Possibility Studies of a Boiling Water Cooled Traveling Wave Reactor

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Abstract. This paper investigates the possibility of a boiling water cooled traveling wave reactor, which can improve the natural uranium utilization of a Light Water Reactor (LWR) by umpteen times. The high density variation of boiling water through the core is favourable to fission at the lower part of the core and to the fuel breeding at the upper part of the core. This is the case in a low pressure Boiling Water Reactor (BWR). A serial axial fuel shuffling, which makes fuel moving, is considered. The natural uranium oxide fuel is fed in from the top of the core and discharged from the bottom of the core, as the water at the saturation point is fed in from the bottom of the core as in a boiling water reactor. The asymptotic state of the breeding/burning wave is searched theoretically and numerically, where the power (neutron flux) and the water density are fitted to each other to form a fission-breeding mixed reactor configuration. The major parameters of power, coolant mass flow rate, and the fuel shuffling speed are coupled to each other and determined by numerical solutions. A theoretical model for the water boiling is established based on a slip ratio model of two phase flow. The critical heat flux limit has been taken into account. The neutronics and burn-up calculations are performed with the ERANOS2.2 code, where models of axial fuel shuffling and coolant density change have been implemented. The 1-D preliminary numerical results are encouraging and show that the breeding is sufficient to make the core be critical and the maximum burn-up can reach up to 40%, meaning the natural uranium utilization of BWR can be improved in an order of magnitude of 50 times.

Key Words: Traveling wave reactor, Boiling water reactor, High fuel utilization, Neutronic and thermal hydraulic coupling.

1. Introduction

The mechanism of a self-propagating nuclear breeding/burning wave in a fertile medium of ^{238}U or ^{232}Th becomes more and more attractive [1-8]. Such a self-propagating nuclear wave leads to an almost constant reactivity and a high burn-up and, consequently, a very high utilization of fuel. It is known as “CANDLE” concept [4]. This mechanism can be used for constructing a so-called traveling wave reactor (TWR) for long operation duration with high fuel utilization, thus reducing or avoiding reprocessing needs. Such a TWR concept recently received a wide attention due to the engagement of Bill Gates and his funded company TerraPower, where a radial fuel shuffling strategy was proposed for developing a newer variety of TWR based on sodium-cooled fast reactor (SFR) technologies [9,10].

However the sodium cooled TWR faces two major challenges, namely the possible positive coolant void worth and the high clad irradiation damage. People always hope to solve this problem with light water reactor technologies, which are much easier than the sodium technologies. There were several attempts to apply the TWR concept in supercritical water reactor (SCWR) [11-13], which show the fuel utilization improvement potential, but they are not fully succeeded [14,15]. The reason is that there is still too much water in the upper core and the breeding is not sufficient.

Comparing to SCWR, the boiling water reactor (BWR) is even more favourable for the TWR concept due to possibly high variation of water density of the water/vapour mixture in the core. If the vapour quality is 10% and the core pressure is low, the density at the core outlet could be 1/100 of that at the core inlet. This implies that the neutron spectrum at the upper part of core could be sufficiently hard. Consequently the lower part favours fission, while the upper part does breeding. Thus the breeding/burning wave that moves into the breeding part, namely upwards, can be easily generated in a natural way. In this aspect the BWR is a very good application possibility of the TWR concept. The major concern here is if the breeding is sufficient to make the traveling wave mode be critical.

The light water reactors (LWR) have been have been commercially operated for more than 50 years. They are established existing technologies. However LWRs have a rather low natural uranium utilization rate of barely 0.6 % [16]. This means that from the all uranium used for the nuclear fuel production, only less than 1 % uranium undergoes fission reaction and the rest uranium becomes deleted uranium and spent fuel. From scientific point of view, this low uranium utilization is actually a kind of natural resource waste, which is not compatible with the aim of nuclear energy sustainability. The fuel reprocessing and cycle are only a partial solution of the problem, because of high costs and technological difficulties of highly irradiational spent fuel handling. There were scientific and engineering efforts to improve it directly, e.g. to increase the conversion ratio by using thorium fuel and heterogeneous breeding zones [16, 17], which have only local and limited improvements. But to our knowledge, an idea with LWR technologies for an overall improvement with a significant order of magnitude, e.g. up to 50 times, has not been presented.

