# NEUTRONIC DESIGN OF A PWR FUEL ASSEMBLY WITH ACCIDENT TOLERANT-COMPOSITE FUEL USING SRAC CODE

VAN KHANH HOANG

*Institute for Nuclear Science and Technology, 179 Hoang Quoc Viet, Nghia Do, Cau Giay, Ha Noi*

*E-mail: hvkhanh21@gmail.com*

**Abstract:** For the future of nuclear power, the design and development of an economical, accident tolerant fuel (ATF) for use in the current pressurized water reactors (PWRs) are highly desirable and essential. It is reported that the composite fuels are advantageous over the conventional $UO_2$ fuel due to their higher thermal conductivities and a higher uranium densities. Due to higher uranium densities of the composite fuels, the use of composite fuels would lead to the significant increase of discharged burnup, thereby enhancing fuel cycle economy compared to that of the $UO_2$ fuel. The higher thermal conductivities of composite fuels will increase the fuel safety margins. For implementation of the accident tolerant fuel concept, this study also investigates on the replacement of the conventional Zircaloy-4 cladding with SiC to minimize the hydrogen production due to interaction of water with cladding at high temperature. In the present work, neutronic investigation of the composite fuels for a PWR has been conducted in comparison with that of the conventional $UO_2$ fuel. Numerical calculations have been performed based on a lattice model using the SRAC code and JENDL-4.0 data library. Various parameters have been surveyed for designing a fuel with the $UO_2$ and composite fuels such as U-235 enrichment, pin pitch. In order to reduce the excess reactivity, Erbium was selected as a burnable poison due to its good depletion performance. The temperature coefficients including fuel, coolant temperature reactivity coefficients, and both the small and large void reactivity coefficients are also investigated. It was found that it is possible to achieve sufficient criticality up to 100 GWd/tHM burnup without compromising the safety parameters. Further analysis of the performance of the $UO_2$ and composite fuels in a full core model of a PWR is being conducted.

**Keywords:** $UO_2$ fuel, composite fuels, PWR assembly, neutronic analysis.

## I. INTRODUCTION

### I.1. Motivation for consideration of alternate fuel and cladding concepts

Nearly all nuclear fuel made with uranium dioxide ($UO_2$) pellets and zirconium-based cladding has been successfully used for all power reactors, [1], [2], [3], [4]. The conventional fuel, $UO_2$, is stable, and has a high melting point 2850 °C, [5]. However, the $UO_2$ has a rather low thermal conductivity, 7 W/m·K at 573 K [5], which decreases with increased temperature and burnup, leading to significant temperature gradients within the ceramic pellets, and would result in thermal stress and potential cracking, [6]. The zirconium alloys, having very low neutron capture cross-section, are used in reactor design to support and contain the fuel pellets, as well as containing fission products. On the other hand, zirconium is vulnerable to oxidation in steam at elevated temperatures. Once an energetic exothermic, hydrogen producing reaction is occurred, it would lead to early cladding failure.

The accident at Fukushima Daiichi in March 2011 and the Three Mile Island accident in 1979 show that the current fuel is not adequate and sufficient for the beyond design basis accidents. These beyond design basis accidents would occur at somewhat higher frequencies than previously predicted, and that the financial liabilities of such accidents can cripple a utility [7]. Following the Fukushima Daiichi nuclear accident in 2011, the world nuclear fuel R&D activities have shifted to pursue new fuel materials that provide significant increases in the time for the reactor operator to respond to unforeseen events before significant releases of the fuel materials and fission products occur [8], [9]. The accident tolerant fuel (ATF) systems have attracted significant attention to mitigate the consequences of a future severe accident, by better retaining fission products and/or providing operators more time to implement emergency measures of commercial light water reactors. The desired ATF needs ~~to~~ against a loss of cooling for a considerably long period, and improve fuel performance while enhancing fuel safety at normal operation. Any developed accident tolerant fuel (ATF) products would improve operating cost, and enhance safety for commercial application. It is described in the previous studies, [10], [11], the development of accident tolerant fuel/cladding systems are focused on:

(1) Improve or replace the ceramic oxide fuel: aims are to increase uranium loading; to increase thermal conductivity; and to extend fuel cycles due to higher energy content of fuel without higher enrichment cost.

(2) Modify or replace the zircaloy cladding: goals are to achieve improved oxidation resistance, including application of coating layer; to increase fuel rod failure temperature, resistance to thermal cycling and irradiation induced degradation; to decrease thermal neutron cross section for cladding; to increase resistance to expansion and warping; to increase thermal conductivity; and to reduce rate of oxidation.

