Hydride LWR Fuel Rod Irradiation

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The principal difference between oxide and hydride fuel is the high-thermal conductivity of the latter. This feature greatly decreases the temperature drop over the fuel during operation, thereby reducing the release of fission gases to the fraction due only to recoil. The maximum fuel temperature can be further reduced by filling fuel-cladding gap with low-melting LM instead of He.

The objectives of this experimental research are to study the materials issues associated with the use of a hydride fuel for power production in light water reactors (LWRs) and to explore the use of Pb-Bi Eutectic liquid metal (LM) as a replacement for helium in hydride fuel elements. Hydrides provide a number of improvements, including the addition of hydrogen neutron moderation within the fuel, thermally induced hydrogen up-scattering that accompanies Doppler Feedback that improves safety, and higher efficiency in elimination of plutonium than achievable with mixed oxide (MOX). The liquid metal bounded fuel-cladding gap assist in lowering the temperature of the fuel.

Project Description

Feasibility and benefits of incorporation of hydride nuclear fuels into the current fleet of LWRs have been investigated in detail using neutronic and thermal-hydraulic calculations and laboratory-scale materials experiments by the materials group in Nuclear Engineering Department at the University of California at Berkeley. Recognizing the necessary shift from laboratory-scale experiments to more relevant environments, an irradiation experiment was then conducted to evaluate the feasibility of an LWR hydride fuel. Sealed fuel rodlets were constructed with U0.17ZrH1.6 fuel pellets and conventional Zircaloy-2 cladding with Bi-Pb LM filled-gap, inner and outer surfaces oxidized to nominal 1 micron to prevent hydrogen attack. They were housed in Ti capsules surrounded with the same LM used for filling gaps between the fuel and cladding. Three irradiated Ti capsules were then irradiated at the Massachusetts Institute of Technology (MIT) research reactor to burnups of 0.19, 0.17, and 0.29 FIMA (%) with up to a maximum 6-MW thermal power. Along with a fourth unirradiated rodlet for reference use, the capsules were then transferred to PNNL for extensive post-irradiation evaluations. In this fiscal year, novel tools were designed and fabricated to handle and extract the highest-burnup rodlet (0.29% FIMA) from the outer Ti capsule. Thin-sliced samples were prepared at two axial locations of the rodlet: close to the axial mid-level of the rod where TC tips were located, and near to the end of rodlet. The full-round cross-sections were mounted and polished to observe the fuel, gap LM, outer LM, cladding, and their interfaces. Because of radioactivity and pyrophoricity concerns, metallographic samples preparation were performed in hot cells under flowing inert gas.

The irradiated samples were evaluated in detail and compared to unirradiated sliced sample from the similar axial location through visual inspection for possible crack development, optical microscopy for detecting dimensional changes in the fuel, cladding, and the gaps and scanning electron microscopy

