DELISA LTO

NEWSLETTER
03/2023

Dear nuclear energy enthusiasts,

You are just reading the third number of the DELISA-LTO project newsletter issued to professionals and the general public. This project deals with the safe lifetime extension of light water reactors (VVER) and the ageing of primary loop components with the aim to support the safe energy supply in Europe for the next decades.

This newsletter follows Newsletter no. 02/2023 and provides the basic information about the project and description of done or planned work/events during the last 6 months of the Project from June to December 2023. We believe that you will find interesting information here and we will meet for the next newsletter.

Team DELISA-LTO

Project HORIZON-EURATOM-2021-NRT-01-01

DELISA LTO

DEscription of the extended LIfetime and its influence on the Safety operation and construction materials performance – Long Term Operation with no compromises in the safety

Project period:
June 2022 – May 2026

Reporting period:
June 2023 – December 2023

Main Goals of the Project

The goals of the project is to increase operational safety and lifetime extension due to:

- Determination of the most critical components from the LTO point of view and description of the LTO effect on the material properties.
- Development of non-destructive techniques, simulation tools and their methodologies for early prediction of failures.
- The setting of recommendations for future NPPs operation and assessment of their lifetime extension.

Works done in the project (since June 2023)

Project Meetings

- **General Assembly meeting**, 5th – 6th June 2023, EK-CER Budapest, Hungary.
- **Project Management Board meeting**, 22nd August 2023, online.

Submitted Deliverables

- **WP2 - D2.7:** Cutting plans and experimental matrix for structural evaluation, November 2023.
- **WP2 - D2.9:** Cutting plans and experimental matrix for mechanical testing, November 2023.
- **WP3 - D3.2: :** Report on developing and testing models of RPV internals for swelling evaluation, September 2023.
- **WP5 – D5.1:** Questionnaire for survey of used guidelines, standards, codes and "best practices„ connection to the relevant SSCs, July 2023.
- **WP6 – D6.2:** Database of relevant publication to project DELISA-LTO published in the past, November 2023.

Funded by the European Union

Cutting plans and Experimental matrix for structural evaluation (D2.7) and mechanical testing (D2.9)

In the DELISA-LTO project, 10 different components of the primary circuit were identified for investigation (e.g. main circulation pipe and pump, steam generator, etc.). Some of these materials consist of several areas, such as the base metal (BM), the weld metal (WM), and the heat-affected zone (HAZ), so a total of 15 different materials are being investigated.

At the outset, the project initially intended to test only the archived materials from V-1 NPP in Jaslovske Bohunice. However, within the first year of the project, the need arose to investigate new critical components from other NPPs - specifically, the Paks NPP in Hungary and the WWER-1000 type reactors in Ukraine (Materials 6 to Material 10 in Tab. 1).

Within the scope of WP2 and WP4, an Experimental Matrix (Tab. 1) was formulated. This matrix outlined the planned experimental methods in conjunction with the specific materials/components. Partners supplemented this by specifying sample dimensions required for individual experimental techniques -an essential aspect for devising cutting schemes.