This paper is a theoretical and numerical attempt to look for a possibility of the TWR concept in a boiling water type reactor. The large variation of the water density leads to a significant change of the neutron spectrum and the corresponding microscopic cross sections. This needs to consider thermal hydraulic/neutronic coupling, meaning a thermal hydraulic solution of water density variation for any given power shape. As a first attempt, a 1-D numerical calculation with the ERANOS code [18] has been carried out. The solution obtained in this paper is for an asymptotic state of TWR. Obviously the asymptotic material distribution can be very useful for the ignition transient phase as well. This initial fuel distribution, e.g., of the plutonium enrichment, can be so arranged, which is similar to the asymptotic state, so that the asymptotic state can be reached finally. But in this paper we will not investigate ignition transients.

An analytic thermal hydraulic solution of water density as a function of power distribution and a certain coolant flow rate is obtained based on a slip-ratio water/vapour mixture model. It is implemented in the ERANOS calculation. The ERANOS code [18] is adopted to perform the neutronics and burn-up calculation, where the axial fuel shuffling scheme and coolant density variation are taken into account. The main objective of the calculation is to assure if the neutron flux and the water density can be suitably coupled during the fuel drifting, so that the breeding is sufficient for the criticality.

2. Numerical Calculation Scheme and Thermal Hydraulic Model

2.1.Axial Fuel Shuffling Strategy and Calculation Scheme

The basic idea is that the pin bundle in a fuel assembly (FA) or the fuel assembly itself is divided into several movable blocks, which is similar to the case studied in [15] and was illustrated in Fig. 1 of [15]. The first fresh fuel pin bundle block is loaded into the core from the top side, simultaneously accompanied by unloading the last spent fuel pin bundle block at

the bottom, while the other fuel pin bundle blocks in between move downwards like a frog leap. The fuel shuffling or fuel jump takes place periodically, e.g. one jump per 1000 days. Beginning from an initial condition with fresh fuels loaded in the whole core, an asymptotic state will be approached after certain steps of fuel jump, in which k_{eff}, the power shape, the nuclide densities and the coolant density distribution tend to stable values. As a first attempt, only 1-D problem is solved here.

The calculation scheme is built up completely corresponding to the axial fuel shuffling strategy, in which the core is firstly divided into several cells along the axial direction and then the initial enrichment, total thermal power and fuel shuffling speed (corresponding to burn-up time steps) are adjustable for achieving the asymptotic state at the desired k_{eff} level. In this paper, the ECCO code with JEFF3.1 (The JEFF-3.1 Nuclear Data Library, 2006) data library is firstly used to generate the nuclides microscopic cross sections, and the ERANOS code is adopted to perform the neutronics and the burn-up calculations with 40 energy groups and 80 fission products by applying the nodal diffusion model of VARIANT.

The coupling between neutronics and thermal hydraulics is an important issue in this simulation. In each time (iteration) step, the neutronic part provides the power distribution to the thermal hydraulic part and then the thermal hydraulic part calculates the coolant density for the next step of neutronic calculation. So the iteration goes on. The thermal hydraulic model will be described in detail in the next subsection.

2.2.Single Channel Thermal Hydraulic Model of Water Boiling

For neutronic calculations we need the water density distribution. The neutronic and thermal hydraulic coupling lies in that the neutronic part provides the power to the thermal hydraulic one, while the thermal hydraulic parts provides the effective water density to the neutronic one. The temperature feedback effects are of second order in this problem and will not be taken into account here.

For the sake of simplicity we assume that the water at the core inlet is 100% liquid at the saturation temperature under a certain pressure. The coolant in the core is a water-vapour mixture with a certain velocity slip ratio, which is determined by a correlation. Since we consider a very long time scale of several years, the coolant flow can be regarded and treated as a steady state.

Now we consider a single subassembly channel flow with the mass flow rate $\dot{m}$ and the flow area *A*. According to the mass conservation, the sum of the mass flow rates of water and vapour should be constant over the whole channel as

$$\dot{m}_w + \dot{m}_v = \dot{m}_{w,0} = \dot{m} = const \tag{1}$$

where $\dot{m}_w$ and $\dot{m}_v$ are the water and vapour mass flow rates, and $\dot{m}_{w,0}$ the core inlet water mass flow rate, being equal to the total water mass flow rate $\dot{m}$.

