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NEUTRONIC ANALYSIS FOR ACCIDENT TOLERANT CLADDING CANDIDATES IN PWR

ABSTRACT

The nuclear fuel performance during accidents became a critical issue after the Fukushima Daiichi nuclear accident in 2011. Currently, various research and development programs are being carried out to enhance the fuel's reliability and durability under such conditions. These programs are collectively known as the Accident Tolerant Fuel (ATF) R&D program, which involves multiple countries, research institutes, and fuel vendors. ATF is an enhanced fuel that can tolerate longer periods of active cooling system failure, without significant fuel/cladding system degradation. Moreover, it can improve fuel performance in normal operations, transients, as well as design-basis accident (DBA) and beyond design-basis (BDBA) scenarios. This paper presents a preliminary neutronics analysis for Accident Tolerant Fuel (ATF) cladding materials for a standard PWR fuel rod (fuel pellet and dimension). The candidate cladding materials were compared with the original Zircaloy-4 cladding material. To confirm the necessary geometry requirements for achieving end-of-cycle fuel reactivity, a parametric evaluation was conducted on fuel and cladding materials. The findings were then compared with the standard PWR reference fuel-cladding system. A number of reactor safety parameters are evaluated for the candidate cladding materials as reactivity, radial power distribution of fuel pellet, reactivity coefficients, spectral hardening. This study used OpenMC code to model two-dimensional space Standard PWR nuclear fuel rods and the neutronics and burnup analysis of ATF cladding materials.

INTRODUCTION

The Fukushima Daiichi nuclear incident in 2011 underscored the critical importance of nuclear fuel performance during accidents. As a response to this concern, extensive research and development initiatives are presently underway to enhance the reliability and durability of fuel under such challenging circumstances. These collective efforts are referred to as the Accident Tolerant Fuel (ATF) Research and Development (R&D) program, involving numerous countries, research institutions, and fuel suppliers. ATF represents an advanced type of fuel capable of enduring extended periods of active cooling system failures with minimal degradation to the fuel/cladding system [1]. Additionally, it has the potential to enhance fuel performance in routine operations, transient events, as well as design-basis accidents (DBA) and scenarios beyond design-basis (BDBA) [1]. Zirconium alloys are known for their inherent resistance to various environmental conditions, making them a popular choice as cladding materials in light water reactors [2,3,4]. Despite their advantages, such as excellent neutron economy and low capture cross sections, these alloys exhibit reduced resistance to oxidation at elevated temperatures during reactor operation. This leads to increased hydrogen absorption, impacting the material's microstructure and causing a loss of ductility over time [2,3,4]. Since 2011, extensive research has explored alternative cladding materials capable of replacing Zirconium [5]. These investigations encompass a wide array of studies on mechanical properties, irradiation behavior, corrosion resistance, including reactions with water, and their interaction with fuel [6]. FeCrAl has been proposed as a promising option for fuel cladding due to its favorable thermo-mechanical attributes, reduced reactivity with steam, and lower propensity for hydrogen generation [7]. Due to its substantial aluminum content, FeCrAl forms a robust Al2O3 layer during high-temperature oxidation, characterized by superior strength [8]. Al2O3 exhibits reduced permeability in comparison to ZrO2, resulting in significantly improved oxidation properties, especially in high-temperature steam environments [8].

We conducted analyses to gain initial insights into the neutron-related aspects of employing alternative fuel cladding concepts within pressurized water reactor (PWR) cores. This study specifically focuses on a particular FeCrAl option. We compare the FeCrAl cladding material with the reference Zircaloy-4 clad fuel pin, utilizing data from depletion calculations, spectral analyses, reactivity coefficient calculations, and radial fission power assessments. Currently, the study's scope is limited to single fuel rod within a PWR; however, future work aims to expand these analyses to the assembly-level or full-core scale.

METHODOLOGY

The pin cell model was generated using the OpenMC code, which is an open-source Monte Carlo code developed by the MIT Computational Reactor Physics Group (CRPG). This code has the capability to perform fixed source, k-eigenvalue, and subcritical multiplication calculations for models constructed using either constructive solid geometry or CAD representations. Neutronic and burnup calculations were conducted using the OpenMC code. For this analysis, the ENDF/B-VII.1 library was utilized, along with the simplified CASL PWR depletion chain consisting of 255 nuclides, owing to its notable accuracy in thermal spectrum reactors. Our analysis utilized a total of 40 million particles, consisting of 400 active cycles and 100 inactive cycles, resulting in a standard deviation of approximately 13 pcm for the multiplication factor k. Figure 1 illustrates the pin cell model representing the reference case, based on a Westinghouse 17 × 17 PWR fuel rod.

