Potential of Silicon Carbide Cladding to Extend Burnup of Pu-Th Mixed Oxide Fuel

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INTRODUCTION

In the current generation of Light Water Reactors (LWRs), the fuel burnup is primarily limited by the performance of fuel cladding material. Mechanical properties of Zr-based alloys degrade with fuel burnup due to corrosion, hydrogen uptake and radiation damage.

Recent studies of silicon carbide ceramic composites [1,2,3] showed that they can withstand longer exposure to LWR operating conditions without degradation of their mechanical properties.

Short sections of silicon carbide triplex clad tubes have been exposed to 300 °C PWR coolant and neutron flux conditions within the MIT research reactor for periods exceeding 2 years, with irradiation exposure still continuing in 2011. It was found that the rate of corrosion depends on the exact nature of the crystal structure, the purity of the SiC and its stoichiometry. Those test specimens with high purity and good stoichiometry exhibited corrosion rates that were less than Zircaloy, and sufficiently low to expect in core lifetimes of SiC cladding in the 6 to 10 year range.

These results suggest that the use of SiC as cladding material would allow achieving significantly higher fuel burnup than the current Zircalloy clad fuel, enabling longer irradiation cycles, higher core power density, or both, improving the overall economic performance of the plant.

However, both power density increase and longer fuel cycles would require higher fuel enrichment, which is currently restricted to 5% U-235. The slightly more favorable neutronic properties of SiC such as lower than Zr absorption cross section and higher moderating power compensates for this enrichment limit, but only to a minor extent [4].

Uranium-Plutonium Mixed Oxide (MOX) fuel does not have an equivalent fissile content infrastructure limitation. Nevertheless, the high Pu loading necessary to achieve high fuel burnup is limited by the core physics considerations, more specifically, by a requirement to maintain negative void reactivity coefficient [5].

Alternatively, Thorium-Plutonium MOX will not have either of the aforementioned limitations [6]. LWR fuel lattice physics suggests that much higher initial Pu loading would be feasible, while still maintaining the core safety characteristics. Thorium has a smaller fission cross section with a higher energy threshold than U-238, while U-233 (which would be continuously produced from neutron captures in Th-232 and contribute significant fraction of the total power) has less pronounced increase in the number of neutrons released per absorption with spectrum hardening than Pu-239. These unique features of Th-containing fuel result in a negative coolant density coefficient of reactivity even at high void fractions and high Pu loadings.

The thoria (ThO_2) fuel provides a number of additional advantages over the urania (UO_2) fuel. It has a higher melting point, which allows for a higher allowable pin linear heat generation rate. It also has higher thermal conductivity, which reduces the operating fuel temperature. The lower temperature may lead to reduced fission gas release and fuel material swelling. Therefore, higher burnup would be more tolerated in fuel pins of thoria than of urania.

DESCRIPTION OF WORK

In this study, some of the neutronic effects related to the replacement of Zr cladding with SiC for Th-Pu mixed oxide fuel were investigated. We determined Pu loading requirements necessary to achieve burnups of up to 125 MWd/kg, quantified the impact of SiC cladding and high burnup on the residual Pu content in discharge fuel and evaluated a basic set of reactivity feedback coefficients.

The calculations were performed with two dimensional lattice transport code BOXER [7]. The assembly transport calculations were performed in 70 energy groups using cross section library mostly based on JEF-1 evaluated data file. BOXER code was extensively verified in the past against other state of the art computer codes and experimental data and was found capable of modeling Pu-Th fuel in LWRs with accuracy adequate for the purposes of this study. A standard 17x17 pins Westinghouse PWR fuel assembly dimensions, power density and operating conditions were used. The core average parameters were calculated by applying Linear Reactivity Model [8] to the results of 2D fuel assembly infinite lattice burnup calculations. A typical 3% core leakage reactivity worth and 3-batch refueling scheme were assumed. Reactor Grade Pu isotopic vector was taken from a typical LWR discharge fuel with 4.5% initial enrichment, 50 MWd/kg burnup and 10 years of decay following discharge.

Three cases were analyzed:

- Fuel assembly with Zircaloy cladding of standard thickness (0.057cm),
- Fuel assembly with SiC cladding of standard thickness,
- Fuel assembly with thicker SiC cladding (0.089cm) to account for possible manufacturing constraints. The outer pin diameter and the gap thickness were kept the same, which translated into slightly reduced fuel pellet diameter (0.757cm).