The next step is to consider the energy conservation. Since the water-vapour mixture is already and always at the saturation condition, e.g. T = 100 °C at 1 atm, the nuclear thermal energy is converted totally to the water boiling. Thus, for the single channel with linear power χ, we have

$$\Delta \dot{m}_v h = \chi \Delta z \tag{2}$$

where *h* is the water boiling latent heat. This means the decrease of the water mass flow rate that is the boiling water mass rate is equal to the thermal power. In this case the vapour mass quality can be simply determined as

$$x = \frac{\dot{m}_v}{\dot{m}_{w,0}} = \int_0^z \frac{\chi}{\dot{m}_{w,0} h} dz \tag{3}$$

Herewith we introduce an important parameter defined as

$$k = \frac{\chi}{\dot{m}_{w,0}\, h} \tag{4}$$

which characterizes the water boiling process, has a dimension of reciprocal length and, therefore, is called as *boiling length number*. Obviously it is a variable over the core length, i.e. a function of *z*.

In order to get the effective water density we have to get the relative volume fraction of vapour, the so-called void fraction denoted as

$$\varepsilon = \frac{\alpha_v}{\alpha_w + \alpha_v} \tag{5}$$

which α_v and α_w are vapour and water volume fractions. The void fraction ε is one of the most important parameters used to characterize two phase flows. For more information [19] is referred. The simplest analytic solution is the homogeneous water-vapour mixture model, where it is assumed that water and vapour have a same velocity. In terms of the total mass flow rate $\dot{m}$, the vapour quality *x* and the void fraction ε, the vapour and water velocities can be expressed as

$$u_v = \frac{\dot{m}}{\rho_v}\left(\frac{x}{\varepsilon}\right), \quad u_w = \frac{\dot{m}}{\rho_w}\left(\frac{1-x}{1-\varepsilon}\right) \tag{6}$$

where subscripts *w* and *v* stand for water and vapour, respectively, and ρ the theoretical density. Thus, equating above expressions for equal velocities in two phases, we obtain the homogeneous void fraction, denoted as ε_h,

$$\varepsilon_h = \frac{1}{1+\left(\frac{1-x}{x}\right)\frac{\rho_v}{\rho_w}} \tag{7}$$

In general, the homogeneous void fraction model is reasonable accurate for only a limited range of circumstances. In the special case, where the density ratio is very small, ε_h is significantly overestimated, which leads to a too small effective density in the upper part of the core.

A natural correction of this model is to assume that the two phases have two different mean velocities, u_v and u_w, and to introduce their ratio into the homogeneous model. The velocity ratio is usually referred as the so-called slip ratio *S*

$$S = \frac{u_v}{u_w} \tag{8}$$

Substituting the velocity expressions (6) into (8) we obtain immediately

$$\varepsilon = \frac{1}{1+\left(\frac{1-x}{x}\right)\frac{\rho_v}{\rho_w}S} \tag{9}$$

Of course, if $S = 1$, (9) will return to the homogeneous result (7).

There are a lot of analytic solutions and empirical correlations for the slip ratio S. For our purpose we adopt the Zivi's first model [19, 20].

$$S = \left(\frac{\rho_w}{\rho_v}\right)^{\frac{1}{3}} \tag{10}$$

Finally the void fraction is expressed as

$$\varepsilon = \frac{1}{1+\left(\frac{1-x}{x}\right)\left(\frac{\rho_v}{\rho_w}\right)^{\frac{2}{3}}} \tag{11}$$

where the vapour quality x has been obtained from the energy balance (3). It is only remarked without detailed discussions that this expression leads to reasonable void fraction value in the mass flux range around 400 kg/(m^2 s), while the homogeneous model provides an upper limit of the void fraction, see Fig. 17.8 in [19]. The vapour to water density ratio can be regarded as a constant in the core, as the pressure change is negligible. It can be seen e.g. that $\rho_v/\rho_w \approx 1/1600$ at the atmosphere pressure and $\rho_v/\rho_w \approx 1/172$ and 1/59 at the pressure of 10 and 28 bar, respectively. The relative effective water density in terms ε of is finally expressed as

$$\frac{\rho}{\rho_0} = (1-\varepsilon) + \frac{\rho_v}{\rho_w}\varepsilon \tag{12}$$