According to the previous investigations, [11], [12], the silicon carbide fiber-reinforced SiC matrix ceramic composites (SiC/SiC) is a potential cladding material due to their low thermal neutron absorption cross section, retention of strength up to very high temperatures, good radiation resistance, and good oxidation resistance in air and steam up to temperatures of at least 1600 °C. The study in [13] shows that because of a low neutron absorption, the SiC cladding material could meet lifetime requirements even with a 0.1% reduction in enrichment. Regarding the nuclear fuel, high density fuels including uranium-molybdenum fuels, uranium nitride fuels, uranium carbide fuels, and uranium silicide fuels are being considered for ATF solutions. Uranium mononitride (UN) fuel forms has a long historical application for power reactors [13]. Due to a high uranium loading and high thermal conductivity, the uranium mononitride is desirably used as a nuclear fuel [14], [15]. However, the reactivity of UN with water has been a concern in nuclear reactor applications [14], [15]. For this reason, uranium sesquisilicide ($U_3Si_2$) and $UO_2$ have been combined with the UN as a composite fuels to provide a protective barrier. It is reported in previous study, [16], that a fuel composed of UN and $U_3Si_2$ will significantly improve the fuel's thermal conductivity over $UO_2$ and increase uranium density and therefore enhancing fuel loading. The studies, [14], [17], also show that the $UO_2$-UN composite fuels are advantageous over the conventional $UO_2$ fuel due to its higher thermal conductivity and higher uranium density. In particular, the $UO_2$-UN composite fuel with 33 vol. % of $UO_2$ has a higher uranium density of about 13% and a higher thermal conductivity of about 100% at 800°C compared to the $UO_2$ fuel.

The classic approach to generate nuclear energy is to use fuel made with the uranium dioxide ($UO_2$) pellets and zirconium-based cladding. This method is successfully implemented on industrial scale level for power reactors. Usually, the fuel concept enrichment requires uranium with U-235 fraction less than 20% (low enrichment uranium, LEU). This LEU is not treated as a nuclear material for direct use in weapon manufacturing, therefore it gives an upper limitation for challenging the uranium fuels for the long-life core. The approach adopted for this study is to use conventional fuel, $UO_2$, and composite fuels, (including UN- 30 wt. % $U_3Si_2$ [16] and 33 vol. % $UO_2$-UN [17]), combining with SiC cladding material to estimate the attainable burnup for a wide range of combinations of lattice pitch, P (referred to as "geometries") of interest and for a number of different uranium enrichments for the long-life core with once-through burning.

### I.2. Study objective

The primary objective of the present study is to estimate the attainable burnup, 100 GWd/tHM burnup without compromising the safety parameters, for a wide range of combinations of lattice pitch, P, of interest and for a number of different uranium enrichments. The fuel cell is~~are~~ made with $UO_2$, composite fuels, (including UN- 30 wt. % U3Si2 referred to as UNSi and 33 vol. % $UO_2$-UN referred to as UNO), and SiC cladding. The attainable burnup is the maximum burnup of the fuel discharged from a once-through burning fuel subjected to negative reactivity coefficients during the fuel life. Four reactivity coefficients are considered ~~those associated with~~including the fuel temperature, coolant temperature, small (5%) void and large (90%) void. An infinite multiplication factor (k-inf) value at the end of cycle (EOC) is conservatively assumed to be 1.05 for the lattice investigations.

### I.3. Study scope

Two types of composite fuels are considered - UNSi and UNO. As far as it is known, these composite fuels have been fabricated, even though laboratory experience exists. The material properties of these composite fuels have been extensively studied and are summarized in some companion papers, [5], [14], [17]. It is shown that these composite fuels were tested and found suitable for reactor operation.

The first part of the current study is devoted to a scoping study of PWR unit cell that investigated a wide range of combinations of lattice pitch~~ (P - hereby referred to as "geometries"),~~ and different uranium enrichments of different fuel types including $UO_2$, UNSi, and UNO. The aim of this investigation is to determine the neutronically attainable burnup for each of the geometries and the different fuel compositions, subjected to negative reactivity coefficient constraints. The reactivity coefficient constraints are all negative for coolant temperature coefficient of reactivity (CTC), prompt fuel temperature coefficient of reactivity (FTC), and the reactivity effect of both small voiding 5 % (SVRC) and large voiding 90 % (LVRC) of the coolant. For the examinations with high U-235 enrichment fuel, it would lead to an initial high reactivity excess. It opens a necessary application of burnable poisons (BP) to reduce initial high reactivity excess as in previous studies [18], [19], [20], [21], [22]. Among these mentioned researches, it is found that selected Erbium as a most promising candidate for the long-life core with once-through burning fuel. Thus, in this study, the excess reactivity is compensated by adding burnable poisons of Erbium.

The second part of the study is devoted to a scoping study of $UO_2$, UNSi, and UNO fueled PWR assembly. A detailed neutronic analysis of the maximum burnup fuel offering a minimum uranium enrichments and no expanding beyond the present day fuel cycle technology that the fuel is burnt up to 100 GWd/tHM [23] is presented.

## II. METHODOLOGY

### II.1. Analysis tools and calculational model

The calculations for this study were performed with the SRAC code system [24] applied to the lattices configuration using the Pij module with~~derived~~ 16 energy group libraries generated using the JENDL-4.0 [25]. In this paper, neutronic study investigation is limited to infinite pin cell and assembly level calculation with material, temperature, and fuel cell characteristics listed in Table 1 and Fig. 1.