DELISA-LTO – WP2/WP4 Experimental matrix				A6 - Scanning Electron Microscopy					A7 - Optical Microscopy					A8 - TEM			A9 - Other methods							
				CVR	STUBA	EK-CER	VUJE	VTT	CVR	STUBA	EK-CER	VUJE	VTT	CVR	STUBA	VTT	STUBA			VUJE		CVR		
Material No.		Material (*description, provider, operational temperature*)	Material Type	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
1		Main circulation piping - NPP V1 - initial state	BM	■	■	-	■	■	■	■	-	■	■	■	■	■	■	■	■	■	■	■	■	■
				■	-	-	-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	■	■	■
2		Pressurizer surge line - NPP EBO3 - initial state	BM	-	■	-	■	■	-	■	-	■	■	-	■	■	■	■	■	■	■	■	■	■
				-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	■	■	■
3	3a	Main circulation piping + WM - NPP V - after 28 years in operation	BM	■	■	-	■	■	■	■	-	■	■	■	■	■	■	■	■	■	■	■	■	■
				■	-	-	-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	■	■	■
	3b		WM	■	■	-	■	■	■	■	-	■	■	■	■	■	■	■	■	■	■	-	-	-
				■	-	-	-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	-	-	-
	3c		HAZ	■	■	-	■	■	■	■	-	■	■	■	■	■	■	■	■	■	■	-	-	-
				■	-	-	-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	-	-	-
4	4a	Main circulation piping - triple T-junction + WM - NPP V1 - after 28 years in operation	BM	-	■	■	■	-	-	■	■	■	-	-	■	-	■	■	■	■	■	-	-	-
				-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	-	-	-	-	-	-
	4b		WM	-	■	■	■	-	-	■	■	■	-	-	■	-	■	■	■	■	■	-	-	-
				-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	-	-	-	-	-	-
	4c		HAZ	-	■	-	■	-	-	■	-	■	-	-	■	-	■	■	■	■	■	-	-	-
5		Pressurizer surge line - NPP V1 - after 28 years in operation	BM	-	■	-	■	■	-	■	-	■	■	-	■	■	■	■	■	■	■	■	■	■
				-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	■	■	■
6		VVER-1000 - Reactor flange fastening parts after 30 years in operation (IPP-C)	BM	-	■	-	-	-	-	■	-	-	-	-	■	-	■	■	■	-	-	■	■	-
				-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	■	■	-
7		VVER-1000 · Steam generator collector with HE tubes after 30 years in operation (IPP-C)	BM	-	■	-	-	-	-	■	-	-	-	-	■	-	■	■	■	■	■	-	-	-
8		VVER1000 - Main circulation piping + SS cladding - initial state	BM	-	■	■	-	-	-	■	■	-	-	-	■	-	■	■	■	■	■	■	■	-
				-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	-	-	-	■	■	-
9	9a	VVER-440 - Capsules of the thermal specimens used in NPP Paks - 24 years in operation	BM	-	-	■	-	-	-	-	■	-	-	-	■	-	■	■	■	-	-	-	-	-
				-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	-	-	-	-	-	-
	9b	VVER-440 Capsules of the thermal specimens used in NPP Paks - 32 years in operation	BM	-	-	■	-	-	-	-	■	-	-	-	■	-	■	■	■	-	-	-	-	-
				-	-	■	-	-	-	-	■	-	-	-	-	-	-	-	-	-	-	-	-	-
10		VVER-440 Main circulation pump - Guide wheel - 27 years in operation	BM	-	■	■	-	-	-	■	■	-	-	-	■	-	■	■	■	-	-	-	-	-

Tab. 1 The Experimental matrix.