This water density variation is exactly what we need for the neutronic calculation. Sometime we use the relative water volume fraction β, which is defined as

$$\beta = \frac{\alpha_w}{\alpha_w + \alpha_v} = 1-\varepsilon \tag{13}$$

2.3.Discussions on Major Parameters and Boiling Crisis

The linear power is defined as power per length as $\chi(z) = dP/dz$ for a certain pin or a certain sub-assembly channel or even whole reactor core. The axial power distribution is usually normalized by its average as

$$\overline{\chi}(z) = \frac{\chi(z)}{\chi_0} \tag{14}$$

where is χ_0 the average value of the linear power. The maximum value of $\overline{\chi}(z)$ is actually the power peaking factor in the 1-D case. In the special case here, the boiling length number k can be normalized as

$$k(z) = k_0\frac{\chi(z)}{\chi_0} \quad \text{with} \quad k_0 = \frac{\chi_0}{\dot{m}_{w,0}\,h} \tag{15}$$

Since the length unit used in the neutronic code is cm, χ is in W/cm and k in cm^{-1}.

The integrating $k(z)$ over whole the core is actually the ratio of the total power to the total water boiling latent power, i.e. $P/\dot{m}_{w,0}\,h$, which is exactly the vapour mass quality at the core outlet. Thus, the vapour mass quality at the core outlet can be expressed in terms of k_0 as

$$x_{outlet} = \frac{P}{\dot{m}_{w,0}\,h} = k_0 L_{core} \tag{16}$$

where L_{core} is the core length. The vapour mass quality in a BWR is usually in the range between 10% and 20%. We may choose these two typical values for our calculation examples. This means the reactor power is 10% or 20% of the total water boiling latent power.

2.4.Critical Heat Flux

There is another important issue to be discussed here. That is the boiling crises. Usually the boiling crises or the critical heat flux (CHF) can be characterised by a critical value of the ratio of the heat flux q_x to the mass flux G, i.e.

$$\frac{q_x}{G} < \text{critical value in kJ/kg} \tag{13}$$

As we consider the fuel subassembly flow, the average heat flux and the mass flux can be expressed as

$$q_x = \frac{\chi_{pin}}{\pi d_{pin}} = \frac{\chi_{FA}}{\pi d_{pin} N_{pin}}, \qquad G = \frac{\dot{m}}{A_{coolant}} \tag{14}$$

where d_{pin} is the pin diameter, N_{pin} the number of pins in a fuel assembly and A_{coolant} the coolant flow cross sectional area. Therefore we have the relationship between k_0 and q_x/G as

$$\frac{q_x}{G} = \frac{\chi_{FA} h A_{coolant}}{\dot{m} h \pi d_{pin} N_{pin}} = k_0 \frac{h A_{coolant}}{\pi d_{pin} N_{pin}} = \frac{k_0 D_x}{4} h \quad \text{with} \quad D_x = \frac{4 A_{coolant}}{\pi d_{pin} N_{pin}} \tag{15}$$

where D_{x} can be called as the equivalent heat flux tube diameter. This is of course the average heat flux to the mass flux ratio. The maximum ratio has to be multiplied the total power peaking factor f_{peak}.

$$\frac{q_{x,\max}}{G} = \frac{f_{peak}}{4} k_0 D_x h \tag{16}$$

As reported in [21], this critical value is more than 2 kJ/kg for various cases. To be conservative it is sufficient to put the limit in this study as

$$\frac{q_{x,\max}}{G} < 2\ \text{kJ/kg} \tag{17}$$

2.5.Configuration of Fuel Assembly and Pins and Coolant Density Variation

As a continuation to the numerical SCWFR study [15], the SCWFR seed SA/pin design [12] are chosen and modified. Only a one-dimensional case is considered in current numerical

simulations and the core is assumed to be filled with fuel subassemblies. The number of fuel pins is changed to 169, as the usual 6 control pins are replaced by fuel pins. The fuel pin diameter is modified to be 1.083 cm and its pitch 1.158 cm. Therefore the fuel, steel, and coolant volume fractions that are used in the neutronic calculations are 54.9%, 20.7% and 24.4%, respectively. Their theoretical densities of 10.45, 6.23 and 0.958 g/cm^2 are taken in the calculation. The necessary geometric data are given in TABLE I. The core length is 165 cm and the fuel power density is 200 W/cm^3. This is a very tight arrangement of fuel pin. What we did is just to choose the boiling length number k_0, so that the criticality condition and the CHF condition are satisfied. Further check of the thermal hydraulic conditions and improvement of the design should be done.