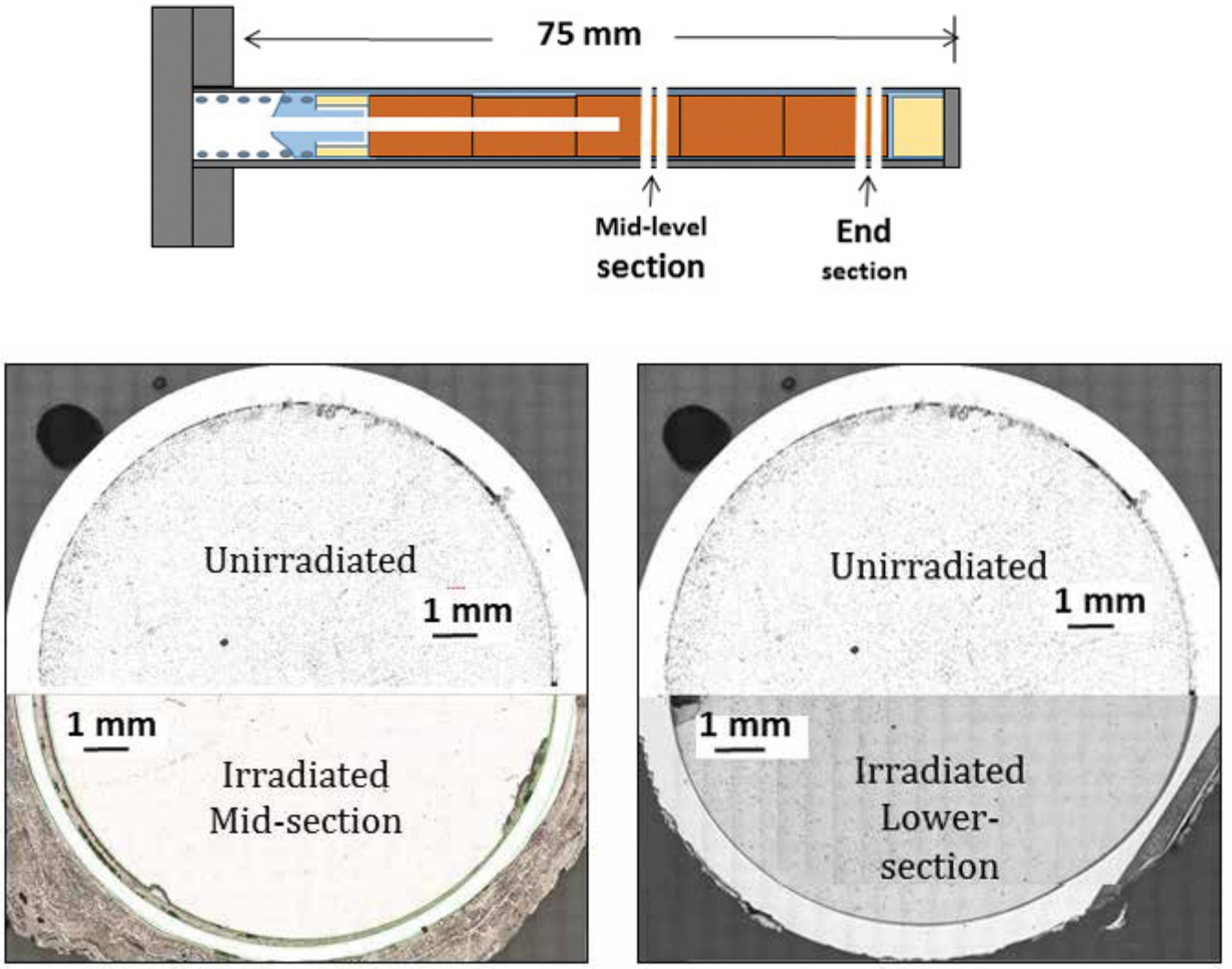

Figure 1. Mid- and lower-level cross-sectional optical images of irradiated fuel rodlet as overlapped to unirradiated fuel.

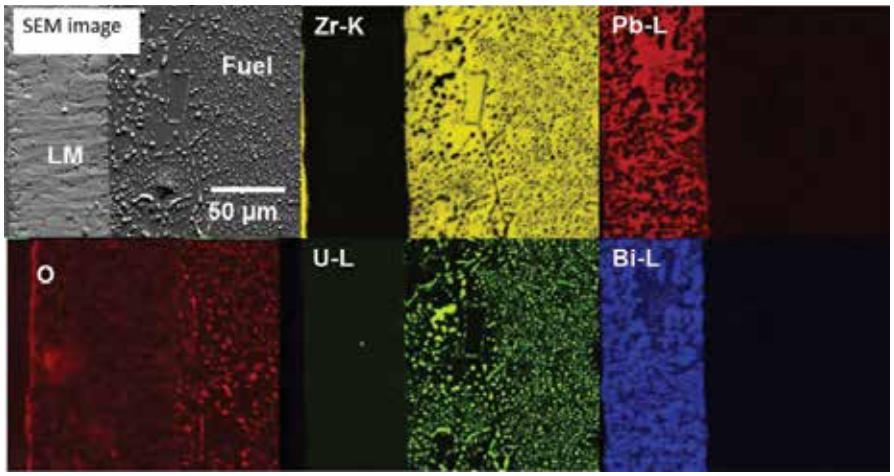

Figure 2. Elemental map including inner liquid metal, fuel, and cladding of the unirradiated fuel rodlet.

Collaboration with first-class material scientists and access to PNNL advanced facilities resulted in the successful completion of the project.

— Mehdi Balooch, Co-principal Investigator

with X-ray microanalysis (SEM/ energy dispersive X-ray spectroscopy [EDS]) for detection of swelling, void formation, corrosion of cladding, dissolution of zircaloy and fuel in LM gap, and finally LM chemical changes due to temperature cycles during irradiation.

Accomplishments

In this progress report, we present the post-irradiation analysis of one of the rodlets, the 0.29% FIMA, using high-resolution optical and SEM/EDS at two height-level axial positions (mid- and lower-level).