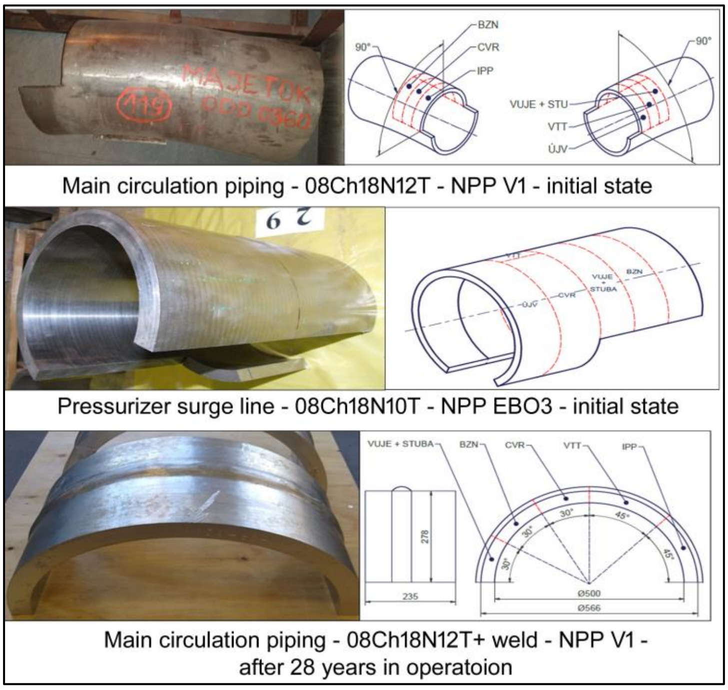
Funded by the European Union

DELISA LTO

The experimental matrix contains steels 08Ch18N10T or 08Ch18N12T, with welds 04Cr19Ni11Mo3 from V-1 NPP, 38KhN3MFA and 10GN2MFA steels with austenitic cladding ТУ 24-10-003-73 from VVER-1000 in Ukraine and steels 08Kh18N10T and 10H18N9TL from VVER-440 in Paks, Hungary (Chemical composition in Tab. 2). The Experimental matrix was developed considering the aim that initial and archive materials have to be artificially thermal aged to investigate the thermal ageing effects up to 60+ years of NPP lifetime. A mandatory criterion was set that at least two partners must contribute the same measurement technique to ensure robust comparative analysis and validation of the results.

After the finalization of the Experimental Matrix, precise cutting plans were formed. This involved creating technical drawings for each component and specifying the necessary cuts. Detailed plans for each partner were developed based on the Experimental Matrix. After several steps of the cutting schemes' improvement, the ultimate cutting plans were formed and then approved by all partners. Fig. 1 illustrates examples of these individual cutting plans, displaying both the actual material and accompanying technical drawings with partner markings on each piece of material.

Following the cutting plans, the materials will undergo cutting and samples will be distributed between the DELISA-LTO labs. The works will continue by artificial thermal ageing, after which the materials will undergo testing within WP4 to investigate the effects of thermal ageing in LTO.



Main circulation piping - 08Ch18N12T - NPP V1 - initial state

Pressurizer surge line - 08Ch18N10T - NPP EBO3 - initial state

Main circulation piping - 08Ch18N12T+ weld - NPP V1 - after 28 years in operatoion

Fig. 1 Example of the real material and their individual planned cuts.

Tab. 2 Chemical composition of the investigated samples within the project.

Steel	08Ch18N10T		08Ch18N12T	10GN2MFA	Cladding	Sv 04Ch19N11M3 (Weld)
	Forgings, sheets bars	Pipes				
C	max. 0,08	max. 0,08	max. 0,12	0,08-0,12	0,08	max. 0,10
Si	max. 0,80	max. 0,80	max. 0,80	0,17-0,37	1	max. 1,00
Mn	1,00-2,00	max. 1,50	max. 2,00	0,70-0,90	1,50-2,50	0,80-2,00
Cr	17,00-19,00	17,00-18,00	17,00-19,00	0,3	17,50-20,50	15,00-20,00
Ni	9,00-11,00	9,00-11,00	9,00-11,00	1,70-2,00		9,00-12,00
Ti	5x%C-0,60	5x%C- 0,60	5x%(C-0,02) - 0,60			
Co	max. 0,05	max. 0,05			0,05	
N	max. 0,05	max. 0,05				
S	max. 0,02	max. 0,02		0,02	0,02	max 0,02
P	max. 0,035	max. 0,035		0,02	0,03	max. 0,03
Mo				0,4-0,7		1,50-3,00
V				0,04		
Cu				0,3	0,3	
Nb					0,7-1,1	

Normative approaches, current practice and experience of RPV internals calculations (swelling evaluation) for LTO justification (D3.1)

As part of project in document D3.1, the partner countries presented their experiences with the evaluation of swelling and its calculation for the internals of reactor pressure vessels (RVI) of VVER reactors under design and Long-Term Operation (LTO). The main objective of the swelling assessment is to demonstrate that the gaps between the "core baffle-core barrel" and the "core baffle-fuel assembly" (Fig. 1) are within the available limits and allow safe operation until the next periodic safety review.

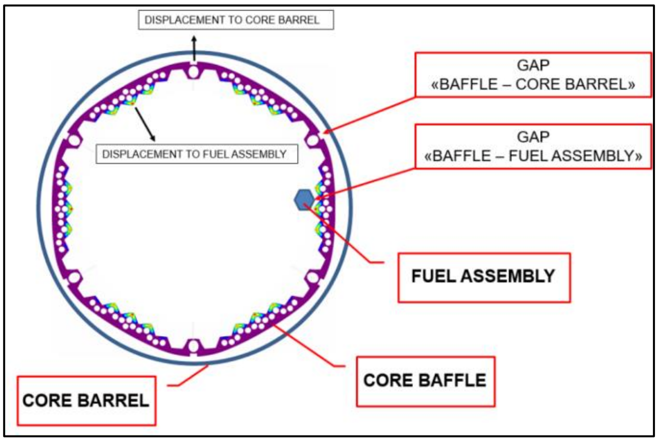


Fig. 1 The gaps between the "core baffle-core barrel" and the "core baffle-fuel assembly.