**Table 1. Parameters of the fuel cells and assembly**

| Parameters | Reference | New design |
|---|---|---|
| Fuel diameter | 0.8192 | 0.8192 |
| Clad inside diameter, [cm] | 0.8357 | 0.8357 |
| Clad outside diameter, [cm] | 0.9500 | 1.0357 |
| Lattice pitch, P, [cm] | 1.2598 | Varialbles |
| P/D, [-] | 1.3262 | Varialbles |
| Equivalent pin pitch, [cm] | -- | 1.3118 |
| Equivalent P/D, [cm] | -- | 1.2666 |
| Rod array, [-] | 17x17 | |
| Assembly pitch, [cm] | 21.5 | Varialbles |
| Linear heat rate, [W/cm] | 176.5 | |
| Average coolant temperature in core, [K] | 576.5 | |
| System pressure, nominal, [Mpa] | 15.5 | |
| Average temperature for fuel, [K] | 950.0 | |
| Average temperature for clad, [K] | 607.0 | |

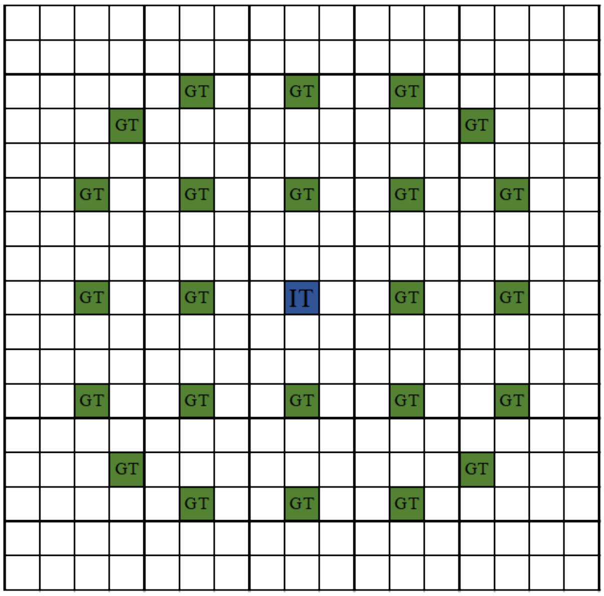


**Fig. 1. Layout of fuel assembly (GT: guide thimbles; IT: instrumentation thimble; others: Fuel rods**

The reference geometry and specific power assumed for fuel cells are given in Table 1. The data for the reference unit cell correspond to the Westinghouse PWR fuel design that loaded fuel of the 4.45 % wt. U-235 enrichment, [26]. The typical Westinghouse PWR fuel assembly (FA) of 17x17 array, comprises of 289 total lattice locations, of which 24 are for control rod and 1 in the center is instrument thimble, [27]. In simplified fuel assembly calculational models, no water reflector is modeled and spacer grid effects are neglected as well. As described in the previous section, in order to enhance strength and ductility ATF cladding mitigating~~e~~ against severe, SiC is selected as the cladding material [28]. For the high burnup, i.e., long-life core, especially with a burnable poison of Erbium, it is reasonably expected a hardener neutron spectrum and higher pressure of gaseous fission products compared to the reference case. Thus, for the high burnup, up to 100 GWd/tHM, the fuel would experience in a condition of high porosity. In this study, the porosity of the fuel is conservatively chosen of 15 %.

## II.2. Calculated characteristics

The investigations in this study are U-235 enrichment, ranging from 5 to 20 % and lattice pitch-to-diameter ratio (P/D) ranging from 1.05 to 2.65. Calculated for each of the cases studied are the achievable once-through burnup and the reactivity coefficients along the fuel life without soluble boron in the water. The achievable burnup is assumed basing on combining of negative reactivity coefficients and infinite multiplication factor (k-inf) value at the end of cycle (EOC) is 1.05. For the fuel assembly model, there is no water reflector is modeled and spacer grid effects are neglected as well.

The analysis of each fuel model is included the calculation of the achievable burnup and of reactivity coefficients of a once-through burning fuel. The reactivity coefficients examined are included~~ing~~ the fuel temperature coefficient of reactivity (FTC), the coolant temperature coefficient of reactivity (CTC), and the small and large void coefficients of reactivity (SVRC and LVRC). The FTC is evaluated by increasing the fuel temperature by 100 K - from 950 to 1050 K. For the CTC the water temperature is increased from the nominal value of 576.50 K by 10 K to 586.50 K. In case of void coefficients, both small and large, the temperature of

