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Thermal-Hydraulics Assessment of SiC Compared to Other ATF Cladding Materials and Performance to Mitigate CRUD

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Program: L-1 Accident Tolerant Fuels

ABSTRACT:

Since the 2011 Fukushima accident, a world effort has been initiated in the development of accident tolerant fuel (ATF) cladding materials to replace zirconium with oxidation-resistant materials. Among the several ATF materials proposed, chromium-coated zirconium, FeCrAl, and SiC are highlighted. While SiC has the lowest Technology Readiness Level of the three, if it can be made to work with a sufficient seal connection on the end of the fuel pins, it has the potential to eliminate any hydrogen generation and increase the maximum operation of the cladding to a point where a critical heat flux (CHF) event may not damage the cladding. This would have significant benefits to increasing safety and may allow significant uprating. A further evaluation of the effect of these ATF cladding materials on CRUD (Chalk River Unidentified Deposits) or insoluble potentially radioactive particles that could affect fuel performance is necessary. The different clad materials may help to mitigate/reduce such deposits due to different surface energy. CRUD can not only limit the heat transfer through an added conduction resistance but also affects the surface wettability and could alter CHF. Understanding these impacts requires specific testing along with detailed measurements of the contact angle when CRUD is present. The proposal aims to experimentally investigate the thermal-hydraulics performance of these cladding materials (specifically focusing on SiC) under accident scenarios, including both Departure from Nucleate Boiling (DNB) and dryout conditions, and evaluate the effect of the ATF materials in suppressing CRUD.

The following six objectives will be realized during the course of the project.

1. Develop and build SiC cladded heaters that able to operate as simulated fuel pins under LWR conditions.
2. Generate simulated CRUD deposits on zirconium, Cr coated zirconium and SiC, and assess their effect on the contact angle under LWR conditions in a steam environment.
3. Obtain CHF data under prototypical LWR conditions for SiC.
4. Evaluate the behavior and survivability of SiC under RIA and quenching events.
5. Assess the post-CHF heat transfer and behavior of SiC claddings under prototypical LWR conditions.
6. Train several graduate students who will contribute to the nuclear industry