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FLUORIDE SALT HIGH TEMPERATURE Reactor Materials IRRADIATION TESt at the MIT Research Reactor

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A new irradiation test of structural materials and surrogate TRISO fuel particles in a molten, fluoride-based salt has been designed and carried out at the Massachusetts Institute of Technology (MIT) Research Reactor (MITR). The MITR is owned and operated by MIT and is licensed to operate at a power of up to 6 MW. The reactor power density and spectrum are similar to those found in light water power reactors, which makes the MITR an extremely useful test bed for a wide range of nuclear material and fuel research. In recent years, the MITR has utilized four types of in-core experimental facilities:

(1) a flexible, general purpose in-core irradiation space (the in-core sample assembly - ICSA) for dry experiments at temperatures up to 900°C,

(2) a high temperature water loop that can be operated at pressurized water reactor (PWR) or boiling water reactor (BWR) conditions,

(3) an experimental fuel irradiation facility, and

(4) a very high-temperature irradiation facility (HTIF) for irradiations in the range of 1200-1400°C.

The irradiation test described here is part of an ongoing joint research program being conducted at MIT, the University of California-Berkeley (UCB), and the University of Wisconsin-Madison (UW). The objective of the overall research program is to develop a path forward to a commercially viable, fluoride-salt-cooled, high-temperature reactor (FHR). The baseline FHR concept combines a fluoride salt coolant called flibe (a mixture of LiF and BeF2), with a graphite-matrix, coated-particle fuel. The boiling point of flibe is above 1200°C, and the FHR fuel failure temperature is above 1600°C. However, due to the limitations of existing structural materials for reactor vessels and heat exchangers, the proposed peak operating temperature for the FHR is 700°C. The objectives of the first FHR irradiation experiment at the MITR are:

(1) to assess the corrosion and compatibility of 316 stainless steel, Hastelloy N®, SiC and SiC/SiC composites, and surrogate TRISO fuel particles in molten flibe, and

(2) to examine the partitioning of tritium (produced when the flibe is subjected to neutron irradiation) among the various media in the experiment.

The experiment includes six, flibe-filled chambers in a graphite sample holder, each chamber containing one type of test material. Two of the chambers are lined with 316SS and nickel for testing 316 SS, Hastelloy N respectively. The sample assembly is sealed inside a nickel capsule which allows for sampling gases produced during the irradiation such as gas-phase tritium species. The test will be performed with a maximum temperature of 700°C. Parallel corrosion and compatibility tests are being performed, without irradiation, by researchers at UW. The experimental design, thermal analysis, and trace elemental composition of the flibe are presented.