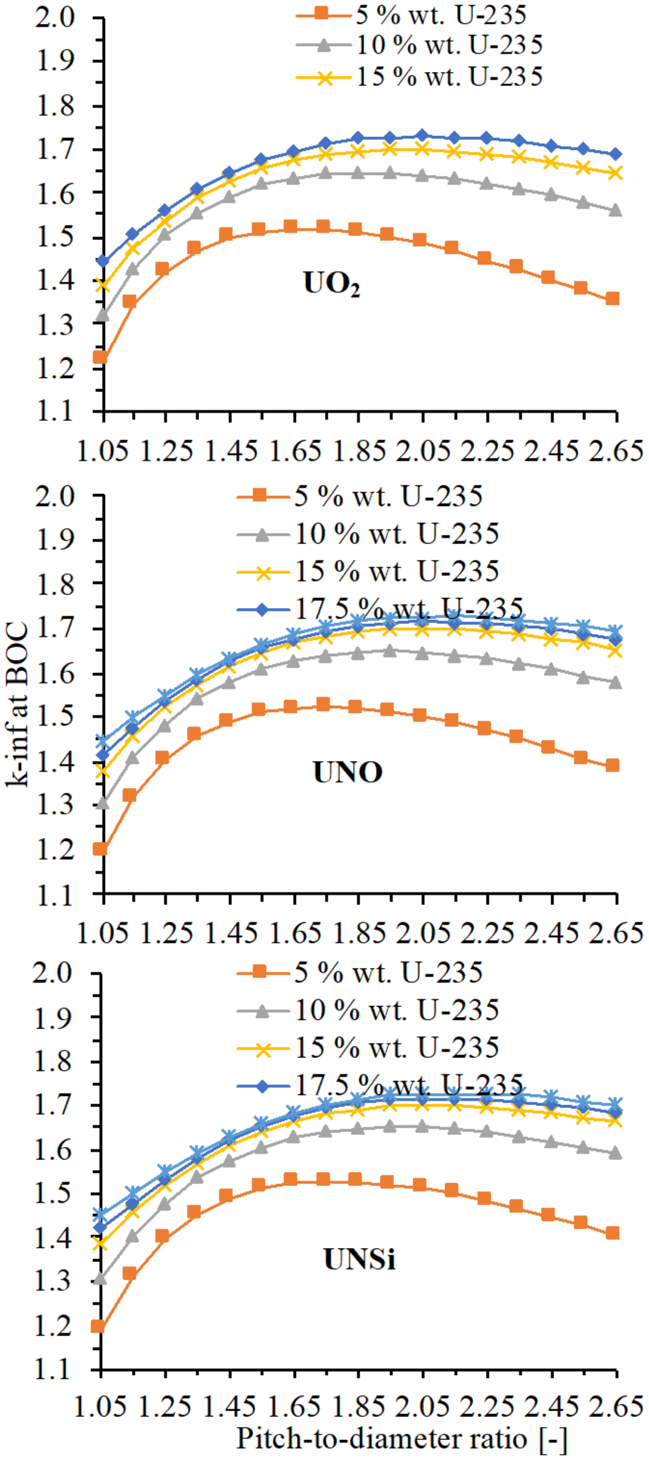


Fig. 2. k-inf at BOC as a function of P/D.

the water is left unchanged while the density of the moderator is reduced by, respectively, 5 % or 90 %.

## III. PARAMETRIC STUDY RESULTS

### III.1. Single fuel cell analysis

The parametric study is undertaken to estimate the effect of P/D on the attainable discharge burnup. The attainable discharge burnup is assumed to be subjected to negative reactivity coefficient constraints and k-inf value at the EOC is 1.05. The pin pitch is considered as a design variable. The soluble boron in the coolant, water, is not accounted for in this study. The burnable poison, Erbium, ~~is~~ doped into the fuel, helps to reduce the high excess reactivity.

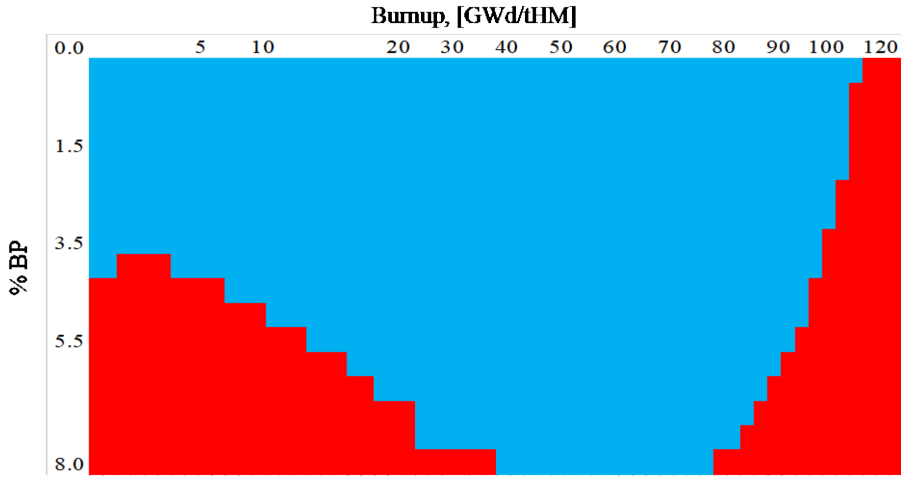


Fig. 3. Design space of $UO_2$ fuel cell loaded 15 % U-235 enrichment fuel with BP