The swelling is not a problem in VVER-440 as the operating temperature is relatively low. However, the assessment of the swelling of the RVI in VVER-1000 needs to be performed as part of the lifetime extension and should be reassessed every 10 years. This shows the potential depletion of the gaps in the "baffle-core" and "baffle-fuel" area, which probably starts after 30 years of operation.

The main problem in the assessment of RVI swelling lies in the empirical mathematical model for the radiation-induced swelling/creep of 08Kh18N10T steel and the lack of failure criteria for the irradiated components.

Therefore, new calculations and the creation of a finite element model (FEM), including core baffle rings, core baffle fasteners, threaded rods and the core barrel with the faceted belt, is one of the most important steps in the evaluation of swelling. This model will be a basement for simulation in ANSYS software (IPP and SSTC NRS), in ABAQUS software (UJV) and in MSC Mark software (BZN). The general procedure for carrying out the numerical calculation to estimate the irradiation-induced creep and swelling is shown in the Fig. 2.

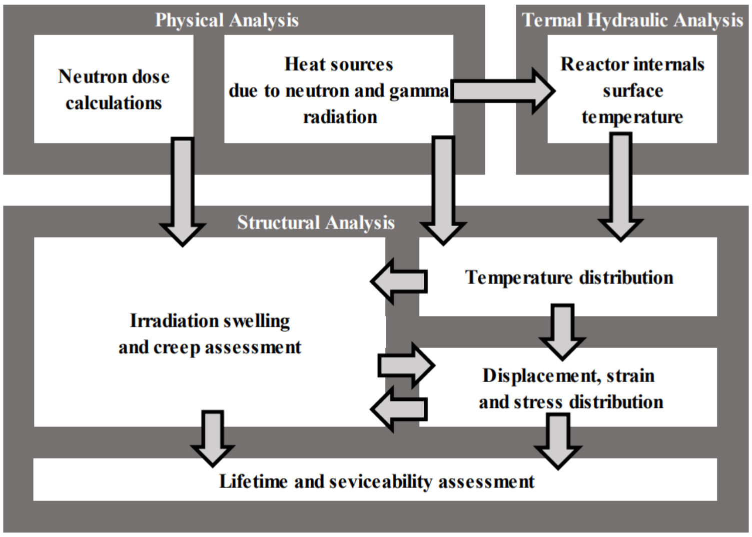


Fig. 2 The general procedure of numerical calculation.

The calculation and modelling of the RVI swelling are supplemented by experimental measurements of the actual geometry of the core baffle during the operation of the VVR-1000 with measuring tools in Ukraine and in the Czech Republic. The main objectives of regular measurements are to collect information to assess the initial state of the core baffle and the dynamics of changes in geometric dimensions, to determine and predict the swelling of the core baffle caused by the development of physical processes in the material under the influence of irradiation with high-energy neutrons and gamma rays.

Developing and testing models of RPV internals for swelling evaluation (D3.2)

The task 3.2 within WP3 involved conducting benchmark analyses on modelling and calculation methodologies utilized by project partners (IPP, SSTC NRS, ÚJV, and BZN) to evaluate radiation-induced swelling, serving as the initial step in predicting the lifetime of systems, structures and components (SCCs).

The participating partners commenced this task with a simplified analysis using a 2D CAD model (Fig. 1) representing the horizontal section of the VVER-1000 active zone baffle. This step was crucial for standardizing the methods used and ensuring comparability of results. More intricate and precise 3D models will be developed in Task 3.3.

The model portraying the internal components of the RPV was constructed and analysed using various finite element codes: ANSYS Plane183 by IPP (Fig. 2), ANSYS-3D 20 node solid element by SSTC NRS, Abaqus 2023 by ÚJV, and MSC MARC2023.1 by BZN.

