MECHANICAL PROPERTIES OF IRRADIATED ZIRCALOY-4 FOR DRY CASK STORAGE CONDITIONS AND ACCIDENTS

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The mechanical integrity of fuel cladding is essential for mitigating fuel dispersal during pre-storage vacuum drying and transfer, during dry cask storage, and during transfer from the dry cask to a permanent repository site. Rupture may occur during handling or transportation accidents if cladding ductility has been reduced due to the separate or combined effects of pre-existing flaws, hydride concentration and orientation, radiation-induced hardening, and creep strain accumulated during vacuum drying and dry storage. Therefore, it is important to determine the cladding mechanical behavior under a variety of microstructural conditions, temperatures and strain rates. Axial tensile tests are in progress at ANL to address part of the data needs for dry cask storage. These tests are being conducted in the temperature range of 20-400°C and the strain-rate range of 0.1-100%/s. Zircaloy-4 (Zry-4) test materials include: non-irradiated, low-Sn Zry-4, irradiated-and-dry-cask-stored Zry-4 at 36 GWd/MTU, irradiated low-Sn Zry-4 at 50 GWd/MTU, and irradiated Zry-4 at 67 GWd/MTU. All Zry-4 materials are from 15×15 PWR cladding that has been cold-worked and stress-relieved. Additional tests are in the planning stage for pre-annealing the high-burnup Zry-4 at 400°C for ˜ 1 week to simulate the effects of vacuum drying and for performing three-point bend and impact tests on thermal creep samples that have experienced >1% creep strain.

After years in dry storage, it is possible for cladding to experience permanent plastic strains resulting from thermal creep (applied stress < yield strength), which may reduce the residual ductility during high stress (> yield strength) loading associated with handling or transportation accidents. Although results from ANL thermal creep tests suggest that cladding rupture does not occur even after 5% hoop creep strain, the increase in defect density (namely from creep-induced dislocations) may further degrade the strain-hardening properties or ability to plastically deform in a material already hardened by radiation-induced defects. With the current regulatory guidance limiting pre-storage and storage temperatures to =400°C, it is important to understand the effects of thermal annealing on cladding strength and ductility. Depending on the dry storage time, cladding temperatures may be reduced to about 150-200°C at the end of dry storage and prior to the repository-site transfer. Thus, it is important to determine the mechanical properties in the dry-storage-relevant temperatures of 150-400°C, as well as the pool-storage temperature of <100°C, for a range of strain rates.

Moreover, the residual ductility after creep deformation may further be reduced due to pre-existing flaws or cracks oriented along the radial direction of the cladding. These cracks may initiate within a layer of highly-dense, circumferentially-oriented hydrides near the outer surface and may result in stress-concentration sites (sharp cracks), or they may limit load-carrying capacity of the cladding (blunt cracks). Furthermore, radially-oriented hydrides may be present in the cladding due to temperature-pressure history, increasing the possibility for through-wall crack initiation and propagation due to strain incompatibility with the surrounding base alloy.

This paper presents a preliminary database for the temperature-dependent mechanical properties of the following irradiated, cold-worked, stress-relieved Zircaloy-4 cladding materials, as well as the baseline properties of non-irradiated Zircaloy 4:

- Surry-2 to 0.7×10^{26} n/m^2 (E > 1 MeV), average fuel burnup of 36 GWd/MTU, =40 µm oxide layer, and containing 250-300 wppm hydrogen. This material was placed in a dry storage cask (Castor-V/21) for 15 years with varying environmental and thermal histories during code benchmarking studies conducted prior to storage;

- Three Mile Island-1 to 0.9×10^{26} neutrons/m^2 (E > 1 MeV), average fuel burnup of 50 GWd/MTU, =30 µm oxide layer, and containing ˜ 170 wppm hydrogen; and

- H.B. Robinson to 1.4×10^{26} neutrons/m^2 (E > 1 MeV), average fuel burnup of 67 GWd/MTU, =100 µm oxide layer, and containing =800 wppm hydrogen.

Initial tensile testing of these cladding materials indicate significant strength hardening and reduction in strain hardening between room temperature and 400°C, as compared to non-irradiated materials. Hardening of irradiated material is manifested as an increase in mechanical strength and a reduction in ductility, consistent with mechanisms caused by a high density of radiation-induced defects, such as dislocation loops. Additional testing is scheduled to expand the mechanical properties database for Zircaloy-4 and to increase our understanding of the effects of hydrogen content and neutron exposure on these properties.