(A) (B)

Fig.1. Westinghouse 17 × 17 PWR pin cell model (A), fuel-cladding system with fuel region divided into 10 rings (B).

Table 1 displays the elemental compositions for both a baseline zirconium alloy (Zircaloy-4) and an iron-chromium aluminum alloy (FeCrAl). Table 2 presents the density and the microscopic thermal neutron absorption cross-section (σ_a) associated with each cladding material. Table 3 presents cases that were utilized in reactivity calculations. Case 1 serves as the reference case, while the other cases aim to increase the concentration of heavy metals and fissile materials within the fuel rod. This is achieved by either expanding the pellet diameter at the expense of reducing cladding thickness (Cases 2–4) or by enhancing the uranium enrichment (Cases 5).

Table 1 Cladding material compositions [8]

Material	Wt%				
	Fe	Cr	Al	Zr	Sn
Zircaloy-4	0.15	0.1		98.26	1.49
FeCrAl	75	20	5		

Table 2 Density and microscopic thermal neutron absorption for cladding material [8]

Material	Density (g/cm3)	σ_a (barns)
Zircaloy-4	6.56	0.200
FeCrAl	7.1	2.43

Table 3 Various cases used for reactivity calculations

Case #	Material	Pellet OD [mm]	Clad ID [mm]	Clad OD [mm]	Clad Thickness [μm]	U-enrichment	Specific Power (MW/MTU)
1 (ref)	Zircaloy-4	8.1915	8.3566	9.4996	571.5	4.9	38.33
2	FeCrAl	8.1915	8.3566	9.4996	571.5	4.9	38.33
3	FeCrAl	8.5345	8.6996	9.4996	400	4.9	35.31
4	FeCrAl	8.6345	8.7996	9.4996	350	4.9	34.50
5	FeCrAl	8.6345	8.7996	9.4996	350	5.5	34.50
Notes: OD = Outer Diameter; ID = Inner Diameter.							

RESULTS AND DISCUSSION

Figure 2 displays the relationship between reactivity and Burnup (GWd/ton) in fuel rod. Specifically, the discharge burnup values for Zircaloy-4 and FeCrAl were 36.52 and 28.52 GWd/ton, respectively. At a burnup level of 60 GWd/ton, FeCrAl incurred a reactivity penalty of -4250 pcm, primarily due to its larger neutron absorption cross section. Table 4 presents the end-of-burnup values for different cases. In Case 1, the end-of-burnup value for Zircaloy-4 was 36.52 GWd/ton, which FeCrAl must achieve to have an equivalent the cycle length of Zircaloy-4. Cases 2-4, the FeCrAl cladding still did not achieve the cycle length of Zircaloy-4. In Case 5, the end-of-burnup value for FeCrAl was 36.45 GWd/ton, which was close to the end-of-burnup value of Zircaloy-4. After numerous simulation iterations, the optimal enrichment level was determined to be 5.5084%.

Figure 3 illustrates the neutron flux spectrum in BOC, The increased neutron-absorption cross sections of the FeCrAl cladding material lead to a hardening of the thermal neutron flux spectrum. Figure 6 displays the normalized power distribution concerning the relative radius. During operation, FeCrAl exhibits slightly higher relative power near the surface, as it produces more 239Pu in the outer ring due to spectral hardening.