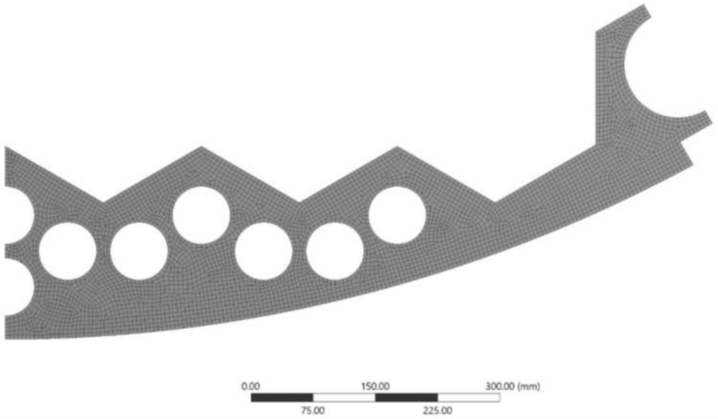


Fig. 2 IPP model – finite element model mesh.

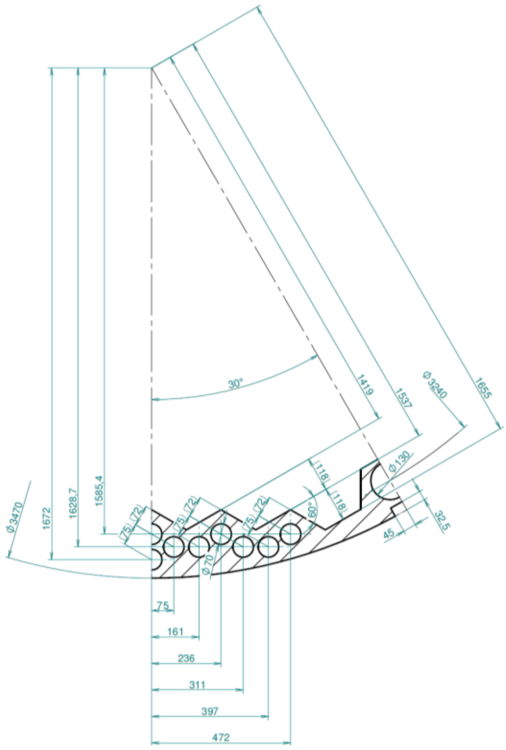


Fig. 1 VVER-1000 core baffle cross-section to be used within Task 3.2.

The initial benchmark analysis covered:

- Evaluation of the internal heat source in the model (Fig. 3),
- Analysis of neutron dose distribution in the model (Fig. 4),
- Analysis of temperature distribution in the model (Fig. 5),
- Stress assessment (axial, failure, hydrostatic) in the model,
- Assessment of plastic deformation in the model,
- Examination of swelling deformation in the model (Fig. 6) .

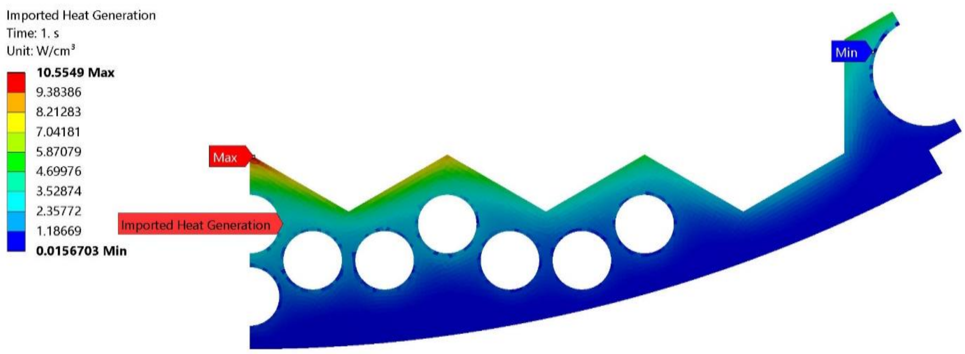


Fig. 3 IPP model of internal heat source distribution.