The core pressure is assumed to be 1 bar, because we need a very strong water density reduction at the core outlet for sufficient breeding there. TABLE II shows density differences in typical cases, e.g. at the atmosphere pressure (the first line) and at a typical BWR pressure (the last line). It can be easily calculated that for 10% vapour quality under the atmosphere pressure, the relative water-vapour mixture densities at the outlet with respect to the inlet one are 0.0062 and 0.063 according to the homogeneous and Zivi's slip ratio models, respectively. The pressure can control the density reduction at the core outlet, which is important for breeding. This is also why we choose the low pressure in this study.

TABLE I: FUEL ASSEMBLY AND PIN GEOMETRIC DATA.

Variable	Value, Unit
Pitch of FA	160 mm
Gap between FAs	2 mm
Thickness of FA wrapper	2 mm
Number of fuel pins per FA	169
Fuel pin diameter	10.83 mm
Fuel pin pitch	11.58 mm
Clad thickness	0.56 mm
Active fuel height	1650 mm

TABLE II: WATER AND VAPOUR DENSITIES AND THEIR RATIO AT SEVERAL TYPICAL SATURATION PRESSURES.

Pressure In bar	Temperature in °C	Boiling latent heat in kJ/kg	Water density in kg/m^3	Vapour density In kg/m^3	Vapour to Water density ratio
1.013	100	2257	958.4	0.5975	0.0006234
10.02	180	2015	887.1	5.154	0.005810
19.06	210	1900	852.8	9.580	0.01123
74.37	290	1477	732.2	39.12	0.05343

3. Numerical Results

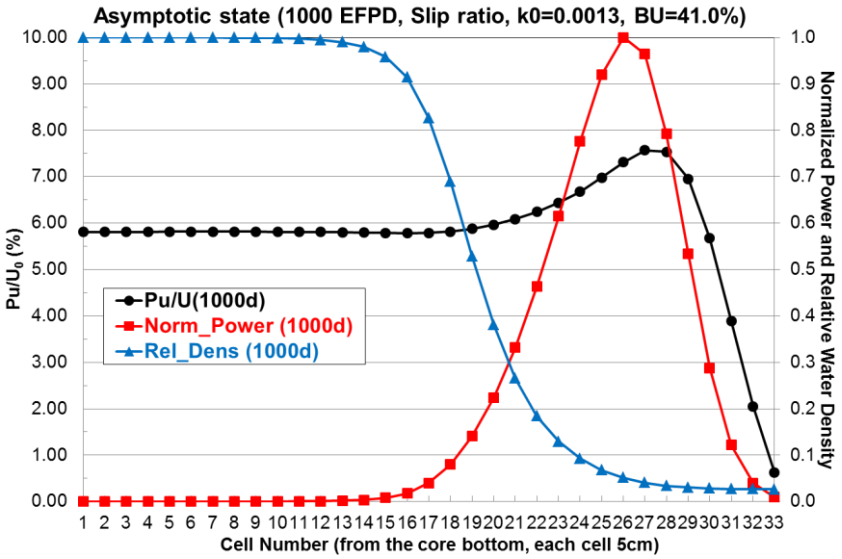
The 1-D numerical model is a single pin surrounded by coolant with a perfect reflexion boundary condition. The fuel power density, i.e. per fuel volume is assumed to be 200 W/cm^3, which is roughly 100 W/cm^3 per total volume. The core length is assumed to 165 cm. This means the total pin power is 26.965 kW and the average linear pin power is 163.426 W/cm. The core length is divided into 33 nodes, each node 5 cm. Physically, the pin bundle in a

subassembly (SA) is divided into 33 movable blocks, which is similar to the case studied in [15] and was illustrated in Fig. 1 of [15]. The first fresh fuel pin bundle block is loaded into the core from the top, while the last spent fuel bundle block is discharged from the bottom. By each fuel shuffling every fuel block moves downwards at a jump step. Therefore the fuel shuffling speed can be expressed as 5 cm per 1000 days for example. The fresh fuel is the natural uranium oxide. A typical BWR steel is used here.

