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Engineered Zircaloy Cladding Modifications for Improved Accident Tolerance of LWR Fuel

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Program: IRP: Advanced Nuclear-Cladding and Fuel Materials with Enhanced Accident Tolerance for Current Generation & GEN III + Light Water Reactors

ABSTRACT

An integrated research project (IRP) to fabricate and evaluate modified Zircaloy LWR cladding under normal BWR/PWR operation and off-normal events is proposed. A combination of computational and experimental protocols will be employed to design and test modified Zircaloy cladding with respect to corrosion and accelerated oxide growth, the former associated with normal operation, the latter associated with steam exposure during loss of coolant accidents (LOCAs) or reactivity insertion accidents (RIAs) and low-pressure core refloods. Cladding performance evaluation will be incorporated into a reactor system modeling effort of fuel performance, neutronics, and thermal hydraulics, thereby providing a holistic approach to accident tolerant nuclear fuel. The proposed IRP brings together personnel, facilities, and capabilities across a wide range of technical areas relevant to the study of modified nuclear fuel and LWR performance during normal operation and off-normal scenarios.

Two pathways toward accident tolerant LWR fuel are envisioned, both based on the modification of existing Zircaloy cladding. The first is the modification of the cladding surface by the application of a coating layer designed to shift the M+O→MO reaction away from oxide growth during steam exposure at elevated temperature. This pathway is referred to as the “surface coating” solution. The second is the modification of the bulk cladding composition to promote precipitation of minor phase(s) during fabrication. These precipitates will be stable under normal operation, but dissolve during the temperature excursions; the migration of solute elements to the free surface will then shift the reaction away from oxide formation. This pathway is referred to as the “bulk self-healing” solution. A synergistic response of the fuel rod is anticipated in which the combined mitigation of brittle exothermic oxide formation and associated reduction in cladding temperature lead to accident tolerance with respect to cladding failure. The proposed cladding modifications potentially may influence neutronics and thermal hydraulics, both under normal operation and off normal scenarios; a favorable reactor system response must therefore be demonstrated for both solution pathways.

The objectives of the proposed IRP is four-fold: i) demonstrate of performance of modified cladding material under normal BWR and PWR operation with respect to corrosion, in particular, stress corrosion cracking (SCC) and irradiation-assisted stress corrosion cracking (IASCC); ii) the mitigation of accelerated cladding oxidation during off-normal scenarios that fall below unchecked LOCA/RIA events,

as well as uncovering scenarios that involve used fuel in on-site storage pools; iii) the benchmarking of the fuel performance code against the databases developed in i) and ii); and iv) demonstration of overall reactor system performance with the proposed modifications to the pellet and cladding.