Funded by the European Union

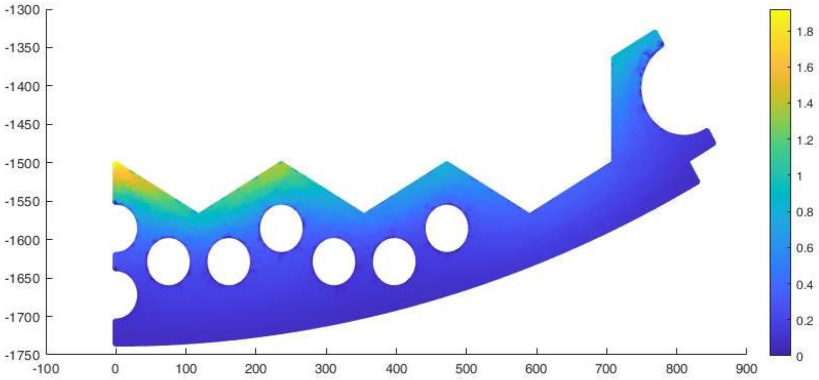


Fig. 4 IPP model of the dpa distribution.

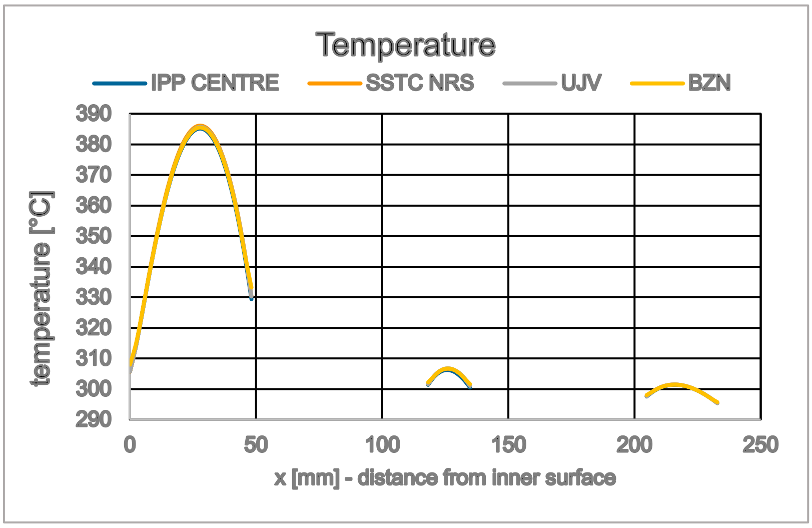


Fig. 5 Comparison of temperature profiles in radial cross-section in symmetry plane across 2 channels.

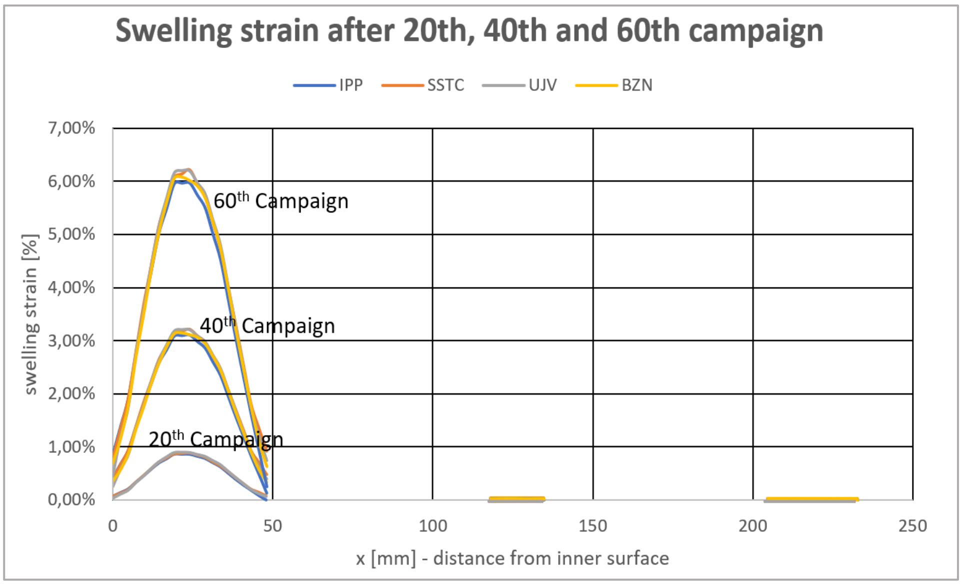


Fig. 6 Comparison of swelling strain profiles in radial cross-section in symmetry plane across 2 channels after 20th, 40th and 60th campaigns.

