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Coupled analysis for new fuel design using UN and UC for SCWR

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A B S T R A C T

All-ceramic bi-material coated fuel pellet design is proposed and analyzed for thermal Super Critical Water Reactor (SCWR). Uranium mono nitride (UN) with Zirconium Carbide (ZrC) coating and Uranium mono carbide (UC) with Silicon Carbide (SiC) coating are analyzed. Carbide and nitride ceramic fuels offer the advantage of high thermal conductivity as compared to UO_2. Use of coating can solve the problems of harder spectrum and reactive nature for UN and UC which arise when these fuels are used in light water thermal reactors. Larger heavy metal density of UN and UC can lead to equivalent or even larger values of heavy metal loading using coated pellet concept. Coating can give the extra advantage of working as yet another barrier against release of radioactive fission gases in accidental scenarios. Due to large coolant density variation along active height of fuel, coupled neutronics/thermal hydraulic analyses are performed using MCNP4c/SACoS coupled system. Design proposed by Shanghai Jiao Tong University (SJTU) for thermal SCWR is chosen to perform the calculations. Coupled analyses show that coated fuel pellets lead to more uniform radial power density which leads to smaller value of hot channel factor. A significant decrease in fuel centerline temperature is seen due to high thermal conductivity of carbide and nitride ceramic fuels. A staggering difference of more than 500 °C is seen between the maximum average centerline temperatures for UO_2 and coated fuel pellets. Slightly smaller values of average clad surface temperature are obtained using coated fuel pellets as compared to UO_2. A maximum coating temperature of less than 800 °C is observed. Effect of varying coating thickness is studied by performing analyses for different coating thickness values. Keeping the same fuel pin radius and pitch, increasing the thickness of coating materials can lead to increased value of thermal neutrons but will have a negative effect on the burnup due to smaller heavy metal loading. Reactivity coefficients, i.e. Doppler, coolant and moderator, show negative values. Considering the favorable neutronic and thermal hydraulic properties along with a potential increase in cycle length, all-ceramic bi-material coated fuel pellets can be a very good fuel option for SCWR.

1. Introduction

Supercritical Water Reactor (SCWR) is one of the six ideas chosen for further study under the banner of Generation IV (GEN-IV) nuclear reactors. It is the only concept in the list of GEN-IV reactors which is cooled and moderated by light water. SCWR is believed to be the next logical evolutionary step for Light Water Reactors (LWR's). As no change of phase for coolant/moderator takes place, significant simplifications of plant are possible. It can combine the tried and tested technologies of LWR and supercritical fossil fuel fired power plants. SCWR design studies span a wide range of reactor ideas i.e. thermal (Liu and Cheng, 2009b; Oka et al., 1992; Squarer et al., 2003), mix (Cheng et al., 2007) and fast (Oka and Koshizuka, 1998) neutron spectrum reactors along with pressure tube (Shan et al., 2009) and pressure vessel (Cheng et al., 2007; Liu and Cheng, 2009b; Oka et al., 1992; Squarer et al., 2003) design to house the desired neutron spectrum assemblies. Due to large variation of coolant and moderator density along the active length of fuel, coupled analyses for design and safety are deemed necessary for SCWR (Chaudri et al., 2012; Maráczy et al., 2010; Waata, 2006; Yamaji et al., 2005).

Shanghai Jiao Tong University (SJTU) proposed design for thermal SCWR (Liu and Cheng, 2009a, b) is quite a promising design concept. This design proposes two rows of fuel rods as compared to one fuel rod row between water boxes being used in other designs (Yamaji et al., 2005). This approach leads to a more uniform radial

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power distribution due to enhanced thermalization which, in turn, leads to decreased values of clad surface temperature, centerline fuel temperature and nuclear hot channel factor.

Current thermal designs of SCWR (Liu and Cheng, 2009b; Oka et al., 1992; Squarer et al., 2003) have been evaluated using UO_2 as possible fuel. UO_2 offer the vast operation experience from LWR's. A well developed fuel cycle is also a big plus for this kind of fuel. But the problem with UO_2, being used in high temperature reactor applications, is its small thermal conductivity which decreases further at higher temperatures. GEN-IV reactors are being developed with the promise of high thermal efficiency so that cheap electricity can be produced. High thermal efficiency means higher outlet temperature which implies that UO_2 might not be able to do the job with required safety margins at these higher temperatures (Grande, 2010). Recently, neutronics/thermal hydraulic coupled analyses have been performed to evaluate the option of hydride fuel (Ammirabile, 2010) to be used in European design of SCWR, known as High Performance Light Water Reactor (HPLWR) (Squarer et al., 2003). Thermal aspects of different fuels (UC, UC_2, UN, ThO_2, MOX etc) to act as alternatives for UO_2 have been studied in various recent works (Grande, 2010; Pascoe et al., 2010). Other studies have also been conducted to check the neutronic feasibility of alternate fuels like ThO_2 (Csom et al., 2012) or MOX fuel (Reiss et al., 2010) in SCWR. These works indicate the trend of nuclear community towards searching alternate fuel options for SCWR.

