Temelin Irradiated Cladding Project – TIRCLAD

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Abstract. It is widely known, that zirconium-based alloys have been used for nuclear fuel
cladding fabrication for decades [1], [2]. Therefore, it is crucial for fuel vendors, utilities as well
as safety authorities to understand their behaviour during reactor operation with strong fast
neutron and gamma irradiation. To complement the information about material properties for
current as well as prospective advanced zirconium-based alloys, a long term research project
implemented by ALVEL has been carried out, in cooperation with its partners (ČEZ, Škoda
Nuclear Machinery, UJV Rez, Research Centre Rez) [3], [4]. The project is implemented on the
background of a large material research program sponsored by JSC TVEL and ČEZ (this
program started in 2012 and it is planned to be finished in 2024). It is fully independently
implemented in the Czech Republic.

Irradiation of material samples started at Temelin NPP in 2014 and finished in spring 2020. The
first three batches were already transported to UJV Rez and Research Centre Rez for post-
irradiation evaluation. After finishing testing of reference non-irradiated samples testing in 2018
determination of initial state, mechanical properties and as-received microstructure), six batches
of irradiated material have been analysed sequentially with their increasing neutron fluence. The
final irradiation damage of the last sixth batch corresponds approximately to fuel burnup of 80
MWd/kgU.

This paper presents the developed methodologies and the expected experimental and other
important outcome. Several material properties are being studied on different scales including
high resolution microstructural and chemical analysis, creep and mechanical behaviour,
nanoindentation, oxide layer morphology. The main objectives of the project include evaluation
of materials' microstructural and bulk properties and the derivation of dose- and temperature-
dependent correlations that can be implemented into FEM models and fuel performance codes
such as FRAPCON/FRAPTRAN [5], [6] or Transuranus [7]. These models can be then used to
support licensing of new designs influenced by fuel cladding behaviour during long term dry
storage, justification of dry storage for an extended period of time or justification of new safety
criteria that can lead to enhanced operation of VVER reactors and spent nuclear fuel storage [8].

Keywords: fuel cladding, irradiation, fuel performance, post-irradiation examination, FEM
models

1 Introduction.
The presented project is realized on the background of a large material research program sponsored by
JSC TVEL and ČEZ. It is fully independently implemented in the Czech Republic. The project has
started in 2012 and it is scheduled to finish in 2024. Figure 1 shows the timeline of the project.

For this purpose, a set of six Material Cluster Assemblies (MCA) was designed by experts from
OKBM, Nizhniy Novgorod and Bochvar institute, Moscow, and then manufactured in
Mashinostroitelny zavod (MSZ) in Elektrostal, Moscow region. Their main components from the project...
point of view are the ampoules made of E110 alloy cladding material containing three types of neutron fluence monitors and material samples. All the MCAs were loaded into the reactor core of Temelin NPP Unit 1 in 2014. After completion of 1 to 5 irradiation cycles (expected neutron fluence after 5th irradiation cycle is 5.3 \times 10^{26} \text{ neutrons / m}^2 \text{ according to core design models} = \text{VVER fuel burnup of approx. 80 MWd/kgU}), ampoules with their content are separated from the cluster assembly in the unique POMA device, an equipment specially designed and manufactured for their separation by Skoda Nuclear Machinery in Pilsen. Separated ampoules with samples are transported to the hot-cell facilities in Rez for material analyses, testing and further investigation. Before transport to the Czech Republic and their loading into the reactor core of Temelin NPP Unit 1, a licensing process at the State Office for Nuclear Safety (SONS) was successfully concluded. The design of MCAs is presented in Figure 2 and their main parameters are presented in Table 1. Reference non-irradiated materials are studied first to determine initial states and as-received microstructure [9], [10]. Six batches of samples will be studied over time with increasing neutron dose of every batch. The neutron dose and neutron spectra of every batch are evaluated both experimentally and by simulation [11]. The neutron dose and spectra are calculated using sophisticated models and they are also evaluated based on activation of three types of neutron fluence monitors that are placed inside the assemblies, in the middle of ampoules.