All participants contributed comprehensive and technically reliable results for the 2D simplified benchmark, achieving substantial agreement among the outcomes. Based on the analysed data, several conclusions can be drawn:

- The maximum temperature of 385 °C is reached at the midpoint of the first ligament.
- The maximum volumetric swelling strain, approximately 6% after the 60th campaign, occurs at the midpoint of the first ligament.
- Creep strains exhibit an accelerating trend during NPP operation. The maximum equivalent creep strain, approximately 2.5% after the 60th campaign, is observed at the midpoint of the first ligament. This is attributed to the peak temperature and swelling strain at this location, as well as the maximum dpa.

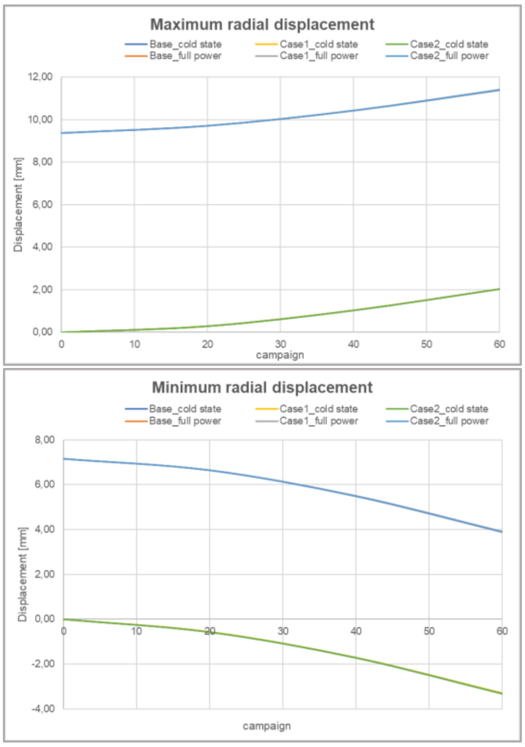


Fig. 7 Evolution of maximum and minimum radial displacement during the NPP operation in [mm].

Database of relevant publications to project DELISA-LTO (D6.2)

[Deliverable 6.2](https://delisa-lto.eu) describes a compiled database of relevant publications for the Delisa LTO project. This database is intended to streamline and expedite the work with publications and enables the collection of knowledge on material ageing and long-term operation of VVER reactors. It can also serve as a valuable resource for future publications and citations.

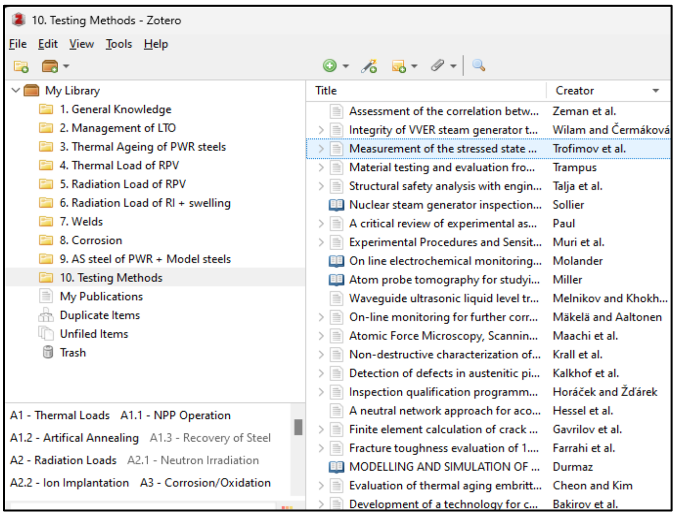


Fig. 5 The Database in Zotero – reference manager software.

The database was compiled from publications published between October 1983 and September 2023. These publications have been included in the most prestigious journals to ensure the highest quality of results and knowledge. The search for these publications was carried out via online database systems such as "Scopus", "Science Direct" and "Web of Science". The database is a publication list and, due to the journals' copyright protection, does not contain the publications themselves, but there are links to the official publications.

