INVESTIGATION OF THE PROCESSES IN THE SPENT FUEL DURING INTERIM STORAGE IN IGNALINA NPP

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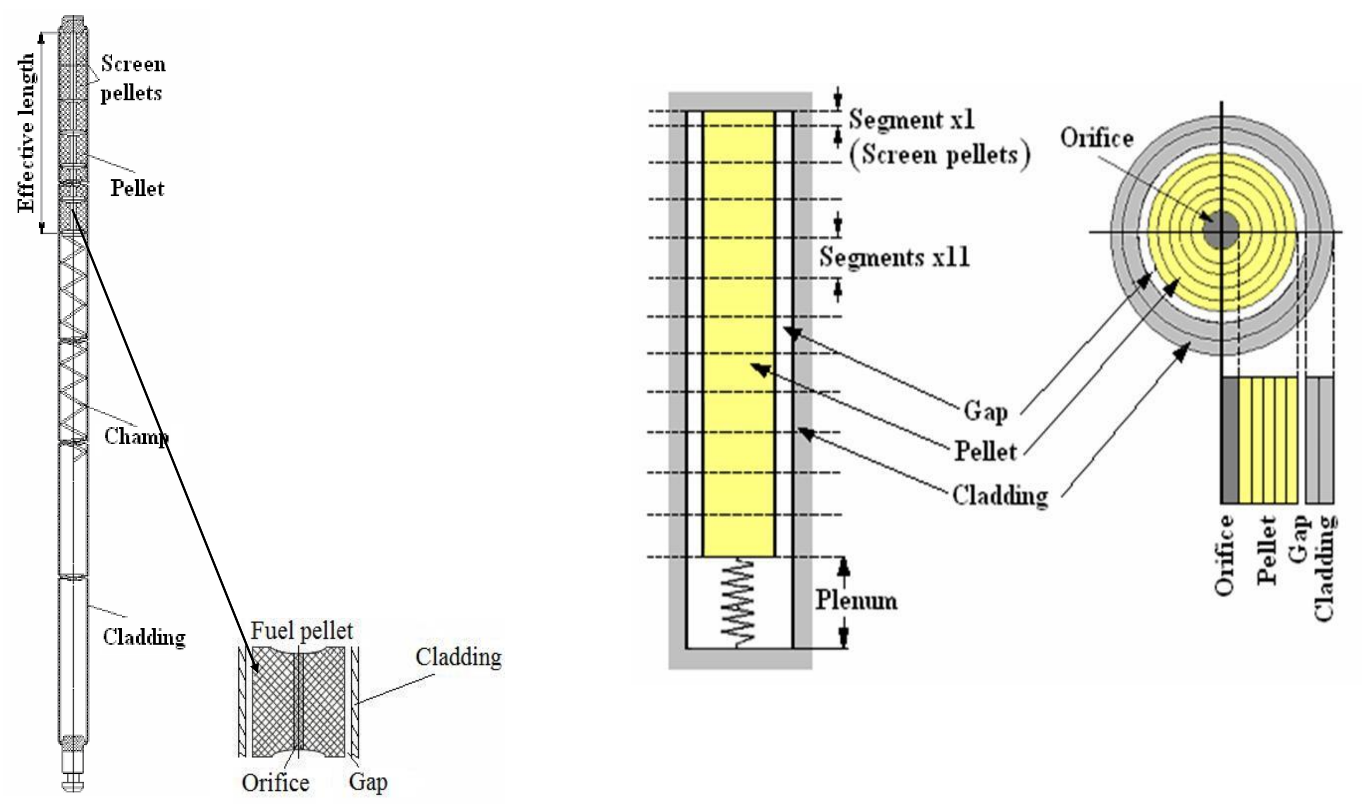
1. Introduction:

Ignalina Nuclear Power Plant in Lithuania operated two RBMK-1500 reactors, shut down in 2004 and 2009. Around 22,000 spent nuclear fuel (SNF) assemblies remain in dry storage, where they will stay for at least 50 years before transfer to a final repository, with the location yet to be determined. Each RBMK assembly consists of two 3.5 m fuel bundles. After cooling in spent fuel pools, assemblies are processed in a hot cell, separated, and placed into casks for storage or transport. Understanding SNF composition and behavior is crucial for ensuring nuclear safety, radiation shielding, and thermal performance during storage.