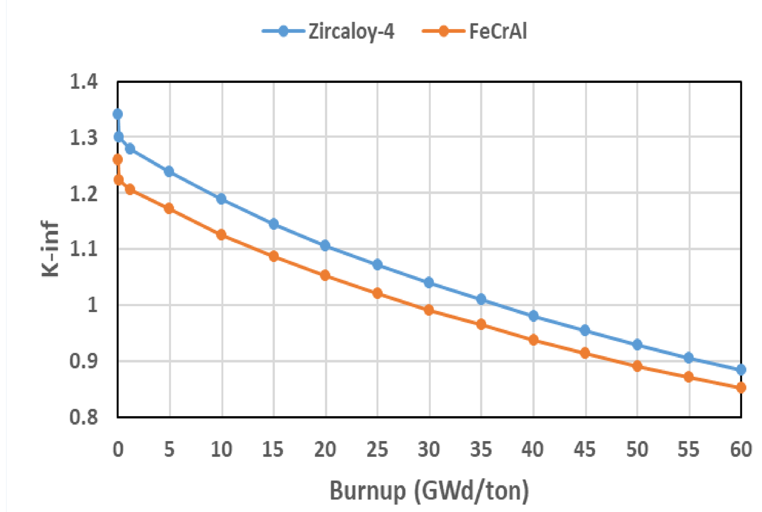

Fig.2. Infinite multiplication factor for cladding materials

Table 4 End of burnup values for various cases

Case #	Material	Clad Thickness [μm]	U-enrich	End of burnup (GWd/ton)
1 (ref)	Zircaloy-4	571.5	4.9	36.52
2	FeCrAl	571.5	4.9	28.52
3	FeCrAl	400	4.9	30.89
4	FeCrAl	350	4.9	31.45
5	FeCrAl	350	5.5	36.45

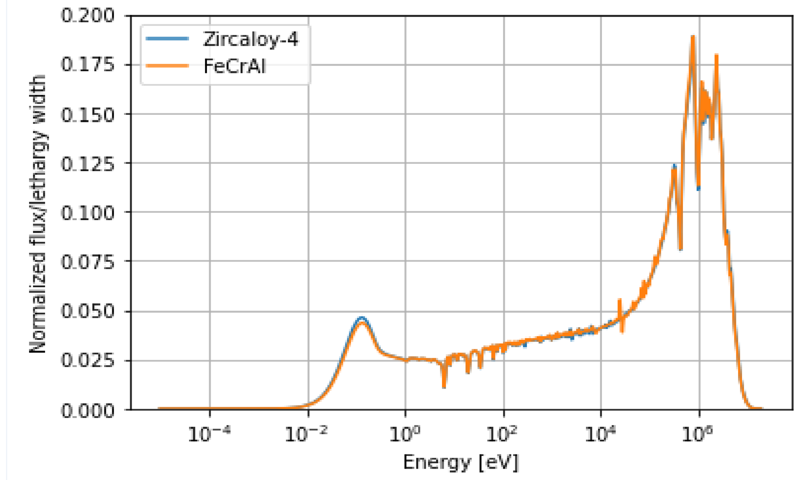

Fig.3. Neutron flux spectrum (BOC)

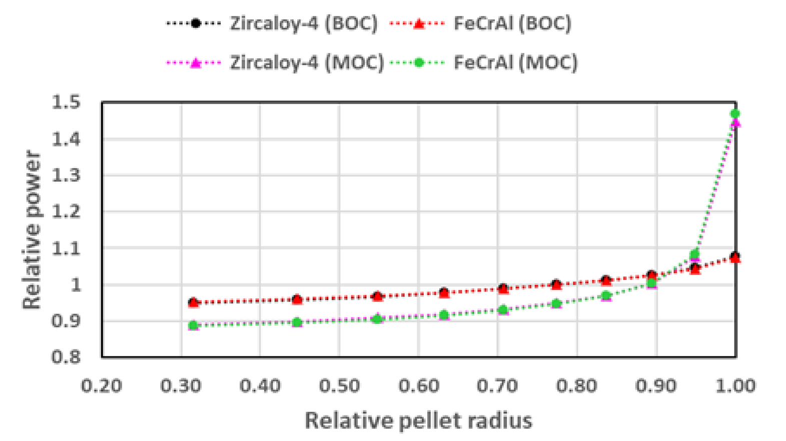

Fig.4. Relative radial power distribution

CONCLUSIONS

This study has presented a preliminary analysis of the neutron-related aspects of utilizing FeCrAl as an alternative fuel cladding material within pressurized water reactor (PWR). Several key findings have emerged from this study:

Firstly, the FeCrAl exhibited a reactivity penalty of -4250 pcm at a burnup level of 60 GWd/ton when compared to Zircaloy-4.This penalty was primarily attributed to FeCrAl's larger neutron absorption cross section. Secondly, the increased neutron-absorption cross sections of FeCrAl led to a hardening of the thermal neutron flux spectrum. Thirdly, the FeCrAl exhibited slightly higher relative power near the fuel rod's surface due to spectral hardening effects. Finally, the optimal enrichment level for a cladding thickness of 350 μm was found to be 5.5084%.

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