Small modular reactor core design for civil marine propulsion using micro-heterogeneous duplex fuel. Part II: Whole-core analysis

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Abstract

Civil marine reactors face a unique set of design challenges. These include requirements for a small core size and long core lifetime, a 20% cap on fissile loading, and limitations on using soluble neutron absorbers. In this reactor physics study, we seek to design a core that meets these requirements over a 15 effective full-power-years (EFPY) life at 333 MWth using homogeneously mixed all-UO₂ and micro-heterogeneous ThO₂-UO₂ duplex fuels. In a companion (Part I) paper, we found assembly designs using 15% and 18% ²³⁵U for UO₂ and duplex fuels, respectively, loaded into 13×13 pin arrays. High thickness (150 μm) ZrB₂ integral fuel burnable absorber (IFBA) pins and boron carbide (B₄C) control rods are used for reactivity control. Taking advantage of self-shielding effects, these designs maintain low and stable assembly reactivity with little burnup penalty.

In this paper (Part II), whole-core design analyses are performed for small modular reactor (SMR) to determine whether the core remains critical for at least 15 EFPY with a reactivity swing of less than 4000 pcm, subject to appropriate constraints. The main challenge is to keep the radial form factor below its limit (1.50). Burnable poison radial-zoning is examined in the quest for a suitable arrangement to control power peaking. Optimized assemblies are loaded into a 3D reactor model in PANTHER. The PANTHER results confirm that the fissile loadings of both fuels are well-designed for the target lifetime: at the end of the ~15-year cycle, the cores are on the border of criticality. The duplex fuel core can achieve ~4% longer core life, has a ~3% lower initial reactivity and ~30% lower reactivity swing over life than the final UO₂ core design. The duplex core is therefore the more successful design, giving a core life of ~16 years and a reactivity swing of less than 2500 pcm, while satisfying all the neutronic safety parameters. In particular, one of the major objectives of this study is to offer/explore a thorium-based candidate alternative fuel platform for the proposed marine core. It is proven by literature reviews that the ability of the duplex fuel was never explored in the context of a single-batch, LEU, SBF, long-life SMR core. In this regard, the motivation of this paper is to understand the underlying physics of the duplex fuel and ‘open the option’ of designing the functional cores with both the duplex and UO₂ fuel cores.

Keywords: Civil marine propulsion, Small modular reactor (SMR), Soluble-boron-free (SBF), Long life, Micro-heterogeneous thorium-based duplex, Burnable poison zoning (BPZ), Radial form factor (RFF)

1. Introduction

A nuclear-powered ship – be it a surface ship or a submarine – receives its propulsion energy from a nuclear power plant on board, and can be dubbed an small modular “atomic battery” (Ragheb, 2011). The main advantages of nuclear marine propulsion are that atomic engines are reliable, compact sources of energy and can operate for years without new fuel while not producing exhaust gas (Hirdaris et al., 2014). Furthermore, with increasing attention being given to greenhouse gas emissions arising from the burning of fossil fuels for international air and marine transport and the excellent safety record of nuclear-powered ships, renewed interest in marine nuclear propulsion is likely (Mitenkov et al., 2003). Looking at medium- to long-term options (Hirdaris et al., 2014), given that hydrogen is not yet ready for shipboard installation (Aspelund et al., 2006), there is currently no solution that eliminates all emissions and no other solution that can offer significant CO₂ reduction (Anger, 2010). Since nuclear fission produces no direct emissions, it clearly enjoys key environmental advantages over current diesel engines. The rationale for considering nuclear-powered propulsion systems as a means to de-carbonise commercial freight shipping and an overview of the history and development of nuclear naval vessels were presented in the Introduction to this paper’s Part I companion (Alam et al., 2019c). In the interest of brevity, these will not be repeated here.

The applicability of this design study relies on developing a set of technical requirements, identifying suitable assumptions and imposing appropriate constraints (Alam, 2018). Since pressurized water reactors (PWRs) have a proven maritime operation record (Ragheb, 2011), the study considers such a reactor with an assumed thermal power of 333 MWth, typical of past marine reactors, such as those on the Savannah and Otto Hahn, and proposed future reactors (Alam et al., 2019a, Namikawa et al., 2011, Ragheb, 2011). In the companion (Part I) paper (Alam et al., 2019c), we focused on assembly-level design and considered the design of burnable poisons (BPs) and control rods for a soluble-boron-free (SBF) 333 MWth marine PWR using 15% and 18% ²³⁵U fuels for all-UO₂ and micro-heterogeneous ThO₂-UO₂ duplex¹ (Alam et al., 2019d,b, Zhao, 2001, MacDonald and Lee, 2004, Shwageraus et al., 2004) fuels loaded into 13×13 assemblies.

We examined gadolinia, erbia and integral fuel burnable absorber (IFBA) as candidate BP materials, and concluded that, taking advantage of self-shielding effects, the design using IFBA offers low, stable assembly reactivity with relatively little burnup penalty. The duplex fuel shows greater promise by offering ~20% lower reactivity swing and ~4% lower initial

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¹We use the term ‘duplex’ to refer to the micro-heterogeneous ThO₂-UO₂ duplex fuel throughout this study.

reactivity than all- UO_2 fuel. We investigated boron carbide (B_4C), hafnium and Ag-In-Cd alloy as candidate control rod materials. All the candidate materials show greater rod worth for the duplex design. For both fuels, B_4C has the highest reactivity worth, providing $\sim 5\%$ more control rod worth for duplex fuel than all- UO_2 .

In this paper (Part II), whole-core analysis is performed to identify the designs of components and parameter values that will optimize performance. The main challenge in designing a long-life SBF core is to keep the power peaking factor below its limit of 1.50. In order to control the through-life reactivity swing, a large number of thick-coating IFBA pins are needed. Unfortunately, for these IFBA pins, the absorption effect is so strong that they are essentially black absorbers at beginning-of-life (BOL), shielding the fissile isotopes within the pin, producing little power and depressing the local neutron flux. Thus, when the poison layer has burned off, these pins have a high fissile content, leading to unacceptable power peaking factors. Burnable poison radial-zoning arrangements are therefore investigated with the aim of keeping the radial form factor (RFF) below 1.50 over the core life. We have also examined homogeneous assembly loading (loading the entire core with the same fuel derived from Part I) to confirm our expectation that such an arrangement cannot successfully control power peaking.