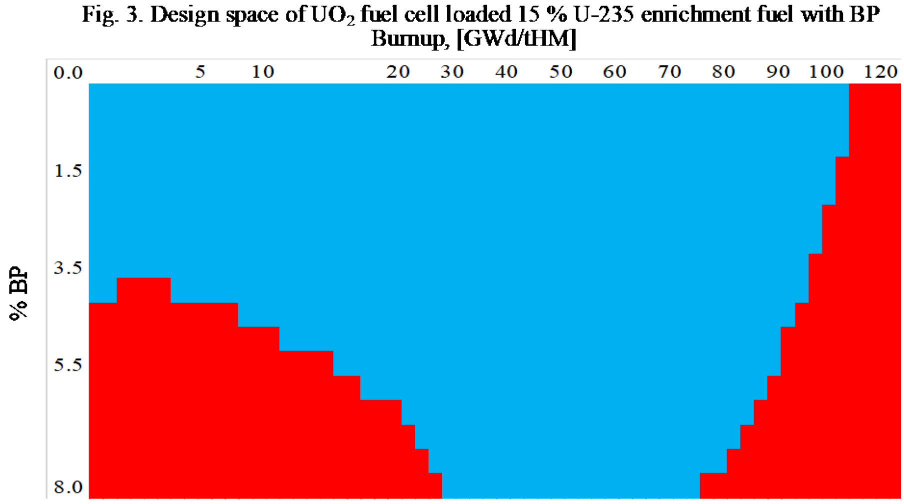


Fig. 4. Design space of UNO fuel cell loaded 15 % U-235 enrichment fuel with BP

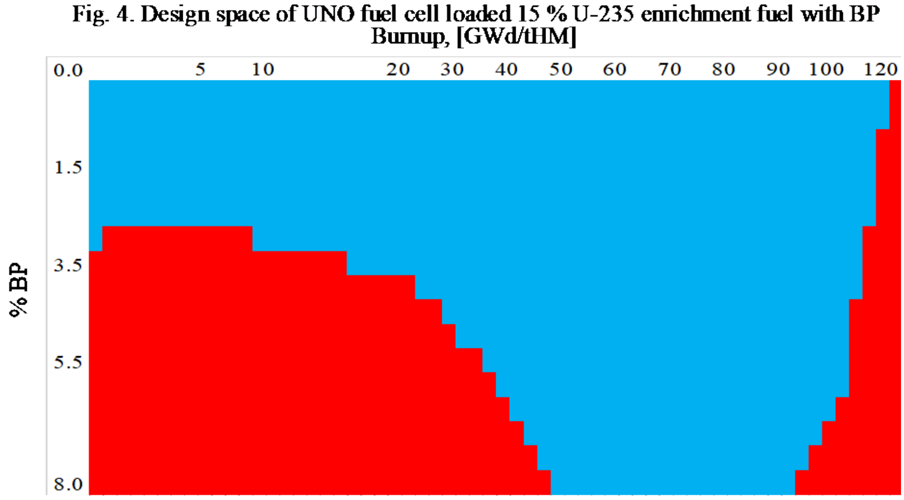


Fig. 5. Design space of UNSi fuel cell loaded 15 % U-235 enrichment fuel with BP

Table 2. Fuel cell selected characteristics versus P/D and U-235 enrichment.

| Enrichment | Max. burnup, [GWd/t] | | | P/D for burnup ≥ 100 GWd/t | | | k-inf at BOC, [-] | | |
|---|---|---|---|---|---|---|---|---|---|
| [wt. %] | UNO | UNSi | $UO_2$ | UNO | UNSi | $UO_2$ | UNO | UNSi | $UO_2$ |
| 4.45 | -- | -- | 30.0 | -- | -- | -- | -- | -- | 1.3950 |
| 5.00 | 40.0 | 40.0 | 40.0 | -- | -- | -- | -- | -- | -- |
| 10.00 | 80.0 | 82.5 | 80.0 | -- | -- | -- | -- | -- | -- |
| 15.00 | > 120 | > 120 | > 120 | 1.25-1.95 | 1.35-1.95 | 1.25-1.85 | 1.5224-1.6984 | 1.5693-1.6984 | 1.5368-1.6959 |
| 17.50 | > 120 | > 120 | -- | 1.15-1.95 | 1.15-2.05 | -- | 1.4789-1.7151 | 1.4779-1.7156 | -- |
| 20.00 | > 120 | > 120 | > 120 | 1.15-2.05 | 1.25-2.05 | 1.15-1.95 | 1.4977-1.7268 | 1.5474-1.7267 | 1.5062-1.7265 |

Table 2 summarizes the selected characteristics calculated for pin cells fuelled with various different initial fuel compositions having different P/D values. Increasing the U-235 enrichment results in increasing of both maximum achievable burnup and k-inf value at the BOC as given in Table 2 and Fig. 2. Higher U-235 enrichment in fuel gives larger P/D ranging to achieve the high burnup. This is because of the increase of fissile isotope, U-235, in the heavy metal inventory. It is found that the fuel of ≥15 % wt. U-235 enrichment is potential for a long-life core design. In order to enhance economy of fuel usage and minimize the high excess initial activities, the fuel of 15 % wt. U-235 enrichment is selected for the $UO_2$, UNO fuel types, and 17.5 % wt. U-235 enrichment is chosen for the UNSi fuel form. The required P/D ranging is from 1.25 to 1.85, 1.25 to 1.95, and 1.15 to 2.05 for fuel cell with, respectively, 15 % wt. U-235 of $UO_2$, 15 % wt. U-235 of UNO, and 17.5 % wt. U-235 of UNSi. The potential maximum achievable burnup would reach up to 120 GWd/tHM as shown in Table 2.

As mentioned in the previous section, the main idea behind the present paper is to use LEU as a once-through burning and no expanding beyond the present day fuel cycle technology that the fuel is burnt up to 100 GWd/tHM. Therefore, the P/D = 1.27, belonged to the required P/D ranges, is preferably chosen option in following investigations. Following the logic of the present paper described in the previous section, the high initial reactivity excess is expected to be suppressed by adding burnable poison of Erbium. In this study, the BP is assumed to be homogeneously mixed to the fuel.

