycle
BOL	Beginning of Life
BP	Burnable Poison
BWR	Boiling Water Reactor
CBC	Critical Boron Concentration
CONFU	Combined Non-fertile and UO_2
DBA	Design Basis Accident
DRC	Doppler Reactivity Coefficient
ECCS	Emergency Core Cooling System
EFPD	Equivalent Full Power Days
EOC	End of Cycle
EOL	End of Life
FCVS	Filtered Containment Ventilation System
FeCrAl	Iron-Chromium-Aluminum
FTC	Fuel Temperature Coefficient
HEU	High Enriched Uranium
HTGR	High-Temperature Gas-cooled Reactor
IFBA	Integrated Fuel Burnable Absorber
IMF	Inert Matrix Fuel
IMDP	Inert Matrix Dispersion Pellets
KAERI	Korea Atomic Energy Research Institute
LEU	Low Enriched Uranium
LHGR	Linear Heat Generation Rate
LWR	Light Water Reactor
MOC	Middle of Cycle
MOL	Middle of Life
MTC	Moderator Temperature Coefficient

NGNP	Next Generation Nuclear Plant
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PBAT	Particle-Based Accident Tolerant
PCI	Pellet Clad Interaction
PWR	Pressurized Water Reactor
QUADRISO	Quadrupule Isotropic
R&D	Research and Development
SBO	Station Blackout
SS	Stainless Steel
TMI	Three Mile Island
TRISO	Tristructural Isotropic
TRU	Trans Uranic
VRC	Void Reactivity Coefficient
WABA	Wet Annular Burnable Absorber

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