The main goals of this whole-core design study are:

1. To verify that the assembly design has sufficient reactivity for the target 15 effective full-power-years (EFPY) period of operation;
2. To ensure the RFF is below 1.5 throughout the core life;
3. To confirm that the through-life reactivity swing does not exceed 4000 pcm in order to reduce the dependency on control rods in the long-life SBF core;
4. To evaluate the reactivity feedback parameters, core form factors, axial offset and shutdown margin.

In each case, consistent comparisons are made between the all- UO_2 and duplex fuel cores in order to determine their relative performance. The detailed analyses of the underlying physics of the neutronic safety parameters seek to go beyond those presented in previous studies related to long-life SBF small modular reactor (SMR) cores, especially for thorium-based duplex fuel cores (Mart, 2013, Thomet, 1999, Dai and Cao, 2013, Kim et al., 1998).

It is also important to address the main motivations behind this study:

1. The objective is not at all to establish or justify the superiority of the duplex fuel over the UO_2 fuel. As addressed, the ability of the duplex fuel was never explored in the context of a single-batch, LEU, SBF, long-life SMR core. Therefore, it is a strong motivation rather to understand the underlying physics of the alternative fuel platform (duplex fuel) and observe the neutronic performance of the proposed duplex fuel with respect to the UO_2 fuel.
2. To obtain the satisfactory neutronic performance for both the candidate fuel cores and conclude which fuel provides the better neutronic characteristics.

3. The idea of this paper is not just to satisfy the safety parameters, but to successfully “open the option” of designing the proposed SBF, marine SMR core with both the duplex and UO_2 fuel cores.
4. It will be many years before civil marine propulsion achieves widespread commercialisation, and rapid improvements in materials and fabrication lead to permitting ever-higher burnup. Therefore, higher local burnup is not considered as a constraint (as with current commercial reactors) for the design of long life core, instead, it is an objective. For the purpose of this study, it is assumed that suitable materials and technology that can withstand prolonged core life and high burnup will be available for future use for the long life marine core.

2. Analysis methods

The assembly and core sizing calculations and the reactor physics calculational routes used in these studies, using the WIMS-10 lattice physics code (Newton et al., 2008) and the PANTHER whole-core nodal code (Hutt, 1992). For our study, the assembly design analysis relied on the WIMS-10 software package using the JEF 2.2 nuclear database available from the IAEA (Newton et al., 2008). For each burnup step, WIMS completes a 172 “fine” group solution to the transport equation in a smeared geometry. It then refines this solution using a few-group calculation in a precise geometry. For our calculation we have considered 6 energy group structure as shown in Table 1. Furthermore, we have used PANTHER (Hutt, 1992) for whole-core design which is an advanced 3D nodal code for reactor core analysis. PANTHER imports few-group nuclear data from the lattice code WIMS. The neutronic powers are calculated using the PANTHER using a nuclear data library prepared with WIMS. WIMS produces “Flat file” for subassemblies with specific neutronic properties. By using an output module “LED”, WIMS homogenizes neutronic data for subassembly and exports the “flat file” that is readable by PANTHER. This allows user to load subassemblies prepared in WIMS into PANTHER in order to perform whole-core calculations. Therefore, WIMS and PANTHER are interconnected in this study to perform the subassembly design and whole-core analysis. The detail calculational route is described in the Part I companion paper (Alam et al., 2019c).

Group	1	2	3	4	5	6
Upper fine group	1	23	46	93	136	153
Lower fine group	22	45	92	135	152	172
Upper (eV)	19.64×10^6	820.85×10^3	9.12×10^3	4.00	625×10^{-3}	140×10^{-3}
Lower (eV)	820.85×10^3	9.12×10^3	4.00	625×10^{-3}	140×10^{-3}	110×10^{-6}

Table 1. 6-group WIMS energy structure.

In that paper, we developed an assembly design for consistent and controllable reactivity over a long core life. In this whole-core study, we now load assemblies into a whole-core model so we can assess the effects of neutron leakage and spatial flux variation. Ideally, we would like to find a loading pattern (LP) that gives a constant power-shape over time—such

a power-shape will typically yield the lowest power peaking factors (the ‘Haling principle’ (Glasstone and Sesonske, 1994))—and the RFF should never exceed the standard industry limit of 1.5 (Pramuditya and Takahashi, 2013).

System parameter values for the assembly-level and whole-core PANTHER calculations are shown in Table 2 (Alam et al., 2018c,d, 2016a,c,b, 2017b).

We include a reflector around the core, defined in WIMS by the smeared properties of core baffle and water. In our whole-core radial-zoning studies, the operational conditions are all control rods out (ARO) and hot full power (HFP).

Whole-core system parameters		Assembly system parameters	
Parameter	Value	Parameter	Value
Thermal power (MWth)	333.33	Assembly size	13×13
Coolant pressure (MPa)	15.5	Control rods per assembly	16
Soluble boron concentration (ppm)	0	Pin pitch (mm)	12.65
Total coolant flow rate (kg/s)	8370.2	Fuel pellet diameter (mm)	8.19
Core inlet temperature (°C)	295.6	Cladding thickness (mm)	0.605
Number of reactor channels	164	Gap thickness (mm)	0.0498
Number of core channels	112	Total rod diameter (mm)	9.5
Number of channel columns	14	Pitch/diameter ratio	1.33
Number of channel rows	14	Hydrogen-to-heavy metal (HHM) ratio	3.99
Number of core axial layers	16	Assembly side length (cm)	16.45
Number of reactor axial layers	18	Assembly area (m ²)	0.03
Active core height (m)	1.79		
Core diameter (m)	1.97		
Number of radial reflector channels	52		

Table 2. Whole-core and assembly-level system parameter values for PANTHER simulations.

3. Homogeneous assembly loading

The first design considered is a simple loading of the 15% ²³⁵U enriched all-UO₂ and 18% ²³⁵U enriched duplex fuel assemblies from our WIMS analysis into all 112 core locations, as shown in Fig. 1a. We have used 33 IFBA pins with 150 µm coating for both fuels since we optimized this BP loading as our final design in our assembly-level study (Alam et al., 2019c). The core life is estimated from the reactivity curve (Fig. 2) at the point where the effective multiplication factor $k_{eff} = 1.0$.

Fig. 2 shows that the reactor cores remain critical throughout the designated core lifetime of 15 years for both candidate fuels. After 15 years, both cores are on the border of criticality, showing that the fissile loadings found in the assembly-level study were well-designed (Alam et al., 2019c). In fact, the duplex core can achieve ~5% longer core life (~16.9 years) than the UO₂ core (~16.1 years) due to the higher thermal ‘fertile-capture:fissile-absorption’ ratio (Argala et al., 2011), which is discussed further in Sect. 4.1.1.