Figure 1. Timeline of the project

Figure 2. Schematics of the MCA irradiated at the Temelin core (not in scale). 1 – MCA end plug; 2 – Ampoules with samples to be separated and analysed; 3 – Joint with the thinned region for separation; 4 – Spacer rodlet (finger); 5 – Screws; 6 – MCA (RCCA) Spider.

| Parameter                        | Value |
|----------------------------------|-------|
| Number of ampoules in each MCA   | 6 pieces |
| Length of the ampoule (separated part) | \~ 300 mm |
| Outer diameter of cladding       | 9.1 mm |
| Cladding Material                | E110  |
| Mass of one MCA                  | 8 kg   |
| MCA design irradiation time      | 6 years |

Table 1. Main parameters of MCAs

2 AIM OF THE PROJECT AND SCOPE OF ANALYSES.

2.1 Material tests and characterization
The main objectives of the project are to study cladding performance during normal operation of reactors and during wet and dry storage of the spent nuclear fuel. Therefore, a set of experiments were defined
to support the development of new models and correlations that can be used in (fuel performance) codes. These models that are directly derived from experiments with irradiated materials can be later used for licensing of the new design of fuels, an extension of dry storage periods and for support of current or new operational limits and technical specifications. The geometry of samples and ampoules are being measured to estimate irradiation growth including anisotropic effects before destructive mechanical testing and characterization [12]. Material analyses of samples placed inside the irradiated ampoules are the part of the above mentioned research program sponsored by TVEL and ČEZ. These analyses include PDE measurement and evaluation, irradiation induced growth measurement, metallographic analysis, TEM analysis and neutron fluence evaluation.

2.1.1 Mechanical Testing. The experimental matrix of mechanical tests includes axial and ring tensile specimens. The shape and geometry of the specimens were optimized based on the FEM modelling and testing with non-irradiated materials in order to produce reliable relevant data. The objectives of geometry optimization are to produce as many samples as possible taking into account limited volume of irradiated alloy but still produce reliable results comparable to data produced elsewhere. The temperature for mechanical tests ranges from room temperature up to 400°C with the main focus on operational and storage temperatures between 300-350°C. Standard tensile tests and creep tests, were defined, for example, in [13] and [15]. The deformation rates and stress amplitudes are defined based on FEM modelling and expected magnitudes during normal operation, wet storage, drying of storage casks and dry spent fuel storage. The mechanical tests are supplemented with fractographic analysis of the damage surfaces with quantification of brittle/ductile fractures. Additionally, nanoindentation tests were defined in the same temperature range with a goal of measuring dose- and temperature-dependent properties of a virgin, corroded and hydrided Zr-based alloys [11]. Based on the evaluation of hydrogen pickup and precipitation, the models can account for different hydrogen-related failure mechanisms.

2.1.2 Material Characterization and Microstructure Analysis. The fundamental mechanical tests summarized in the previous section are accompanied by several characterization methods and techniques. The main microstructural tool to be used is a High-resolution Transmission Electron Microscopy (HRTEM) including detailed microstructure analysis of the deformed (after mechanical tests) and as-irradiated samples. Scanning electron microscopy (SEM) is also used to analyse the TEM foils mainly with EDX and EBSD detectors [12], [15]. The standard metallographic analysis is done according to the established ISO and ASTM standards and includes morphological analysis of the oxide layers, quantification of precipitates and hydrides and determination of the present phases [16] – [18]. Microhardness measurements are done on the metallographic samples in both longitudinal and transverse directions.

2.2. Modelling and simulations. FEM-based models with material properties and models currently used for licensing purposes have been utilized to optimize the geometry of samples for mechanical testing. Sample geometries originating in the French program PROMETRA and Hungarian institute MTA EK were used as a reference and further optimized [13], [19], [20]. A finite element model of axial tensile test with results of Von-Mises is shown in Figure 3. Based on the experimental testing, change of materials’ microstructural and bulk properties are evaluated and new correlations derived. They will include dependence on temperature as well as neutron dose. They are then implemented into codes used for nuclear fuel licensing such as ABAQUS, Frapcon and Fraptran [5], [6], [21]. The models account also for an anisotropic behaviour of Zr-based alloys thanks to the testing in the two main directions. The models can be later implemented also to integral and system codes and more precise behaviour of the fuel system can be then simulated.

2.3 TESTING OF THE REFERENCE NON-IRRADIATED MATERIAL
The testing started with non-irradiated (reference) samples serving as a reference for future comparison between irradiated cladding samples and non-irradiated. The methods used were developed based on the testing with reference material and the testing parameters were optimized in order to focus on normal operation and spent fuel storage conditions. As-received Zr1Nb material in its initial state, pre-oxidized as well as hydrided samples were included to study reference material in various expected states. The
geometry of samples for mechanical testing was optimized and the final sample shapes of axial and ring samples are shown in Figures 3 and 4. The microstructure has been studied using TEM, SEM and LOM. The EBSD map of the as-received material from the SEM of the 100 x 100 µm area is shown in Figure 5. The step size is 0.1 µm and the resolution 10,000 x10,000 pixels.

