LOW-ENRICHED RESEARCH REACTOR FUEL: POST-IRRADIATION EXAMINATIONS AT SCK•CEN

Background

Generally, research and test reactors are fuelled with fuel plates instead of pins. In most cases in the past, these plates consisted of high enriched (>95% ^{235}U) UAl_3 powder mixed with a pure Al matrix (called the "meat") in between two aluminium alloy plates (the "cladding"). These plates are then assembled in fuel elements of different designs to fit the needs of the various reactors.

Objectives

Since the 1970's, efforts have been going on to replace the high-enriched, low-density UAl_3 fuel with high-density, low enriched (<20% ^{235}U) replacements. This search is driven by the attempt to reduce the civil use of high-enriched materials because of proliferation risks and terrorist threats. American initiatives, such as the Global Threat Reduction Initiative (GTRI) and the Reduced Enrichment for Research and Test Reactors (RERTR) program have triggered the development of reliable low-enriched fuel types for these reactors, which can replace the high enriched ones without loss of performance. Most success has been obtained with U_3Si_2 fuel, which is currently used in many research reactors in the world. However, efforts to search for a better replacement have continued and are currently directed towards the U-Mo alloy fuel (7-10 w% Mo).

Principal results

SCK•CEN has been involved in the international development of these low enriched Research Reactor fuels for several years now, particularly through the collaboration with the CEA in their fuel qualification and licensing efforts for the Jules Horowitz Reactor (RJH). Over the past years and also at the current moment, post-irradiation examination (PIE) campaigns have been carried out in the framework of irradiations of novel fuel plate designs that were performed in the BR2 reactor. Two campaigns, including the current one, have revolved around the further optimisation of U_3Si_2 fuel plates, while a third one was oriented towards the development of U-Mo alloy fuel. These studies have led to the development of novel or adaptation of existing non-destructive testing methods to use them on plate geometries instead of pin geometries for which they are designed and to several unique microscopy and spectroscopy results.

In the framework of the first irradiation, which was intended to survey the performance limits of U_3Si_2 fuel plates with AG3-NE cladding, 2 fuel plates were examined after failure under irradiation. It was found that the failure of the fuel plates was entirely related to the deterioration of the cladding and not to the degradation of the fuel [1].
Non-destructive analysis of the plates by profilometry, using a measuring bench specifically developed for measuring on curved fuel plates and by gamma spectrometry, revealed an important swelling at the location of the highest flux in the reactor (see
Figure ***1***).

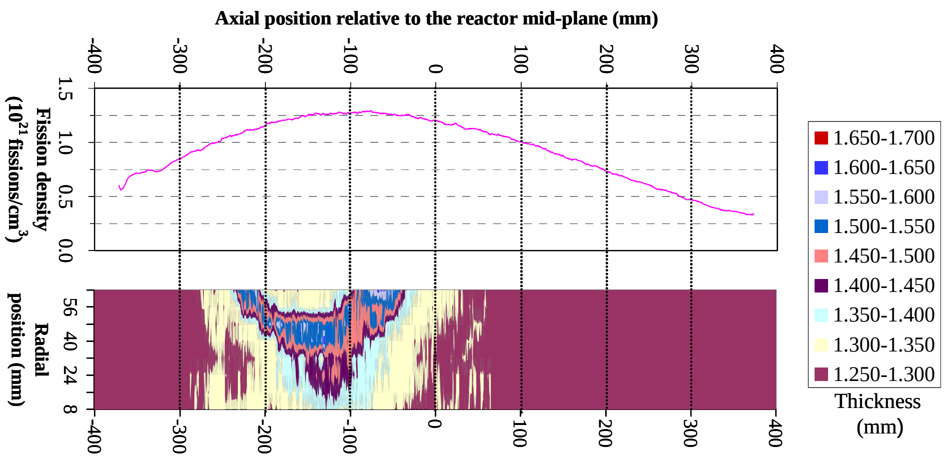

Figure 1 : The burn-up profile (top) and the thickness measurements (bottom) of the fuel plate show that the most deformed area coincides with an area of high burn-up and high heat flux, respectively 1.29×10^{21} fissions/cm^3 and 550 W/cm^2.

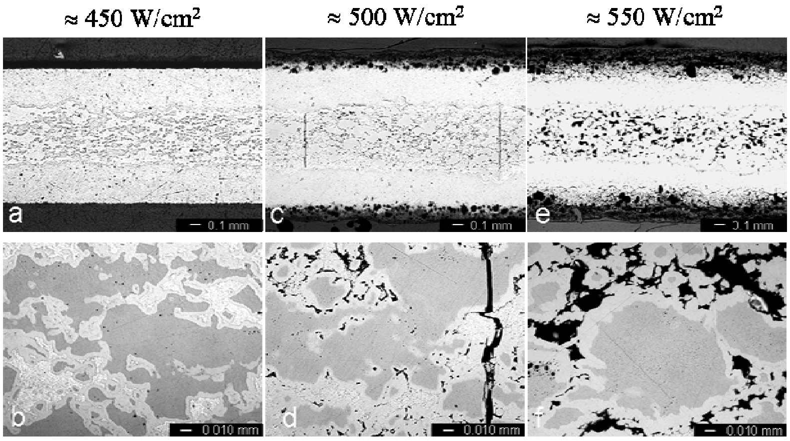

Figure 2 : Image of the fuel plate over the full width and a micrograph of the meat from the section of the fuel plate that was submitted to ≈450 W/cm^2 (a,b), ≈500 W/cm^2 (c,d) and ≈550 W/cm^2 (e,f).