Fig. 1. Core loading patterns: (a) Homogeneous assembly loading; (b) Radial-zoning pattern: blue = Zone A (central), green = Zone B (inner), red = Zone C (outer).

Fig. 2. k_{eff} over time for the homogeneous core LPs.

Fig. 2 also shows that the duplex core has a lower BOL reactivity than the UO_2 core for the same thickness and number of IFBA pins. The duplex fuel exhibits higher thermal ‘total fertile capture’ (as shown in Fig. 3) compared to the UO_2 fuel. The ^{238}U capture rate in both fuels is approximately the same, but capture in ^{232}Th leads to an increased total capture rate in the duplex fuel (Alam et al., 2019a,b), which, in turn, results in lower BOL reactivity.

Fig. 3. Normalized total fertile capture rates at BOL with 33 IFBA pins (at the assembly-level).

It can also be seen that the through-life reactivity swing values are $\sim 6,400$ pcm and $\sim 5,600$ pcm for the UO_2 and duplex cores, respectively. These reactivity swing values are well above the target maximum reactivity swing value of 4000 pcm, suggesting that higher poison loadings and/or heterogeneous LPs are required to reduce the swing and, thereby, the dependence on control rods.

The RFF throughout life is presented in Fig. 4. The homogeneous LPs exhibits unsatisfactory power peaking due to the uneven irradiation between assemblies, and the maximum RFF value is well above the standard limit of 1.50 throughout most of the core life for both candidate fuels. Figs. 5a and 5b illustrate the normalized radial power distributions at different stages of life for both fuels. For both candidate cores, there is strong power peaking in the central assemblies early in life, leading to ‘over-depletion’ of their fissile content. This causes the peak-power area to shift radially away from the centre over time. It can also be seen that the power in the outermost assemblies is well below average throughout life. To mitigate these undesirable features, the power in the inner assemblies should be reduced and the power in the outer assemblies increased.

It can be concluded that the homogeneous LPs offer unsatisfactory performance in terms of reactivity swing and power peaking. Our next analysis will therefore focus on radial-zoning of the core.

Fig. 4. RFF over time for the homogeneous core LPs.

4. Radial-zoning whole-core analysis

We observed in Sect. 3 that a homogeneous LP is not an effective way of controlling power peaking, due to the uneven irradiation between assemblies. One way to reduce RFF is to divide the core into radial zones and adjust the properties of each zone to flatten the power distribution (Pramuditya and Takahashi, 2013). Using PANTHER, we modeled the 112-assembly core, divided into three radial zones (A, B and C, see Fig. 1b). The two most obvious radial-zoning options are fissile zoning (FZ) and burnable poison zoning (BPZ). Since we have observed in a previous study (Alam et al., 2015) that FZ is not an effective way of controlling power peaking for a small marine SBF PWR core, we have investigated BPZ arrangements in this study. In this section, normalized radial power distributions are observed for the BPZ arrangements for both candidate fuel cores.

4.1. Burnable poison zoning

In the BPZ scheme, reference (X%)² ²³⁵U content is utilized.

In order to select the “number of BP pins” in Zones A, B and C for the BPZ design, sensitivity studies in BP loading is performed. This is performed in such a way that minimum reactivity swing can be achieved while obtaining at least 15 years core life. Since duplex fuel exhibits lower BOL reactivity than that of the UO₂ fuel (as suggested by Sect. 3), optimizing the BP loadings for the UO₂ and comparing the neutronic performance with the duplex cores with the same BP loadings might lead to subcriticality. Following the selection of the BP loadings for UO₂ core, a lower BP loading has been used in the duplex core due to their lower BOL k_{∞} .

Figs. 6a and 6b show that the selected BP pin loadings for the candidate fuels exhibit the minimum reactivity swing while obtaining at least 15 years core life. The numbers of IFBA pins in the assemblies in Zones A, B and C for the BPZ scheme are shown in Table 3.

²Here ‘X’ denotes the reference ²³⁵U content: 15% for all-UO₂ and 18% for duplex fuel.

(a)

(b)

Fig. 5. Normalized radial power distributions with homogeneous core loading: (a) UO_2 ; (b) Duplex.

(a)

(b)

Fig. 6. k_{eff} over time for the BPZ core loading patterns (in the legends a-b-c represents BP pins in Zone A, B and C, respectively): (a) UO_2 ; (b) Duplex.

174 In this radial-zoning analysis, we allocated higher numbers of IFBA pins overall in the UO_2
 175 cores due to their higher BOL k_∞ , as suggested by Sect. 3. This strategy uses zoned BP
 176 loading in an effort to levelize radial power.

Zones	$^{235}\text{U}(\%)$	UO_2	Duplex
A	X%	55	53
B	X%	33	31
C	X%	21	17

Table 3. BPZ fissile loading and 150 μm IFBA pins distributions.

177 Figs. 7a and 7b show that initially this scheme results in low power in the central
 178 assemblies (Zone A) for both the UO_2 and duplex cores due to the higher poison content
 179 there. The power share in the peripheral assemblies (Zone C) is correspondingly higher
 180 at BOL (due to the lower BP loading in Zone C) and shifts towards the inner zones over
 181 the core life. When the BP in Zone A burns out after ~ 12 years, the power share in these
 182 assemblies increases, but the end-of-life (EOL) RFF is well below 1.5.

Fig. 7. Normalized radial power distributions with BPZ: (a) UO_2 ; (b) Duplex.

By dividing the 112 assemblies into radial zones and successively adjusting the poison loading and enrichment of each zone, the local power peaking can be kept below 1.5. The highly self-shielded IFBA pins shield fissile isotopes until they are depleted. To mitigate the problem of unacceptable power peaking arising due to high fissile content (after the poison layer has burnt off), poison burn-out is delayed in the BPZ arrangement. The fissile content in the fuel pins at poison burn-out is therefore lower and thus helps to reduce the power peaking (Mart, 2013, Kauchi and Shimazu, 2003).

Next, we will analyze the core life, through-life reactivity swing, form factors (radial, axial and total form factor), and core safety analysis parameters – reactivity feedback parameters (moderator and fuel temperature coefficients), axial offset and shutdown margin – for the BPZ arrangement for the candidate cores.