The selected publications are primarily concerned with the field of materials research, focussing on austenitic and ferritic steels used in the primary circuits of VVER reactors. The publications mainly deal with the reaction of the analysed materials to thermal stress as observed in the experimental part of the project.

The search was carried out using specific keyword combinations, including "VVER," "long-term operation", "ageing management", "austenitic", "thermal stress," "thermal ageing," "thermal stress", "08Ch18N10T"08Ch18N10T", "08Ch18N12T", "10GN2MPA", "18Cr10Ni", "FCC thermal", "FCC hardening", "FCC swelling," "FCC toughness", "AISI321 thermal", "AISI304 thermal", "AISI316 thermal" and "CrNi steel", etc.

The database currently contains 210 articles, which have been categorised into several main categories:

1. General Knowledge,
2. Management of Long-term operation (LTO),
3. Thermal Ageing of Pressurized Water Reactors (PWR) steels,
4. Thermal Load of Reactor Pressure Vessel (RPV),
5. Radiation Load of Reactor Pressure Vessel (RPV),
6. Radiation Load of Reactor Internals (RI) + Swelling,
7. Welds,
8. Corrosion,
9. Austenitic steel (AS) of PWR + Model steels,
10. Testing Methods.

The database is updated at regular intervals to ensure that it is up to date by including new or missing relevant publications. The database can be downloaded from the project website https://delisa-lto.eu/database-of-relevant-publications/ as a file for reference manager software (e.g. Zotero, Mendeley) or is also available online.

Fig. 6 The Database online on the project website.

State of the art on Thermal Ageing Evaluation (D3.4)

Within Task 3.4 a survey regarding thermal ageing as a degradation mechanism and the experiences with its evaluation in NPPs was prepared. The partners' responses are summarized in document D3.4 published on the project website.

In practice, national standards, normative documents and safety codes are used by partners' countries for thermal ageing analysis in addition to existing international standards. That is except Finland, which has no national regulations in this respect.

The basic methodology concerning thermal ageing analysis includes non-destructive and destructive methods for testing mechanical properties, calculation assessment of changes in strength and ductility characteristics, calculations for resistance to brittle fracture in Ukraine and Hungary, and mechanical testing and microstructural characterization in Finland and the Czech Republic.

In the Czech Republic and Finland, thermal ageing effects are not observed over the monitoring. Based on operational experience for VVER-440 thermal ageing is not relevant. In Ukraine and Hungary in case of some SSCs thermal ageing is technically observed, therefore this effect should be monitored during the operation.

For VVER-1000 where could be cast austenitic stainless steel used for Main Circulation Pump (MCP) body it is advisable to measure concentration of delta ferrite, and if it is higher than limit to periodically measure hardness of MCP body to detect any changes. Based on operational experience in Ukraine thermal ageing is mainly observed only due to the data scatter and uncertainties considered in conservative way (not best estimate). However, thermal ageing could take place for LTO for austenitic Stainless Steels, Alloy Steels, Carbon Steels.

Based on operational experience in Hungary for VVER-440 for 22K carbon steels operating continuously at 250 °C and above, and for 08Ch18N10TL austenitic castings and welds (austenitic, transition, and ferritic) the possibility of thermal embrittlement should be investigated and monitored, and additional measures determined as necessary (e.g. delta ferrite content and hardness measurements).

Next Deliverables in M24

- **D2.3** Report on operational experience and non-destructive techniques
- **D2.4** Report on indications of selected V1 NPP steam generator tubes
- **D2.5** Methodology of the structural and substructural changes for evaluation of thermal ageing processes
- **D2.8** Methodology of mechanical testing of evaluation of thermal ageing processes

Next meetings

- **General assembly meeting**, 7th December 2023, online.
- **Technical meeting of WP 2 & 4**, 15th – 16th February 2024, STUBA, Bratislava.
- **Annular project meeting**, 25th -27th June 2024, VTT, Helsinki.

DELISA-LTO

For more information:

Delisa Lto

Contact us on: info@delisa-lto.eu
Website: www.delisa-lto.eu

Funded by the European Union

DELISA LTO

Feel free to forward this Newsletter to friends that can be interested to be informed about DELISA-LTO Project or log-term operation of NPPs.