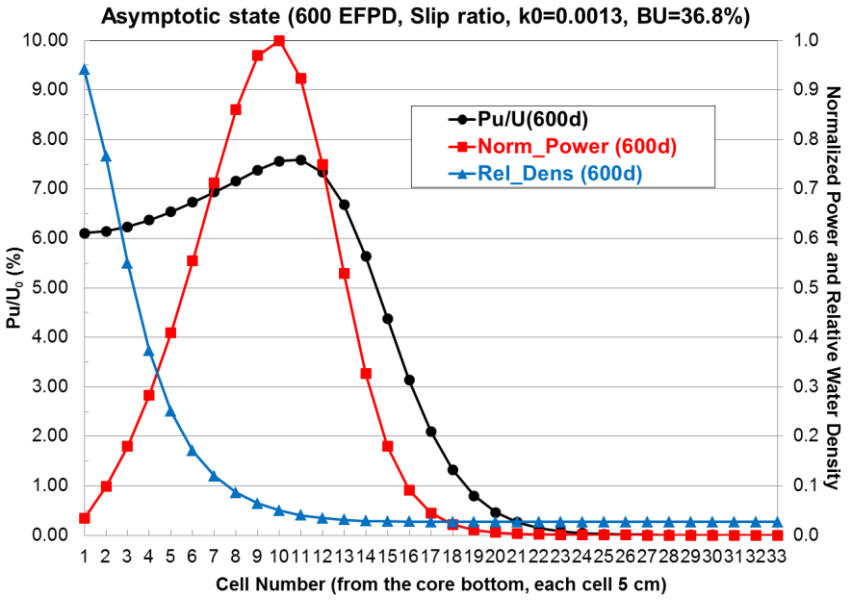
Besides the power, the mass flow rate and the fuel shuffling speed are other two important parameters, which are to be determined. TABLE III summarizes major results of parametric studies. The boiling length number k_0 represents the ratio of power to mass flow rate. The higher the value of k_0, the higher the vapour quality and the lower the smear water density at the core outlet. We tried various values of k_0 and several fuel shuffling speeds and found that the cases of k_0 = 0.0013 cm^{-1} with fuel shuffling speeds of 5 cm per 600 days and 1000 days are interesting. Both fuel shuffling speeds can make the reactor to reach an asymptotic critical state with a power peaking factor about 4 and burn-up about 40%. Fig. 1 (a) and (b) show distributions of power, water density and Pu-enrichment at the asymptotic state for the two fuel shuffling speeds. For the slower fuel shuffling of 5cm/1000d in case (a) of Fig. 1, the power profile still stays at the lower part of the reactor, while for the faster fuel shuffling of 5cm/600d in case (b) of Fig. 1, the power profile moves already to the upper part of the reactor. For both cases the effective power profile length is about 1 m, much smaller than the reactor length. The power profile length is indeed the wave length of this kind of traveling wave reactor, which is an inherent property of the Pu-U conversion process [8]. The short effective reactor length is favourable to a compact reactor design. The high bur-up means not only the high fuel utilization, but also the low nuclear waste produced per unit energy.

TABLE III: OVERVIEW OF MAJOR CALCULATION RESULTS FOR Lcore = 165 cm (SLIP RATIO MODEL).

k_0 [cm^{-1}]	Vapour quality **x**	Relative water density ρ/ρ_0	q_x/G average	k_{eff}	1000d Power peaking factor	Burn-up	k_{eff}	600d Power peaking factor	Burn-up
0.0013	0.2145	0.02668	0.2760	1.0055	4.48	41.0%	1.0176	4.13	36.8%



(a)



(b)

*FIG. 1. Calculated results of power, water density and Pu enrichment in the case of k_0 = 0.0013 and **(a)** 5 cm per 1000 d and **(b)** 5 cm per 600 d.*

4. Conclusion

This paper studies the possibility of boiling water cooled traveling wave reactor. The analytic thermal hydraulic solution of water density is obtained and implemented in the neutronic code. In the low pressure case, it has been shown that the breeding is sufficient and asymptotic traveling wave state exists. It means that the high burn-up, high fuel utilization and low nuclear waste can be achieved with existing technology of boiling water reactor. This is a seeding idea that makes the nuclear energy much more sustainable.