4.1.1. Core life

It is important to determine the achieved core life for the radial-zoning whole-core designs. Fig. 8 shows that the reactor cores with both candidate fuels remain critical throughout the designated core lifetime of 15 years (at least). The duplex core achieves ~4% longer core life (15.82 EFPY) than the UO₂ core (15.25 EFPY) for the BPZ design.

Fig. 8. k_{eff} over time for the radial BPZ core LPs.

In the duplex fuel, an individual fuel pin is composed of a UO₂ centre surrounded by an annulus of ThO₂ (Fig. 9a), and thus the thermal neutron flux from the moderator is incident on the ThO₂ region. This is beneficial for the efficient breeding of ²³³U. The interior UO₂ region experiences a lower magnitude of thermal flux. Fig. 9b confirms that the thermal flux level is higher in the fertile ThO₂ region and lower in the UO₂ region. The lower flux level and harder spectrum in the UO₂ region reduce the depletion rate of ²³⁵U. This also helps in achieving a relatively higher thermal ‘fertile-capture:fissile-absorption’ ratio and power share in the fertile ThO₂ region. The accumulation of fissile material in fertile rods is useful in

prolonging the core lifetime, since it supplements the reactivity at EOL of the UO_2 region (Argala et al., 2011).

Fig. 9. (a) Configuration of the micro-heterogeneous duplex ThO_2 - UO_2 fuel; (b) Normalized flux per unit lethargy in the ThO_2 and UO_2 regions with 33 IFBA pins (at the assembly-level).

The higher thermal ‘fertile-capture:fissile-absorption’ ratio (Jagannathan et al., 2008) (Fig. 10) of the duplex core in the thermal energy range leads to the efficient breeding of fissile ^{233}U over the core lifetime and opens the possibility of a thermal breeder. The higher conversion ratio (CR) and fissile inventory ratio (FIR) of the duplex fuel (Fig. 11a) lead to a longer core life than for UO_2 fuel (Csom et al., 2012, Tsige-Tamirat, 2011). The higher CR and FIR of duplex fuel compared to UO_2 arise from the higher depletion rate of fertile material (Fig. 11b), and the consequent better conversion of fertile material to fissile material (Liu and Cai, 2013).

4.1.2. Reactivity swing

Table 4 shows that the through-life reactivity swing does not exceed 4000 pcm in the BPZ arrangements for the candidate fuel cores.

Configuration	UO_2 (pcm)	Duplex (pcm)	Difference (%)
BPZ	2989	2280	31.10

Table 4. Reactivity swing for the radial BPZ core LPs.

The duplex core exhibits $\sim 30\%$ smaller swing than the UO_2 core. The fissile content of the duplex fuel reduces more slowly due to the enhanced breeding of fissile material (^{233}U in the duplex case vs. ^{239}Pu in the UO_2 case). This effect allows the initial reactivity of the fuel to be lower while achieving the same (indeed longer) cycle length, and accounts for the reduction in reactivity swing. It was addressed that higher reactivity swing is always a

Fig. 10. Fertile-capture:fissile-absorption ratios at BOL with 33 IFBA pins (at the assembly-level).

problem for the SBF core in terms of control rod insertion (to obtain the criticality). Duplex fuel exhibits the ability to reduce the maximum swing. It is helpful for reducing the control rod insertion which helps to improve the peaking factors.

Although the magnitude of maximum reactivity is not a limiting safety criterion for the standard PWR (rather the availability of sufficient shutdown margin, as discussed in in Sect. 4.5), the maximum through-life reactivity (reactivity swing) is a limiting safety criterion for the SBF core concept, which requires higher fissile loading to obtain the long life. Since only control rods and burnable poisons are used in the SBF marine core, it is always recommended (Mart, 2013, Thomet, 1999, Dai and Cao, 2013, Kim et al., 1998) to obtain the maximum through-life reactivity below 1.04 (4000 pcm). Otherwise, it would be seriously challenging to obtain the criticality while achieving the satisfactory power peaking factors. In this regard, almost 30% reduced reactivity swing of the duplex fuel compared to the UO_2 fuel can make the reactivity control system using control rods (to obtain criticality) much easier while not worsening the peaking factors. This is an obvious benefit in terms of controlling excess reactivity without using soluble boron.

Finally, it can be noted that the duplex core outperforms the UO_2 one in terms of BOL reactivity hold-down and reactivity swing, which will make the task of reactivity control through BP design easier in a SBF duplex core.

4.2. Core form factors

It is important to address the core form factor issues which arise from normalized axial and radial power profiles. Local power is restricted with some margin to some absolute limit in order to ensure that, under certain anticipated operational anomalies (such as fuel damage, corrosion, release of volatile fission products, associated clad ballooning and other

(a)

(b)

Fig. 11. (a) Conversion ratio (right axis) and fissile inventory ratio (left axis) vs. burnup with 33 IFBA pins (at the assembly-level); (b) Fertile material depletion vs. burnup with 33 IFBA pins (at the assembly-level).

irradiation-dependent consequences), the fuel is not damaged, or, in the event of a postulated accident, the fuel damage is limited and not underestimated (Barchevtsev et al., 2002, Garcia-Delgado et al., 1999). From a fuel performance perspective, it is recommended to ‘flatten’ the radial and axial profiles in order to minimize corrosion and other irradiation-dependent consequences (Daing and Kim, 2011, Garcia-Delgado et al., 1999). Core form factors: RFF, axial form factor (AFF) and total form factor (F_Q) are discussed in this section.

4.2.1. Radial form factor

As mentioned earlier, one of the main objectives of the whole-core radial-zoning studies is to keep the RFF below 1.50 throughout the core life. Fig. 12 shows the RFF variation over time for the radially-zoned cores.

Fig. 12. RFF over time for the radial-zoning core LPs.

The BPZ scheme enables the RFF to be maintained below the 1.50 limit for both fuels throughout life. In both cases, the RFF increases gradually over life, but is still well below the limit at EOL. The duplex core experiences a lower through-life maximum RFF value than that for the UO_2 core due to the more moderate reactivity change in the duplex fuel (Kotlyar et al., 2017).

4.2.2. Axial form factor

AFF is an important parameter which affects the fuel and cladding temperatures. The variation of the AFF is shown in Fig. 13a. AFF values should not exceed 1.60 at HFP (Kim and Kim, 2000).

Fig. 13a shows that the BPZ scheme enables the AFF to be maintained below its permissible limit throughout the core lifetime for both candidate cores. It is seen, for both cores, that the AFF increases with time, reaching a maximum at ~ 8 – 9 years, and then decreasing. It can also be observed that the AFF values are higher for the UO_2 core at BOL. Things are different after ~ 1 year, where AFF values are higher for the duplex core. Axial relative power (ARP) is therefore observed at BOL and at year 9 (the point of maximum AFF) in order to analyze the consistency of the AFF behaviour.