For the identified fuel pin cells (that of 15 % wt. U-235 of $UO_2$, 15 % wt. U-235 of UNO, and 17.5 % wt. U-235 of UNSi, and P/D = 1.27), Fig. 3, Fig. 4, and Fig. 5 sketch the design space of the fuel pin cells loaded different fuel types with BP addition. The possible designs are colored in blue that fulfill all criteria including reactivity safety parameters, moderator temperature coefficient, void coefficients, and Doppler coefficient along fuel cycle. It is found that, with the fuel of ≤ 1.5 % BP addition, even though the fuel cells are fulfilled all safety criteria, the k-inf values at some beginning burnup stages are higher than that of the reference fuel cell, k-inf being equal to 1.3950 as seen in Table 2. For the fuel of ≥ 3.5 % BP addition, it

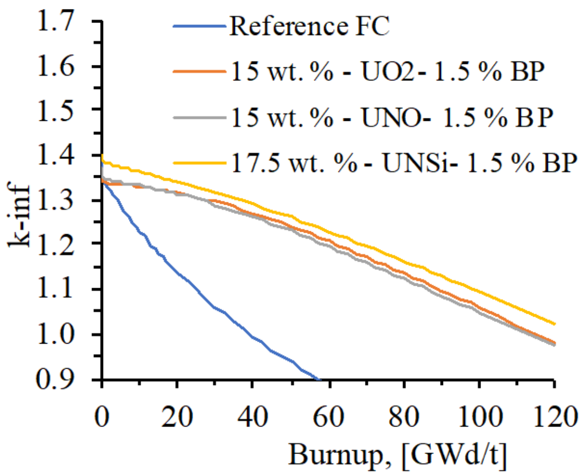


Fig. 6. K-infinity variation with burnup.

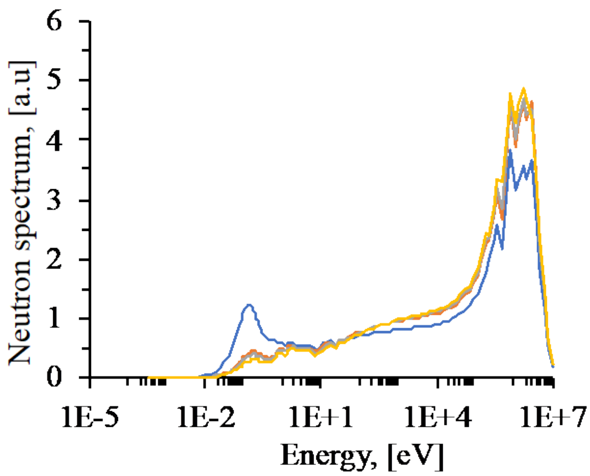


Fig. 7. Neutron flux per lethargy at BOC.

is not preferable for designing because of positive feedback reactivity coefficients for both the $UO_2$ and UNO fuel type. Meanwhile the fuel of ≥ 2.5 % BP addition to the fuel of UNSi is

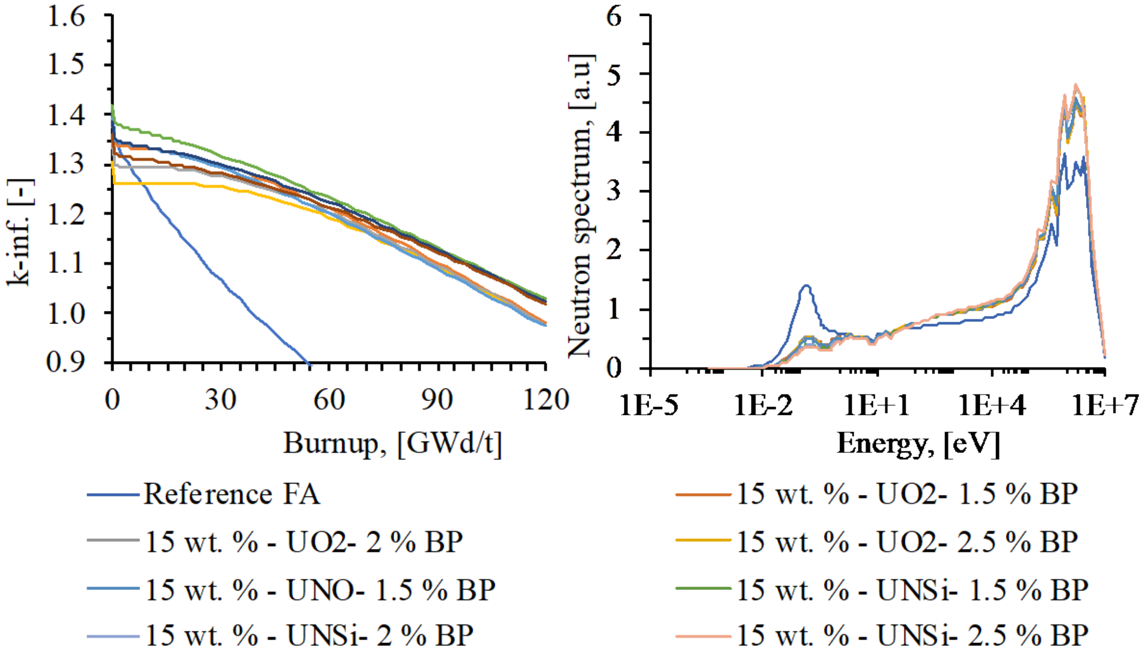


**Fig. 8. K-infinity of FAs versus burnup.** **Fig. 9. Neutron flux per lethargy of FAs at BOC.**

unreasonable for the designing.