Reference

[1] TELLER, E., ISHIKAWA, M., WOOD, L., "Completely automated nuclear reactors for long-term operation", Proceedings ICENES'96, Obninsk, Russia (1996) 151-158.

[2] SEIFRITZ, W., "Solitary burn-up waves in a multiplying medium", Kerntechnik **65** (2000) 5.

[3] VAN DAM, H., "Self-stabilizing criticality waves", Annals of Nuclear Energy **27** (2000) 1505.

[4] SEKIMOTO, H., RYU, H., YOSHIMURA, Y., "CANDLE: The new burnup strategy", Nuclear Science and Technology **139** (2001) 306.

[5] CHEN, X.-N., KIEFHABER, E., MASCHEK, W., "Neutronic model and its solitary wave solutions for a CANDLE reactor", Proceedings of ICENES'05, Brussels, Belgium (2005).

[6] CHEN, X.-N., MASCHEK, W., "Transverse buckling effects on solitary burn-up waves", Annals of Nuclear Energy **32** (2005) 1377.

[7] CHEN, X.-N., KIEFHABER, E., MASCHEK, W., "Fundamental burn-up mode in a pebble-bed type reactor", Progress in Nuclear Energy **50** (2008) 219.

[8] CHEN, X.-N., KIEFHABER, E., ZHANG, D., MASCHEK, W., "Fundamental solution of nuclear solitary wave", Energy Conversion and Management **59** (2012) 40.

[9] WEAVER, K. D., et al., "Extending the nuclear fuel cycle with traveling-wave reactors", Proceedings of Global 2009, Paris (2009).

[10] HEJZLAR P., et al., "TerraPower, LLC traveling wave reactor development program overview", Nucl Eng Technol **45** (2013) 731.

[11] SCHULENBERG, T., BITTERMANN, D., STARFLINGER, J., "Design status of the high performance light water reactor", Proceedings of 16th Pacific Basin Nuclear Conference, Aomori, Japan (2008).

[12] ISHIWATARI, Y., OKA, Y., KOSHIZUKA, S., "Breeding ratio analysis of a fast reactor cooled by supercritical light water", J. Nuclear Science and Technology **38** (2001) 703.

[13] MORI, M., Core Design Analysis of the Supercritical Water Fast Reactor. Dissertation, Institute for Nuclear Energy, University of Stuttgart, Germany (2005).

[14] CHEN, X.-N., ZHANG, D., MASCHEK, W., SCHULENBERG, T., "Solitary breeding/burning waves in a supercritical water cooled fast reactor", Energy Conversion and Management **51** (2010) 1792.

[15] ZHANG, D., CHEN, X.-N., GABRIELLI, F., RINEISKI, A., MASCHEK, W., SCHULENBERG, T., "Numerical studies of nuclear traveling waves in a supercritical water cooled fast reactor", Progress in Nuclear Energy **53** (2011) 806.

[16] VALLET, V., et al., "Introduction of thorium-based fuels in high conversion PWR", Nuclear Technology **182** (2013) 187.

[17] JANIN, D., DOUCE, S., SEIDL, M., MACIAN, R., "High conversion small modular reactor (HCSMR): core simulations with CRONOS2, APOLLO3® AND TRIPOLI-4®", International Youth Nuclear Congress, Hangzhou, China (2016).

[18] RIMPAULT, G., et al., "The ERANOS Code and Data System for Fast Reactor Neutronic Analyses", Proc. of the International Conference on the Physics of Reactors (PHYSOR 2002), Seoul, Korea (2002).

[19] THOME, R. J., Engineering Data Book III, Chapter 17 Void Fractions in Two-Phase Flows, Wolverine Tube Inc. (2004-2010).

[20] ZIVI, S. M., "Estimation of steady state steam void fraction by means of the principle of minimum entropy production", Trans. ASME, J. Heat Transfer **86** (1964) 247.

[21] HAAS, C., Critical Heat Flux for Flow Boiling of Water at Low Pressure on Smooth and Micro-Structured Zircaloy Tube Surfaces. Karlsruhe Institute of Technology, KIT SCIENTIFIC REPORTS 7627, KIT Scientific Publishing (2012).