(a)

(b)

Fig. 13. Axial power profile for the radial BPZ core LPs: (a) AFF over time; (b) Axial relative power.

It can be observed in Fig. 13b that the maximum ARP values are higher at BOL for the UO₂ core and at year 9 for the duplex core, respectively, at an axial level of $\sim 0.8\text{--}0.9$ m, consistent with the maximum AFF values for the corresponding cases (Fig. 13a). Since the duplex core exhibits higher AFF values for most of the core life, it is expected that it will exhibit a higher axial offset than the UO₂ core.

4.2.3. Total form factor

It is also important to observe the total form factor, F_Q . The maximum value of F_Q should not exceed 2.52 at HFP (Kim and Kim, 2000). It is recommended to avoid higher F_Q values, since those lead to higher fuel and cladding surface temperatures causing an increase in the fraction of fission gas released (Barchevtsev et al., 2002, Garcia-Delgado et al., 1999). In addition, a lower maximum F_Q value is beneficial in terms of controlling axial offset and reactivity variation throughout core life (Daing and Kim, 2011). The variation in F_Q is shown in Fig. 14. It can be seen that both the candidate fuel cores enable F_Q to be maintained below the permissible limit throughout the core lifetime.

Fig. 14. F_Q over time for the radial BPZ core LPs.

Fig. 14 shows a general trend in F_Q values for the BPZ configurations. F_Q values decrease with time early in life until a certain point (~ 3 years) due to the depletion of fissile material (Barchevtsev et al., 2002). After this point, F_Q values begin to increase until they reach a maximum at ~ 9 years due to power increase in the central assemblies (due to the BP depletion). However, after ~ 12 years, a sharp increase in F_Q values is observed for both cores, consistent with their RFF values after ~ 12 years (as shown in Fig. 12) (Barchevtsev et al., 2002). In addition, similar to the RFF behaviour (in Sect. 4.2.1), the duplex core experiences lower through-life maximum F_Q values than the UO₂ core due to the more moderate reactivity change in the duplex fuel (Kotlyar et al., 2017).

4.3. Reactivity feedback analysis

Reactivity coefficients play a vital role from a safety point of view, since they are a measure of the stability of reactor operations. A negative reactivity coefficient is required to

make a reactor self-regulating. Two types of reactivity coefficient – moderator temperature coefficient (MTC) and fuel temperature (Doppler) coefficient (FTC) – are considered for both candidate cores at HFP. FTC values are evaluated by increasing the fuel temperatures by 20 K from the reference value of 900 K. MTC values are calculated by increasing the coolant temperature by 20 K and modifying the water density accordingly.

4.3.1. Moderator temperature coefficient

SBF operation offers potential enhanced safety due to the strong, negative MTC of both cores over the entire core life. Strong, negative MTC values are observed due to the elimination of the soluble boron system (Mart, 2013, Fiorini et al., 1999, Kim and Kim, 2000, Alam et al., 2018b).

Fig. 15. MTC over time for the radial BPZ core LPs.

Fig. 15 shows that MTC values are negative throughout for both cores. It also shows that the MTC values are more negative in the duplex core than the UO₂ core at BOL, due to the increased neutron absorption in the ²³²Th resonances as the moderator temperature increases. MTC is higher (less negative) at BOL for both cores, and becomes more negative over the course of the cycle, due to the changes in isotopic composition and the increased variety of isotopes present (Alam et al., 2017a, 2018a).

Fig. 15 also shows that MTC is slightly more negative in the UO₂ core than the duplex core later in life. As the burnup progresses, more plutonium (particularly ²³⁹Pu) is bred in the all-UO₂ fuel than the duplex fuel (Fig. 16). Due to the strong thermal neutron absorption of plutonium, the MTC is more negative in the all-UO₂ core at EOL (Alam et al., 2017a, Oka, 2014).

The differences in MTC between the candidate fuel cores are primarily due to differences in the resonance capture within the fuels (Brown et al., 2015). The resonant behaviour can be appreciated by observing the normalized total capture in the fuel per unit lethargy at BOL, and is indicative of the extent to which resonance capture plays a role, and the resultant impact on MTC (Franceschini and Petrović, 2009, Brown et al., 2015). Fig. 17 shows that duplex fuel exhibits higher resonant capture in the vicinity of ~21–22 eV (top

Fig. 16. ^{239}Pu inventory vs. burnup for the candidate fuels with 33 IFBA pins (at the assembly-level).

Fig. 17. Normalized total capture rates per unit lethargy at BOL with 33 IFBA pins (at the assembly-level).

Fig. 18. (a) Total capture cross-section of ^{239}Pu ; (b) Differences in ^{239}Pu capture rates between duplex and UO_2 fuel at 0.20 GWd/tonne with 33 IFBA pins (at the assembly-level).

right corner of the figure) and thereby provides lower (more negative) MTC values compared to the UO_2 fuel. Fig. 17 also shows the difference in normalized total capture rates per unit lethargy between the duplex and UO_2 fuels at BOL. It can be seen that duplex fuel yields a higher total capture rate over the energy range. The duplex fuel core therefore exhibits a more negative MTC.

It is also worth evaluating the contribution of plutonium to MTC. Plutonium has a larger thermal neutron absorption cross-section than uranium, and its presence hardens the neutron spectrum and makes a negative contribution to MTC (Oka, 2014). Fig. 18a shows that ^{239}Pu has a resonance at ~ 0.30 eV. ^{239}Pu contributes a large fraction of the neutron captures in the fuel due to this resonant behaviour (Rouben, 2016, Franceschini and Petrović, 2009). Fig. 18b shows the differences in ^{239}Pu capture between duplex and UO_2 fuel at 0.20 GWd/tonne: duplex fuel exhibits higher ^{239}Pu capture than the UO_2 fuel in the vicinity of ~ 0.30 eV.

Finally, it can be concluded that, due to the higher total capture rates associated with ^{239}Pu , the duplex core provides more a negative MTC than the UO_2 core.

4.3.2. Fuel temperature coefficient

Fig. 19 shows that the values of FTC are negative throughout the core life for both cores. The FTC values of the duplex core are considerably more negative compared to the UO_2 cores, due to the stronger Doppler effect that arises from the simultaneous presence of ^{232}Th and ^{238}U in the duplex fuel (Björk et al., 2011, Alam et al., 2016a). FTC is higher (less negative) at BOL for both cores, and its value decreases (becomes more negative) with burnup due to the increase in plutonium content over the core life (Pope et al., 2012, Alam et al., 2017a).