Figure 6, Fig. 7 show k-inf evolution as a function of burning time and the BOC neutron spectrum, respectively, for the some outstanding cases examined. It is clear to see that the neutron spectra of the preferable design fuel cells are all harder than that of the reference fuel cell. The higher percentage of BP addition in fuel pellet is, the harder neutron spectrum of the fuel cell is, as shown in Fig. 6. This is because of the BP material strongly absorbs thermal neutrons [18], [19], [20], [21], [21].

### III.2. Fuel assembly analysis

**Table 3. Fuel cell and fuel assembly selected characteristics for various fuel compositions.**

| % Er2O3 | 15 wt. % - UO2 k-inf at BOC Fuel cell | FA | Max. burnup [GWd/t] | PPF | 15 wt. % - UNO k-inf at BOC Fuel cell | FA | Max. burnup [GWd/t] | PPF | 17.5 wt. % - UNSi k-inf at BOC Fuel cell | FA | Max. burnup, [GWd/t] | PPF |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 0.0 | 1.5568 | 1.5805 | -- | -- | 1.5394 | 1.5639 | -- | -- | 1.5419 | 1.5653 | -- | -- |
| 0.5 | 1.4770 | 1.5015 | -- | -- | 1.4693 | 1.4939 | -- | -- | 1.4847 | 1.5076 | -- | -- |
| 1.0 | 1.4145 | 1.4389 | -- | -- | 1.4138 | 1.4380 | -- | -- | 1.4387 | 1.4607 | -- | -- |
| 1.5 | 1.3631 | 1.3870 | 100.0 | 1.090 | 1.3678 | 1.3912 | 100.0 | 1.090 | 1.4001 | 1.4212 | 110.0 | 1.091 |
| 2.0 | 1.3194 | 1.3426 | 100.0 | 1.090 | 1.3285 | 1.3511 | 97.5 | 1.091 | 1.3668 | 1.3870 | 110.0 | 1.091 |
| 2.5 | 1.2814 | 1.3038 | 100.0 | 1.091 | 1.2941 | 1.3159 | 97.5 | 1.092 | 1.3375 | 1.3568 | 110.0 | 1.091 |
| 3.0 | 1.2478 | 1.2694 | 97.5 | 1.091 | 1.2636 | 1.2845 | 95.0 | 1.092 | 1.3112 | 1.3296 | -- | -- |
| 3.5 | 1.2175 | 1.2382 | 97.5 | 1.092 | 1.2360 | 1.2560 | 92.5 | 1.092 | 1.2873 | 1.3048 | -- | -- |
| 4.0 | 1.1899 | 1.2098 | 95.0 | 1.092 | 1.2108 | 1.2300 | 95.0 | 1.092 | 1.2653 | 1.2821 | -- | -- |
| 4.5 | 1.1645 | -- | -- | -- | 1.1875 | -- | -- | -- | 1.2449 | -- | -- | -- |
| 5.0 | 1.1410 | -- | -- | -- | 1.1659 | -- | -- | -- | 1.2259 | -- | -- | -- |
| 5.5 | 1.1191 | -- | -- | -- | 1.1456 | -- | -- | -- | 1.2080 | -- | -- | -- |
| 6.0 | 1.0984 | -- | -- | -- | 1.1266 | -- | -- | -- | 1.1911 | -- | -- | -- |
| 6.5 | 1.0790 | -- | -- | -- | 1.1085 | -- | -- | -- | 1.1750 | -- | -- | -- |
| 7.0 | 1.0605 | -- | -- | -- | 1.0914 | -- | -- | -- | 1.1596 | -- | -- | -- |
| 7.5 | 1.0430 | -- | -- | -- | 1.0751 | -- | -- | -- | 1.1449 | -- | -- | -- |
| 8.0 | 1.0263 | -- | -- | -- | 1.0594 | -- | -- | -- | 1.1308 | -- | -- | -- |

Depletion analysis of the fuel assemblies made with composite fuels and SiC clad is carried out against standard operating conditions and other parameters of the typical Westinghouse PWR fuel assembly. The burnup analysis fuel assembly is carried-out up to 120

GWd/tHM. The identified fuel pin cells (that of 15 % wt. U-235 of $UO_2$, 15 % wt. U-235 of UNO, and 17.5 % wt. U-235 of UNSi, and P/D = 1.27), achieved as results in the previous section are used for fuel assembly investigation. The initial k-inf value of the reference fuel assembly, 1.4205, is chosen as the upper value of initial criticality to be controlled for other fuel designs.

The analysis results are summarized in Table 3. The gray colored numbers indicate the companion designs that those k-inf values are higher than controlled value of 1.4205 or feedback reactivity coefficients are positive. It is clear to see that it is possible to use the $UO_2$, and composite fuels in long-life core with once-through burning fuel, up to 100 GWd/tHM burnup without compromising the safety parameters. The required BP addition to fuel is 1.5 to 2.5 % for both $UO_2$ and UNSi fuel type. Regarding the UNO fuel type, the required BP addition to fuel is 1.5 % for the once-through burning fuel with the target burnup of 100 GWd/tHM.