2. Description of research problem

As part of the IAEA coordinated research project “Spent Fuel Characterization” (T13018), numerical models for the RBMK-1500 fuel assembly and fuel rod were developed using SCALE and TRANSURANUS codes. After validation and optimization, these models were applied to calculate spent fuel decay heat over 100 years post-discharge, relevant for interim storage. The need for this work is grounded in a set of uncommon features present in RBMK type fuel as well as in the need for the application of updated methodologies and tools:

- Irregular Core Design: The RBMK's irregular lattice and graphite moderator complicate the generation of accurate multigroup cross-sections, as existing tools are optimized for regular LWR configurations.
- On-line Refuelling: The ability to refuel while operating alters the location of fuel assemblies based on burnup, necessitating detailed irradiation history for precise modeling.
- Axial Heterogeneity: Variations in water density, influenced by boiling and burnup, affect moderation efficiency and power profiles, requiring specific data for accurate SNF composition modeling.
- Lack of Experimental Data: No experimental isotopic data for RBMK-1500 FA exists, only limited information for RBMK-1000. Previous modelling efforts relied on outdated data libraries that may not account for critical isotopes, such as erbium.
- Updated Nuclear Data: Current models should incorporate the latest nuclear data libraries, which provide more comprehensive isotopic information, including all erbium isotopes, to improve simulation accuracy and reduce uncertainty.



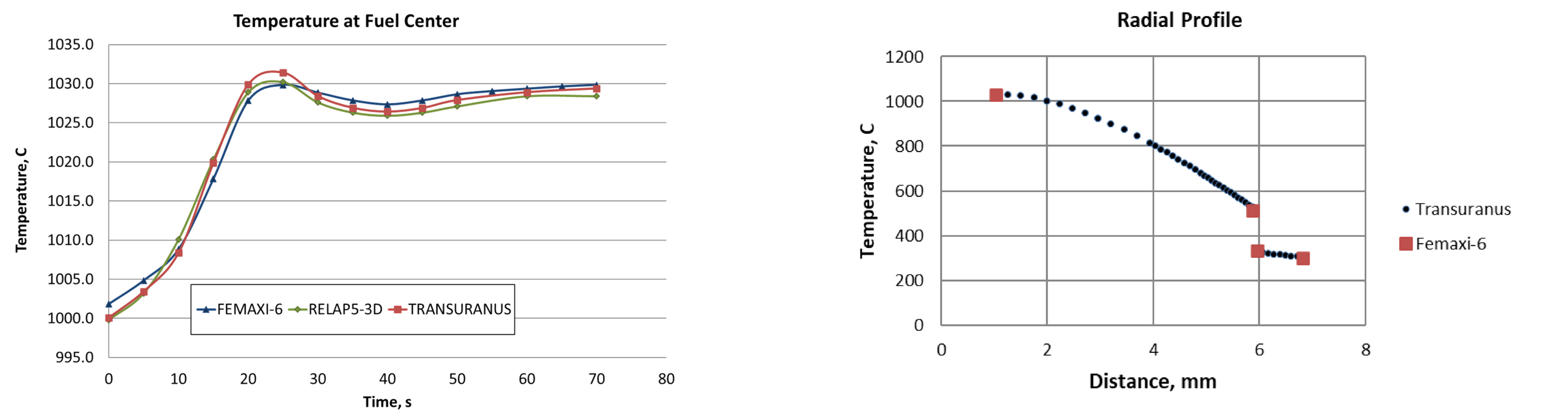
Length of fuel rod, mm	3640	Outside fuel pellet diameter, mm	11.48
The active length of fuel rod, mm	3410	Pellet central orifice diameter, mm	2
Height of screening pellets, mm	30	Fuel enrichment in U^{235}, %	2.8
Length of plenum, mm	170	Edge pellet enrichment, %	0.7
The outside diameter of the fuel rod, mm	13.63	Fuel pellet density, g/cm^3	10.55
Inside cladding diameter, mm	11.90	Mass of fuel within fuel rod, g	3500
The initial pressure of gases in the fresh fuel rod at cold conditions, MPa			0.5