An increase in captures in the ^{238}U and/or ^{232}Th resonances due to Doppler broadening is

Fig. 19. FTC over time for the radial BPZ core LPs.

Fig. 20. Differences in total fertile capture rates between duplex and UO_2 fuels at BOL with 33 IFBA pins (at the assembly-level).

the main phenomenon affecting the FTC (Ganda and Greenspan, 2010, Alam et al., 2017a). Fig. 20 shows the difference in total fertile capture rates per unit lethargy at BOL between the duplex and UO₂ fuels. It can be seen that the duplex fuel exhibits higher total fertile capture than the UO₂ fuel. The capture rate of ²³⁸U in both fuels is approximately the same, but the ²³²Th capture rate leads to an increase in the total capture reaction rate in the duplex fuel. The larger total fertile capture rates in the vicinity of ~6–35 eV for the duplex fuel contribute to the more negative FTC values (Alam et al., 2017a).

4.4. Core axial offset

The axial offset (AO) is an important reactor physics safety parameter used to evaluate the core axial power distribution (Lau et al., 2012, Daing and Kim, 2011). It is calculated as the difference in power between the top half and the bottom half of the core, divided by the total core power.

Fig. 21. Axial offset (%) over time for the radial BPZ core LPs.

The AO is determined from reactor core power distribution calculations, performed at near HFP, near ARO, steady-state operating conditions. In order to improve core stability, generally acceptable AO values range from -10% to +10% for an SBF PWR core (Dai and Cao, 2013, Lau et al., 2012, Daing and Kim, 2011, Alam et al., 2016a). The AO values over the core life for the BPZ arrangements are shown in Fig. 21. The values lie in the range from -5% to +2%, and the mean AO values are negative for the candidate cores. Well-distributed power sharing between the top and bottom halves of the core is illustrated by the low absolute value (-5% to +2%) of the AO and the minor variation over the core lifetime for both cores. In our BPZ core designs, the narrower band of AO variation is an obvious design benefit in an SBF environment, since it contributes to improved uniform axial power distribution around the middle of core axial region (Daing and Kim, 2011, Franceschini and Petrović, 2008, Dai and Cao, 2013).

Fig. 21 shows that the duplex core experiences a higher negative AO than the UO₂ core for most of the cycle. The more negative MTC (as seen in Fig. 15 in Sect. 4.3.1) along with the higher AFF (as shown in Fig. 13a) of the duplex core leads to a bottom-peaked

power distribution compared to the UO_2 core (Lindley et al., 2014, Thomet, 1999). For both candidate cores, the maximum AO values are observed at year ~ 9 (Fig. 21), consistent with the corresponding maximum AFF values (Fig. 13a in Sect. 4.2.2). In addition, the AO gradually increases (towards more positive values) after year 9, and positive AO values are observed at EOL for both cores.

AO values are now analysed at year 9 (most negative AO) and EOL (positive AO) in terms of AO contribution from the radial channels (core cross-section) for the candidate fuel cores.

4.4.1. Negative axial offset contribution

The most negative AO values observed at about year 9 (as seen in Fig. 21) for the candidate fuel cores are explained by the radial channel contributions, as shown in Fig. 22: all the channels/assemblies exhibit negative AO over the core cross-section at this point. It can also be seen that the duplex core exhibits the more negative AO values over the core cross-section (Fig. 22), consistent with the higher AO it experiences (Fig. 21).

Fig. 22. Axial offset (%) over core cross-section for the radial BPZ core LPs at year 9.

It is also important to explain the negative AO contributions in terms of the axial burnup, as shown in Fig. 23. This can also be explained by analyzing the point of maximum AO at about year 9. In respect to the axial burnup effect on the AO, a lower burnup at an axial level in the fuel assemblies relates to a higher concentration of fissile material and vice versa (Lau et al., 2012, Alam et al., 2017b). A higher concentration of fissile material contributes to higher power and more neutrons. It is seen in Fig. 23 that the BPZ arrangements show relatively lower burnup in the lower half of the core for both fuels.

A combination of two effects – (1) lower burnup in the lower regions (leading to a higher concentration of fissile material) (Lau et al., 2012); and (2) lower coolant temperatures in the bottom part of the core (leading to more efficient moderation, allowing more neutrons to

Fig. 23. Axial burnup distribution for the radial BPZ core LPs at year 9.

reach lower thermal energies where fission cross-sections are higher) (Alam et al., 2016a,c) – shift the power from the centre of the core to the bottom region, and contribute to a negative AO. It can, therefore, be said that the duplex core allows more efficient moderation and breeding of ^{233}U , leading to higher AO values than seen for the UO_2 core.

4.4.2. Positive axial offset contribution

The AO gradually increases towards EOL for the candidate fuel cores because of the improved moderation at the top of the core (Fig. 21) (Lau et al., 2012, Alam et al., 2016a,b). Improved moderation along with the lower burnup in the top half of the cores contribute to a positive AO (Lau et al., 2012).

The positive AO values (as shown in Fig. 21) can also be explained by the radial channel contributions at EOL, as shown in Fig. 24. The candidate cores exhibit lower negative AO values in the central assemblies and higher positive AO values in other assemblies, implying that the net AO assembly contribution is positive. Fig. 24 also shows that the UO_2 core exhibits higher (positive) AO values over the core cross-section at EOL, consistent with the higher AO experienced by the UO_2 core at EOL (as shown in Fig. 21).

Finally, it can be concluded that the AO values are consistent with the corresponding AFF values, radial channel contributions, axial burnup and the effect of MTC.

4.5. Shutdown margin & criticality search using control rod

Next, the shutdown margin (SDM) of the control rod assemblies are analyzed to show that the cores have sufficiently large control reactivity, leading to safe shutdown.

The SDM was calculated for a reactor trip from HFP to hot zero power considering no change in xenon worth, with the highest worth rod completely withdrawn from the core. In our 112-assembly marine PWR core, 3 banks of control rods (A, B and C) are used for power maneuvering and 3 other banks (SA, SB and SC) are used for shutdown, as shown in Fig. 25.

Fig. 24. Axial offset (%) over core cross-section for the radial BPZ core LPs at EOL.

Fig. 25. Control rod bank positions.