![Figure 3](image1.png)  
**Figure 3.** Calculation of Von-Mises stresses during the tensile test with the axial tensile sample using the FEM model

![Figure 4](image2.png)  
**Figure 4.** Ring sample used for tensile testing and creep tests

![Figure 5](image3.png)  
**Figure 5.** EBSD map of the as-received non-deformed reference sample. Grain size distribution, misorientation and the levels of recrystallization were calculated using software tools from the maps. Strong preferential orientation of the α-Zr grains is obvious.

The project itself and its outcomes will support the implementation of new nuclear fuel designs and in particular their licensing. Dose and temperature dependent material/physical properties will be determined, the tests to be performed are discussed in the following subsection. Additionally, the data measured will be compared to the large database of results obtained during past decades in various research reactors including the LVR-15 reactor operated in the Czech Republic. In comparison with commercial power reactors, the research reactors might operate with slightly different parameters such as neutron spectra, temperature, thermal loading or water chemistry. Therefore, based on the comparison of the old data and newly produced results, the methodologies used in research reactors can be validated or modified.

These activities are included in the In-pile Creep Studies of ATF Claddings (INCA) project which is a new joint experimental program included in the Framework for Irradiation Experiments (FIDES) coordinated by the OECD/NEA [22]. All of the relevant data produced during the project are summarized in an extensive TIRCLAD database. The structure and content of the TIRCLAD database are shown in Figure 6. The database is generally based on the IFPE format defined and extensively used by OECD/NEA for collecting of nuclear fuel performance-related data [23], [24].

3 PROJECT STATUS

First four batches of ampoules and material samples were separated from the whole MCA in POMA equipment at Temelin NPP in April 2019 (first two batches), April 2020 (third batch) and April 2021 (fourth batch). All the material samples were extracted from the ampoules were transferred to the semi-hot lab for evaluation of their geometry and other material analyses specified above. The remaining tube-formed ampoules (claddings) were transferred to CVR hot cell for evaluation of their microstructure, mechanical and creep properties. The POMA equipment for ampoule separation is
presented in Figure 7, and some moments of ampoule separation are presented in Figure 8. A transport of separated ampoule in a transport cask is presented in Figure 9.

Figure 6. Overview of the structure of the TIRCLAD database to be produced according to the IFPE format requirements defined by OECD/NEA.

Figure 7. POMA equipment for ampoule separation in the transport container shaft

Figure 8. Separation of the first batch of irradiated material.

Figure 9. Transport of the cask with ampoules from Temelin NPP to UJV Rez.

4 DISCUSSION, CONCLUSION AND FUTURE PLANS

An extensive material testing project called TIRCLAD has been initiated in the Czech Republic in 2012. Material assemblies have been first designed to be compatible with VVER fuel design and fuel handling machine [8], [25]. They were then approved by the utility, licensed at SONS and manufactured. Six material assemblies were loaded into the Unit 1 of Temelin NPP. In April 2019 the first and second assemblies were successfully placed into the POMA device, cut and the irradiated material ampoules were transported to hot cells for PIE. Fourth batch of ampoules was already cut in the POMA device and transported to hot cells this April. The remaining two assemblies will follow in the next outages with maximal radiation damage corresponding to fuel cladding irradiated up to 80 GWd/tU. Meanwhile, methodologies for testing were developed on non-irradiated samples. The geometry of samples for mechanical testing was optimized by UJV Řež using detailed FEM models and new test rigs were designed and build in the hot cells in the Research Centre Rez. PIE of non-irradiated and six batches of irradiated samples are focused on material evaluation to support licensing and new designs related to spent fuel behaviour during the long term wet and dry storage. New reliable models will be derived and a database called TIRCLAD will be produced as one of the outcomes of the project as well.

Nowadays, first three batches of irradiated material after first three cycles of irradiation have been cut and transported to UJV Rez and Research centre Rez facilities for testing and evaluation. On-going extension of the project is under preparation. The MCA-ATF program is to be focused on Advanced Technology and Accident Tolerant Fuel claddings with irradiation starting in 2022 [26].

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