The microstructural analysis of samples cut from the plates at positions corresponding to different heat fluxes in the reactor (see Figure 2), shows the evolution of the fuel plate as the corrosion of the cladding progresses. These results effectively demonstrated the perfect behaviour of the U_3Si_2 fuel itself under even the harshest conditions and revealed the limitations of the AG3-NE cladding.

The second irradiation was performed in the framework of the U-Mo alloy fuel development and involved 2 full-size flat fuel plates of atomised U-7w%Mo dispersion fuel in an AG3-NE cladding. The irradiation was performed in the FUTURE rig in the BR2 reactor and was stopped prematurely due to the observation of pillowing of the fuel plates. The observed fuel plate failure was analysed both non-destructively and destructively and was diagnosed as a fuel swelling problem. The experiment demonstrated clearly the relation of this unsatisfactory behaviour to the undesired properties of the U(Mo)-Al interaction phase (see Figure 3) that is formed under irradiation [2].

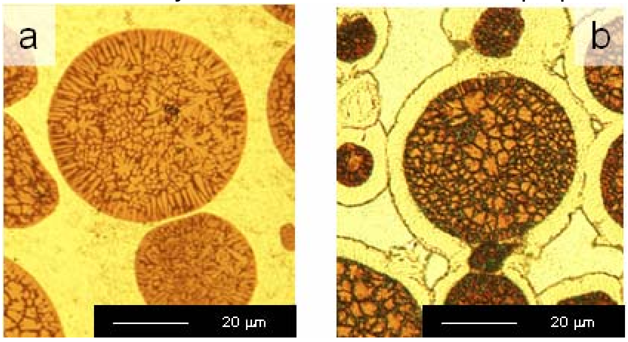

Figure 3 : Fuel particles in the unirradiated U-7wt% Mo fuel plate (a) and in the irradiated fuel plate (b). The interaction layer formed between the fuel particles and the matrix is visible in (b).

The current PIE campaign involves 3 U_3Si_2 fuel plates with AlFeNi cladding, in which the attention is mostly devoted to the alternative cladding material. These plates were irradiated at low temperature, high flux conditions and showed no failure up to their target burn-up (55% ^{235}U). Non-destructive testing methods were developed to measure oxide thicknesses on curved fuel plates with eddy current methods. The results revealed an oxide profile that is in agreement with the flux profile of the plate in the reactor. Microscopic analysis is being performed on samples taken at selected locations on the plate to further examine the cladding and fuel behaviour.

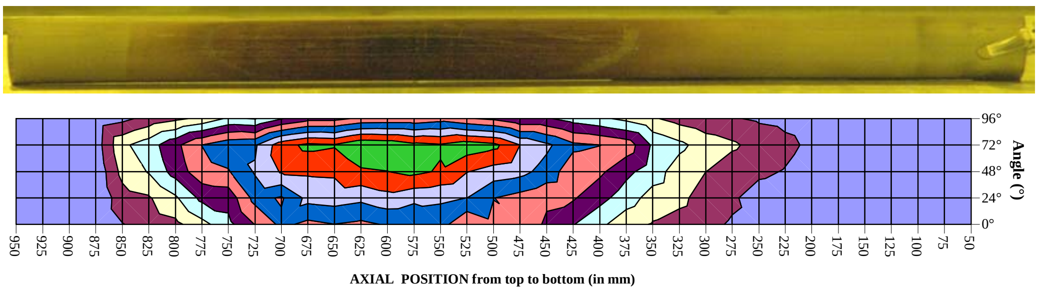

Figure 4 : The oxide thickness distribution on one of the fuel plates involved in the third irradiation. The top image shows the plate itself, while the graph at the bottom shows the distribution of the oxide thickness.

Future work

With the development of the EVITA irradiation loop for testing fuel in true RJH-conditions in the BR2 reactor, the continuation of these PIE programs is expected up to and beyond 2010. More PIE is also expected in the frame of the further development and qualification of the U-Mo alloy fuel or other alternative fuel types.

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Main references

[1] A. Leenaers, S. Van den Berghe, E. Koonen, P. Jacquet, C. Jarousse, B. Guigon, A. Ballagny and L. Sannen, "Microstructure of U_3Si_2 fuel plates submitted to a high heat flux", J. Nucl. Mater. 327(2-3) (2004), 121-129.

[2] A. Leenaers, S. Van den Berghe, E. Koonen, C. Jarousse, F. Huet, M. Trotabas, M. Boyard, S. Guillot, L. Sannen, M. Verwerft, "Post-irradiation examination of U-7wt% Mo atomized dispersion fuel", J. Nucl. Mater. 335 (2004),39-47.