A total of 36 rod cluster control assemblies each of 16 rods are used. We have chosen boron carbide (B_4C) control rods over hafnium or Ag-In-Cd alloy due to the substantially greater rod worth (Alam et al., 2019c). Typically, the SDM should be more than 1300 pcm over the lifetime of a PWR core (Alam et al., 2016a). We have considered a 10% reduction in the total rod worth (TRW) in calculating ‘available rod worth’ (ARW) (i.e. $0.9 \times [\text{total rod worth} - \text{highest rod worth}]$) in order to accommodate modeling uncertainties. Two reactivity values were deducted from the ARW: the power defect (PD) and the control rod insertion allowance (RIA). The RIA was evaluated at HFP assuming all the shutdown banks (SA, SB and SC) to be fully withdrawn and all the other rod banks (A, B and C) to be inserted to 40% length. Finally, SDM values for both fuel cores at BOL and EOL were determined by considering $SDM = ARW - (RIA + PD)$.

Tables 5 and 6 show that the SDM values are greater than 1300 pcm at BOL and EOL for both cores. It can be seen that, for both cores, the EOL SDM is less than the BOL SDM due to the increased magnitude of the reactivity feedback coefficients (Alam et al., 2016a). In addition, BOL and EOL SDM values are higher for the UO_2 core due to the less negative Doppler power coefficient (DPC) compared to the duplex fuel (Fig. 26) (Salazar et al., 2015). A less negative DPC contributes to a lower PD, which is beneficial for achieving an increased SDM (Salazar et al., 2015).

	Items	Cycle stage	
		BOL	EOL
a	TRW	7321	8709
b	ARW	4682	5846
c	PD	1487	2189
d	RIA	380	1313
b-c-d	SDM	2815	2344

Table 5. Reactivity balance (pcm) for the duplex core with the BPZ arrangement.

	Items	Cycle stage	
		BOL	EOL
a	TRW	7389	8451
b	ARW	4754	5694
c	PD	1263	2050
d	RIA	405	866
b-c-d	SDM	3087	2778

Table 6. Reactivity balance (pcm) for the UO_2 core with the BPZ arrangement.

These results suggest that SDM would not be limiting for the candidate fuel cores. It was not an objective of this study to justify the behaviour of the duplex fuel by proving that SDM cannot be resolved (or difficult to resolve) with UO_2 fuel. Therefore, achieving

Fig. 26. DPC over time for the radial BPZ core LPs.

sufficient SDM was not a concern rather this safety parameter for the candidate fuel cores was used as a comparison.

In addition, it is required to observe the criticality search ($k_{eff} = 1.0$) using control rod banks (Fig. 25) while satisfying the key neutronic safety parameters. Fig. 27 demonstrates the critical control rods positions (as shown in Fig. 25 for maintaining $k_{eff} = 1.0$ over life for the radial BPZ core LPs. For maintaining criticality of the reference cores, only Bank C was moved from the fully withdrawn position, whereas other banks were fully withdrawn, suggested by Yoo et al. (Yoo et al., 2017). Fig. 28 shows that the candidate cores with the BPZ arrangements satisfy the power peaking and reactivity feedback parameters while maintaining $k_{eff} = 1.0$ over life. It can also be confirmed that lower reactivity swing of duplex fuel core can help reduce the insertion depth of control rods.

Fig. 27. Critical control rod (Bank C) positions for maintaining $k_{eff} = 1.0$ over life for the radial BPZ core LPs.

Fig. 28. Power peaking factors and reactivity coefficients over time while inserted Bank C for maintaining $k_{eff} = 1.0$ for the radial BPZ core LPs: (a) RFF and AFF (left axis) and F_Q (right axis); (b) MTC (left axis) and FTC (right axis).

5. Practical considerations for the duplex fuel

The major challenge for the micro-heterogeneous duplex fuel arrangements is to meet the thermal-hydraulic margins since the most severe situation will be in the duplex pellet case where most of the power is generated in the UO₂ part of the fuel pellet. The power peaking problem is observed for the duplex fuel for the Zr case plotted as a function of normalized power along the duplex pellet radius at BOL. Fig. 29a shows that normalized power of duplex pellet (at the UO₂–ThO₂ interface) is about a factor of 1.4 at the BOL and almost 20% higher than the all-UO₂ fuel. It is clear that power peaking can be kept below standard industry limit of 1.5 (Pramuditya and Takahashi, 2013). It is worthwhile mentioning that previous studies (Alam, 2018, Shwageraus et al., 2004) exhibit a normalized power of 2.4 for axially micro-heterogeneous duplex fuel and this higher BOL normalized power UO₂ region results in an unacceptably high fuel temperature. In order to avoid this issue, our radial micro-heterogeneous duplex fuel is designed in such a way that ThO₂ region is ~25% of the UO₂ region and normalized power is limited to 1.4, which is an obvious design improvement considering the practical perspective. A detailed discussion of the radial micro-heterogeneous duplex fuel design and analyses can be found in the first author's PhD research (Alam, 2018).

In order to confirm that all the thermal-hydraulic constraints are satisfied for the duplex fuel, 3D neutronic/thermal-hydraulic coupling of hybrid monte carlo MONK with sub-channel analysis COBRA-EN (Basile, 1999) code for hot channel analysis (Alam, 2018, Alam et al., 2019d) has been performed to evaluate key TH parameters such as: minimum departure from nucleate boiling ratio (MDNBR), heat flux, cladding, inner surface and fuel centreline temperatures, and pressure drop. The power imbalance (Fig. 29a) between the uranium and thorium regions can create several design challenges, such as higher fission gas release and cladding temperature gradients. It is required that there be no melting of the fuel and cladding. Since oxide-based fuels have been specified, it is important to recognise that they release non-negligible amounts of fission gas (Todreas and Kazimi, 2012). If this gas release is not managed or restricted, it can pressurize and even burst the fuel pin. Generally, fission gas release for a PWR should be less than 5%, and can be kept lower by limiting the average

Fig. 29. (a) Normalized power density distribution for duplex fuel (at the UO₂-ThO₂ interface) and all-UO₂; (b) Radial temperature distribution of fuel rod in the hottest channel of the duplex core.

fuel temperature to 1600 K and cladding surface temperature to 1000 K (for Zircaloy) in steady-state operation. The average fuel temperature constraint is considered to be more limiting than imposing a peak fuel centreline temperature of 2800 K (Todreas and Kazimi, 2012). Hottest channel analyses show that fuel centreline and cladding temperatures for the duplex fuel core are safely below their limiting values, as shown in Fig. 29b. Our study confirmed that thermal-hydraulic design requirements for the duplex fuel can be met and there will be no melting in the UO₂ region of the duplex pellet. Since thermal-hydraulic design requirements are met by a good margin, it can, therefore, be expected that other issues (e.g. hydriding of cladding, fission gas release, and pellet/cladding mechanical interactions) arising from the large temperature gradients in the UO₂ part of the fuel pellet can be avoided. It is, thus, proven that our proposed duplex fuel resolves it in a practical way without making any other unrealistic assumptions (e.g. astronomic burnup, peak temperatures or power peaking). These issues are out of the scope of this paper and detailed hot channel thermal-hydraulic analysis can be found in the first author's PhD research (Alam, 2018).