Comparing the unirradiated with irradiated fuel rod images by high-resolution optical microscopy (Figure 1) we have observed extensive thinning, mainly from the outside surface of the cladding at the mid-level cross section, where the cladding prematurely failed during irradiation at the MIT reactor. This persuaded, but to much lesser extent, at lower-level cross section where it was limited to ~8%. Limited availability of the LM for dissolution of Zr in the fuel-cladding gap impeded thinning of the inside surface of the cladding to negligible amount. In contrast, the larger outer LM reservoir was a more-efficient sink for dissolution of the cladding, which resulted in massive and surprising thinning of the cladding at mid-level. This zircaloy loss is an artifact of the experiment. In actual use, the outer cladding surface is contacted by water, not by LM.

The loss of cladding material, suggests the dissolution of Zr in LM passes beyond the solubility limit. But it continued by generating new phases such as $ZrBi_2$, ZrBi, Zr_3Bi, Zr_5Pb_3, and possibly $Zr_{3.33}$ $Pb_{0.67}$. U phase of UBi_2 may also be formed. High-resolution elemental imaging of unirradiated and irradiated fuel (Figures 2 and 3) reveal the dissolved Zr was distributed uniformly in the LM gap. The fuel surface interaction with the LM gap was limited—some evidence of dispersed micron-size uranium phase of the fuel in the LM gap adjacent to the fuel surface and its aggregation close to the inner cladding surface was found. The bulk of the hydride fuel appeared intact—no detectable swelling of the fuel, or its microstructural changes within the capabilities of the instruments was observed.

The dissolution of 8% of zircaloy-2 cladding in lead-bismuth eutectic Bi-Pb LM during 1000 hours of exposure was found excessive for use as a gap-filler in a nuclear reactor. However, the addition of

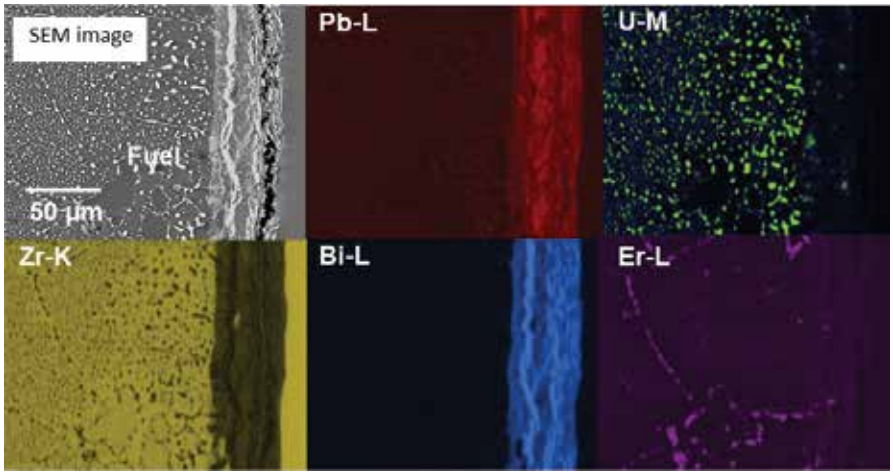

Figure 3. SEM and elemental maps of Pb, Bi, U, and Zr and Er revealing fuel and cladding interaction at their interface of the lower-level section of irradiated fuel rodlet.

third component such as Sn, may limit the dissolution by forming intermetallic compounds that provide a barrier to some and extend kinetically, bringing it closer to an acceptable level.

Future Activities

This project is now complete.

Publications and Presentations*

1. Casella, A. M., D. J. Senor, E. C. Buck, D. R. Olander, K. A. Terrani, P. Hosemann, M. Balooch, "Post Irradiation Experiments on Hydride LWR Fuel Rodlet" In preparation.
2. Kima, S. J., D. Carpentera, G. Kohsea, L-w. Hua, "Hydride fuel irradiation in MITR-II: Thermal design and validation results," *J. Nuc. Mat.*, Vol. 277, 2014, pp. 1–14.
3. Terrani, K. A., M. Balooch, and D. R. Olander, "LWR Hydride Fuel Irradiation: Experiment Design and Fabrication" In preparation.

Distributed Partnership at a Glance	
NSUF and Partners	Facilities and Capabilities
Massachusetts Institute of Technology	Nuclear Reactor Laboratory
Collaborators	
Idaho National Laboratory	James Cole (principal investigator)
Pacific Northwest National Laboratory	Andrew Casella (collaborator), David Senor (collaborator)
University of California, Berkeley	Donald Olander(principal investigator), Mehdi Balooch (co-principal investigator)