The pin-power peaking factor (PPF) of all the proper fuel assembly designs are less than 1.10 at the begin of cycle (BOC), and are all higher than that of the reference assembly (1.068). Figure 8, Fig. 9 show k-inf evolution as a function of burning time and the BOC neutron spectrum, respectively, of the proper fuel assembly designs. The maximum k-inf over cycle of the new designs are comparable to that of the reference assembly at ceiling enrichment of 4.45 wt. % U-235 of $UO_2$ fuel. This ensures that it is possible control core reactivity once loading the new fuel assembly design into the conventional core. The neutron spectrums of the new fuel assembly designs are all harder than that of the reference fuel assembly but no any effects on safety.

## IV. CONCLUSIONS

This paper presents the neutronic analysis of fuel design for a long-life core in a pressurized water reactor made composite fuels, (including UN- 30 wt. % U3Si2 and 33 vol. % $UO_2$-UN), and SiC cladding in comparison to the uranium oxide fuel $UO_2$. It is found that use of the fuel of 15 % wt. U-235 of $UO_2$, 15 % wt. U-235 of UNO, and 17.5 % wt. U-235 of UNSi, with P/D = 1.27 and 1.0 to 2.5 % of Erbium as burnable poison addition makes it possible to design a PWR fuel that achieves high burnup. In addition, using SiC as cladding material would enhance strength and ductility ATF cladding mitigate against severe accidents. The fuel temperature coefficient of reactivity and both small and large void reactivity coefficients of the fuel designs are negative along fuel cycle with the concerned burnup target, 100 GWd/tHM burnup without compromising the safety parameters.

In the future study, this preliminary study would be refined and extended including full-core coupled neutronic-thermal-hydraulic analysis, stability analysis, transients and accidents analysis, as well as economic analysis. Furthermore, how to make use of the once-through burning fuel for energy production with employing fuel reprocessing would be considered in further study.

## ACKNOWLEDGMENT

This research is funded by Vietnam National Foundation for Science and Technology Development (NAFOSTED) under grant number 103.04-2019.37.

Help provided by Dr. Tran Hoai Nam of the Institute of Fundamental and Applied Sciences, Duy Tan University, Ho Chi Minh city, Vietnam is highly appreciated. Author would like to express gratitude to Dr. Pham Nhu Viet Ha of the Institute for Nuclear Science and Technology, VINATOM, Vietnam for giving comment and suggestion.

The calculations in this work have been done on the VINATOM - HPC system.

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# TÍNH TOÁN VẬT LÝ CHO BÓ NHIÊN LIỆU LÒ PHẢN ỨNG NƯỚC ÁP LỰC VỚI NHIÊN LIỆU HỖN HỢP CÓ KHẢ NĂNG CHỊU ĐƯỢC TAI NẠN BẰNG CHƯƠNG TRÌNH SRAC

VAN KHANH HOANG

*Viện Khoa học và Kỹ thuật Hạt nhân, 179 Hoàng Quốc Việt, Nghĩa Đô, Cầu Giấy, Hà Nội*

*E-mail: hvkhanh21@gmail.com*

**Tóm tắt:** Trong tương lai của năng lượng hạt nhân, việc thiết kế và phát triển nhiên liệu có tính kinh tế, có khả năng chịu được tai nạn trong sự cố (Accident Tolerant Fuel, ATF) để sử dụng trong lò phản ứng nước áp lực (PWR) là cần thiết và rất được mong đợi. Nhiên liệu hỗn hợp được đánh giá là có ưu điểm hơn nhiên liệu $UO_2$ truyền thống bởi độ dẫn nhiệt và mật độ Uran cao hơn. Do nhiên liệu hỗn hợp có mật độ Uran cao hơn, việc sử dụng chúng sẽ tăng đáng kể độ sâu cháy tại cuối chu trình nhiên liệu, do đó phát huy tính kinh tế của chu trình nhiên liệu khi so sánh với nhiên liệu $UO_2$. Độ dẫn nhiệt cao hơn của nhiên liệu hỗn hợp tăng các giới hạn an toàn của nhiên liệu. Để đánh giá các ưu điểm của ATF, nghiên cứu này cũng xem xét khả năng thay thế vỏ nhiên liệu Zircaloy-4 truyền thống với SiC để giảm thiểu khí Hydro sinh ra từ tương tác của nước với vỏ nhiên liệu tại nhiệt độ cao. Trong nghiên cứu này, tính toán vật lý cho nhiên liệu hỗn hợp của một lò phản ứng nước áp lực được thực hiện và so sánh với nhiên liệu $UO_2$ truyền thống. Các tính toán được thực hiện với chương trình SRAC và thư viện JENDL-4.0. Các thông số được khảo sát để thiết kế nhiên liệu hỗn hợp là độ giàu U-235 và kích thước ô mạng. Để giảm sự quá ngưỡng của độ phản ứng, Erbi được lựa chọn là chất độc cháy được bởi khả năng cháy tốt của nó. Các hệ số phản hồi theo nhiệt độ bao gồm cho nhiên liệu, chất làm chậm và các hệ số độ phản ứng với độ rỗng nhỏ và lớn cũng được khảo sát. Nghiên cứu này cho thấy khả năng đạt được tới hạn với độ sâu cháy lên tới 100 GWd/tHM mà không ảnh hưởng tới các thông số an toàn. Các phân tích tiếp theo về tính năng của nhiên liệu hỗn hợp trong mô hình vùng hoạt PWR hiện đang được thực hiện.

**Từ khóa:** nhiên liệu $UO_2$, nhiên liệu hỗn hợp, bó nhiên liệu PWR, phân tích vật lý.