However, as addressed in this paper's Part I companion (Alam et al., 2019c) that it will be many years before civil marine propulsion achieves widespread commercialisation, and rapid improvements in materials and fabrication lead to permitting ever-higher burnup (Peakman, 2014, Alam, 2018, Otto, 2013). Therefore, higher local burnup is not considered as a constraint (or limiting factor as with current commercial reactors) for the design of long life core, instead, it is an objective. It is worthwhile addressing that for the purpose of this study, it is assumed that suitable materials and technology that can withstand prolonged core life and high burnup will be available for future use for the long life marine core (Zainuddin, 2015, Shwageraus and Feinroth, 2011, Andrews et al., 2014, Sukjai and Kazimi, 2015).

Fuel performance issues for the duplex fuel are worth investigating the future. It is also worth addressing that the main objective of this paper is to evaluate the neutronic feasibility of the duplex fuel in a SBF environment. We have considered the issues of the associated

complexities (peak temperatures, local burnup) of duplex fuel in the previous study (Alam, 2018) and these issues are, however, out of the scope of this paper since the scope of these papers are limited to the neutronic study.

6. Conclusions

. The idea of this paper is not just to obtain the satisfactory neutronic performance, but to successfully “*open the option*” of designing the proposed SBF, marine SMR core with both the duplex and UO_2 fuel cores. In this regard, the main findings are:

- One of the most important points that can be drawn from these studies is that thorium-based duplex cores consistently achieve longer core life, lower reactivity swing and higher initial reactivity suppression than UO_2 cores in the whole-core environment. The higher initial reactivity suppression and relatively smaller reactivity swing of the duplex core makes the task of reactivity control through BP design easier.
- This study uses homogeneous and radial-zoning arrangements for the whole-core analysis. The homogeneous LPs produce unsatisfactory power peaking values, well above the permissible limit throughout most of the core life, suggesting that radial-zoning arrangements are required to address the power peaking problem. The burnable poison radial-zoning arrangements proposed enable both candidate fuel cores to achieve the required core lifetime with acceptable through-life RFF variation.
- The IFBA poisons are selected in such a way that power peaking factors (radial, axial, etc.) are well below the safety limits and it can, therefore, be expected that other issues (e.g. hydriding of cladding, higher local burnup, fission gas release, and pellet/cladding mechanical interactions) can be avoided.
- The duplex core experiences a higher maximum axial offset value than the UO_2 core for most of the core life. The axial offset values are consistent with the corresponding axial form factor values, radial channel contributions, axial burnup and moderator temperature coefficient.
- The reactivity coefficient (MTC and FTC) values are generally considerably more negative for the duplex core than those for the UO_2 core. However, MTC is slightly more negative in the UO_2 core later in life. Both fuel cores satisfy the minimum shutdown margin requirement of 1300 pcm, suggesting that SDM would not be limiting.
- Considering the whole-core performance, ultimately the most successful design is the duplex core with BP zoning. This core exhibits a reactivity swing of less than 2500 pcm, while obtaining a core life of ~ 16 years. For all these reasons, we plan to continue our studies of thorium-based micro-heterogeneous duplex fuels for civil marine propulsion applications.

We have considered the issues of the associated complexities (peak temperatures, local burnup) of duplex fuel in the previous study (Alam, 2018) and these issues are, however, out of the scope of this paper since the scope of these papers are limited to the neutronic study of assembly-level. Furthermore, higher local burnup is not considered as a constraint (or limiting factor as with current commercial reactors) for the design of long life core, instead, it is an objective. For the purpose of this study, it is assumed that suitable materials and technology that can withstand prolonged core life and high burnup will be available for future use for the long life marine core. Future works will focus on the following aspects:

- One of the important safety aspects is fuel performance issues which needs to be investigated in the future. It is recommended that fuel performance analyses for the proposed high burnup core be undertaken to verify the integrity of the fuel rods including Integral fuel rod performances and fission gas release (FGR), pellet-clad interaction (PCI) phenomena caused by excessively high rod internal gases and pellet swelling; and oxide thickness (corrosion).
- From a fuel manufacturing perspective, while the use of IFBA burnable absorber in UO_2 fuel is a consolidated practice in Westinghouse, its viability for the duplex rods needs to be confirmed in future studies.
- Since power density is an important figure of merit and characterizes design performance, future work will focus on the neutronics and coupled neutronic/thermal-hydraulic analysis of high power density marine SMR candidate core designs.
- Xenon effects, power changing rod and transient power allowance will be investigated in the future study. Xenon transients complicate startup, shut down and power level changes in power reactors. Xenon transients are usually overcome by adjusting chemical shim (soluble boron). Since most power reactors operate at fixed power levels, xenon transients are generally problematic in unplanned outages. However, for a marine SMR core, it will be practically necessary to alter the power level of the core as the ship enters port to load/unload cargos. Therefore, it will be worthwhile to observe the behaviour of the proposed marine core with the changes in power level (power cycling due to perturbations in the flux within a reactor) by varying temperature to accommodate demand change. The main problem relating to xenon-induced oscillations is its effect on local power peaking, which is worth performing in the future from the fuel performance (i.e. fuel failure) perspective.
- The objective of this study was to obtain the maximum attainable core lifetime while achieving the lowest/minimum reactivity swing. Since one of the major objectives was to obtain the lowest possible through-life reactivity swing, a serious effort was to reduce the reactivity values (while $k_{eff} > 1.0$) over the core lifetime. While minimizing the reactivity swing, we didn't consider the nuclear data uncertainties, which is worth investigating in the future.

- In combination with a new fuel type (i.e. duplex Th-U pellet) for prototype SMR for civil nuclear marine propulsion, authors didn't consider from the licensing/manufacturing point of view since this study was a complete neutronic feasibility of alternative fuel platform. This issue is also worth investigating in the future.

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