RESULTS

Selected results of the analyses are presented in Figures 1 and 2. For a fixed fuel cycle length, SiC cladding results in about 80 kg of Pu savings per year per 1000 MWe reactor. Employing thicker SiC cladding improves Pu requirements savings even further: up to 200 kg of Pu/GWe-Y due to additional moderation. Initial Pu content in the fuel required to achieve 100 MWd/kg burnup is 15.9 and 15.3 wt.% for SiC clad fuel with nominal and increased thickness respectively. The corresponding value for the hypothetical Zr clad case is 16.5%.

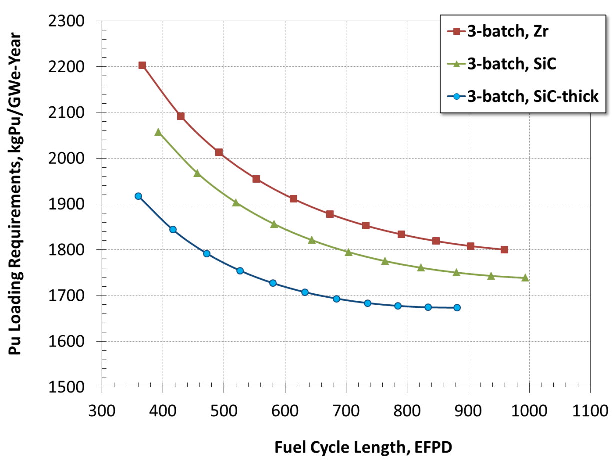

Fig. 1. Pu loading requirements.

High fuel burnup slightly improves the Pu burning capability in Th-Pu MOX. The residual Pu fraction is reduced from about 50% to 43% of the initial loading by increasing the fuel burnup from 50 to 100 MWd/kg. This is because of the relatively slow buildup of U-233 so that high burnup results in higher energy production per kg of initial Pu through more efficient in-situ burning of U-233. Additional moderation due to the use of thick SiC cladding helps to reduce the residual Pu fraction by an additional 1 to 2%.

The calculated moderator temperature, Doppler and soluble boron reactivity coefficients were found to be within the range of typical U-Pu MOX fuel and slightly more favorable for the thick SiC clad fuel cases.

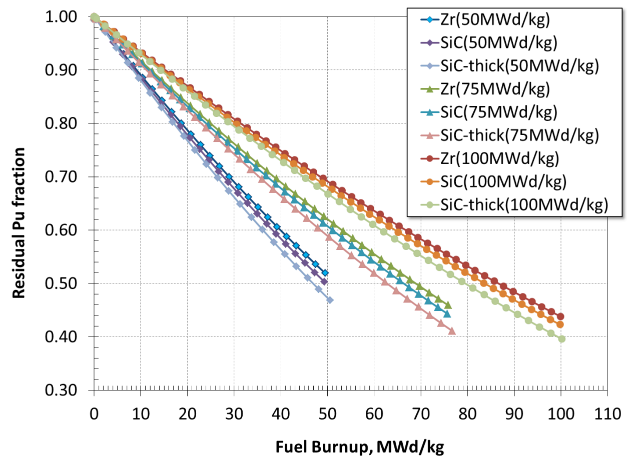

Fig. 2. Fractional Pu burnup.

CONCLUSIONS

Th-Pu mixed oxide fuel in combination with SiC cladding appears to be one of the few practical methods that would allow extending the burnup from current limit of 50 MWd/kg to 100 MWd/kg or beyond. With initial Pu loading of 19%, a batch average burnup on the order of 126 MWd/kg can be achieved, which is greater by a factor of 2.5 than is feasible with Zircaloy clad UO_2 fuel with 5% enrichment. This leads to a major reduction in spent fuel produced per kWh. To achieve such burnup at the standard power density requires three 32.7 months cycles (8.2Y total), which is well beyond the reach of Zircaloy clad fuel.

Apart from the obvious SiC clad fuel manufacturing and Pu-Th MOX fuel performance uncertainties that would need to be resolved to confirm feasibility of such a high burnup and long fuel cycle core design, more detailed neutronic study would be required to assure that all the reactivity control and power peaking design limits are satisfied.

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