Carbide and nitride ceramic fuels have been around in nuclear industry for quite a while (Stoddard, 1974). Ceramic fuels, with the exception of UO_2, possess the property of high thermal conductivity which makes them suitable for high temperature reactor applications. Fig. 1 shows the comparison of the thermal conductivity for UO_2, UN and UC fuels. High density of carbide and nitride ceramic fuels can help the design of compact cores. These are the reasons that UN and UC are considered as potential fuels for nuclear propelled space program (El-Genk and Schriener, 2011). There has been a renewed interest in UN and UC fuels for power reactors as they are being considered as viable fuel candidate in GEN-IV reactors i.e. Gas-cooled Fast Reactor (GFR) (Nosek et al., 2007) and as fuel kernel for TRISO particles in High Temperature Gas Reactor (HTGR) (Kuijper et al., 2006) concepts.

1.1. Uranium mono nitride (UN)

Uranium mono nitride (UN) fuel has been quite a famous choice for space reactors (Poston, 2002). High density, high thermal conductivity and high melting point make it very good option for compact and high temperature reactors. High density, which leads to a higher value of heavy metal as compared to coolant/moderator, can lead to harder spectrum. That is why it has been the fuel of choice in fast neutron spectrum reactors (Smith et al., 2007). High density also makes it a feasible fuel for compact reactors, for example Small Secure Transportable Autonomous Reactor (SSTAR) (Smith et al., 2007) and Hyperion Power Modules (Inc., 2011). As evident from Fig. 1, thermal conductivity value of uranium mono nitride at room temperature is almost 2 times that of UO_2 fuel. This value goes up to 8–10 times at reactor operating at higher temperatures. Melting point for uranium mono nitride is comparable to that of UO_2 i.e. 2850 °C.

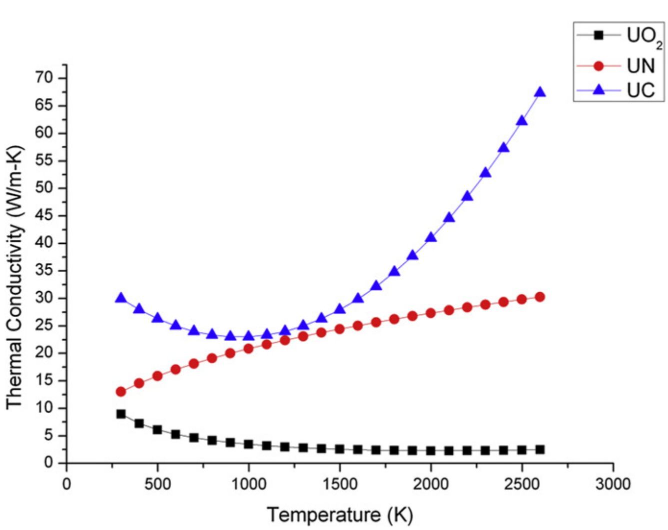

Fig. 1. Comparison of thermal conductivity for UO_2, UN and UC.

Main problems of using uranium mono nitride fuel in thermal SCWR are:

- SCWR core already suffers from under moderation due to large density variation of coolant over active length of fuel. To overcome this under moderation effect in SCWR, water boxes are incorporated in design. Using UN fuel, keeping the current fuel design intact, will lead to harder spectrum.
- At low values of nitrogen vapor pressure, UN is known to dissociate over 1600 °C i.e. well before its melting point. Equation (1) shows the dissociation reaction of UN

$$UN(s) \rightarrow U(liq) + 0.5N_2(gas) \quad (1)$$

- It is chemically reactive to nickel (J.P, 1972), which is present as a major constituent of Alloy-718 (structural and clad material for SJTU thermal SCWR), and water (Dell et al., 1967; Sunder and Miller, 1998). Equation (2) gives the UN reaction with water.

$$UN + 2H_2O \rightarrow UO_2 + NH_3 + 0.5H_2 \quad (2)$$

A solution proposed for thermodynamic stability problem of nitride fuel is adding small amounts of yttrium and titanium which will reduce dissociation of heavy metal and nitrogen and allow normal operation (without dissociation) for temperatures in excess of 1700 °C (Carl A. Alexander, 1986). Most recent works under the banner of Collaboration on Nitride Fuel Irradiation and Modeling (CONFIRM) project has suggested an addition of Zirconium to make (U, Zr) N system (Wallenius, 2001). This (U, Zr) system significantly increases the thermodynamic stability of the fuel as compared to simple UN fuel and allow normal operation without dissociation for temperature as high as 2600 °C. The main drawback of this approach is that substantially large fraction of Zirconium (Zr) is needed which will take away the advantage of using UN fuel for heavy actinide density (Choi et al., 2006).

Another requirement for using uranium mono nitride fuel is that it should use 99.99% N-15 instead of N-14. This is because N-14 can parasitically absorb neutron to transmute into C-14. This has a negative effect on neutron economy. Also, C-14 is a long lived radioactive isotope which will pose a problem when planning to deal with the radioactive spent fuel. Although, enrichment in N-15 will result in high fuel fabrication cost but recent studies done under CONFIRM project have shown that for sufficiently large pitch values, N-15 enriched fuel pins have shown much smaller or even negative nitrogen void worth values as compared to N-14 (Wallenius et al., 2000). Hence enrichment in N-15 is deemed to be a prerequisite and necessity for safety and neutron economy by this study. Research about economic potential of alternate fuels in LWR thermal reactors have concluded that UN can be a good candidate for once through fuel cycle (Oggianu et al., 2003).