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Using diffraction line profile analysis to understand irradiation-induced growth in nuclear fuel cladding

Michael Preuss[a, b], Philipp Frankel[b] and Tamas Ungar[b, c]

[a] *Department of Materials Science and Engineering, Monash University, Clayton, Australia*

[b] *Department of Materials, University of Manchester, UK*

[c] *Department of Materials Physics, Eötvös University, Budapest H-1518 POB, Hungary,*

michael.preuss@monash.edu or michael.preuss@manchester.ac.uk

Nano-scale dislocation loops are a prominent feature in irradiated material affecting greatly the properties of the material. In water-cooled reactor, nuclear fuel is separated from the water-coolant and moderator by zirconium alloy tubes, which are used because of the low neutron cross section, excellent corrosion performance and reasonable mechanical properties of zirconium. Sitting right next to the fuel, the zirconium alloys are exposed to high levels of irradiation during service. In such environment, zirconium is known to display hardly any swelling but instead macroscopic growth behaviour along the axial direction of the tube, with the direction of growth being related to the texture of the alloy.

It is well established that alloy chemistry also plays a very significant role on the growth behaviour with some alloys showing dramatic acceleration of growth, so-called breakaway growth, at high neutron fluence levels. Breakaway growth is typically associated with <c> loop formation, which are not observed during the early stages of irradiation where only nano-scale <a> type dislocation loops are seen.

The presentation will focus on some specific aspects when using diffraction line profile analysis for quantifying small irradiation-induced dislocation loops and present the results obtained in this way in context of trying to understand the irradiation-induced growth behaviour of zirconium alloys.

Keywords: irradiation damage, dislocation loops, zirconium alloys, diffraction line profile analysis