3. Methodology

This particular presentation is focused on the fuel performance calculations which were performed with TRANUSURANUS (v1m6j21) code. First RBMK-1000 fuel rod model was created and after validation it was modified to represent RBMK-1500 type fuel. After this, 4 base irradiation scenarios were considered for differently U-235 enriched fuels at 2%, 2.4%, 2.6% and 2.8%. With each enrichment level, the duration of the irradiation is extended by approximately 200 days and results in 3-4 GWd/MTU additional burnup (20 GWd/MTU and 882 days of operation). Each scenario also consists of 3 stages of fuel management: base irradiation, 5 years at the spent fuel pool, and 5 years at the dry storage (e.g., spent fuel cask). For the spent fuel pool calculations, it is assumed that each fuel rod will be placed in a 45 °C temperature water at 0.22 MPa for 5 years. For the dry storage it is assumed that the rod is being placed in atmospheric pressure with a prescribed outer cladding temperature which in 5 year period gradually decrease from 400 to 250 °C. These assumptions correspond to the probable scenarios and temperature ranges. However, they are not based on actual measurements.

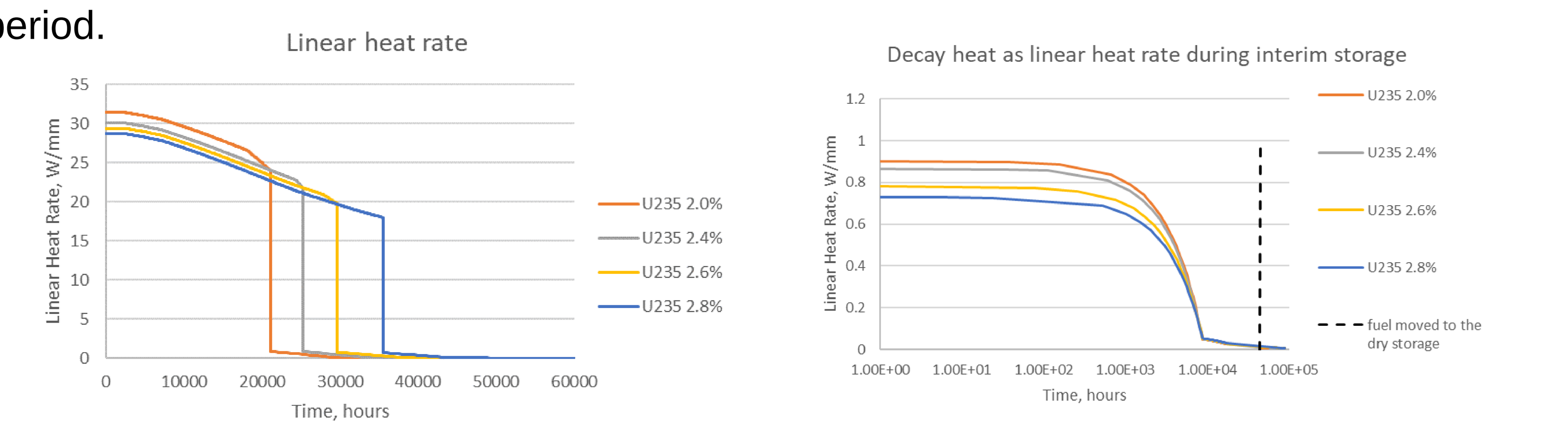
4. Results

TRANSURANUS validation for RBMK-1000 design was conducted based on ramp and burn-up tests from Smolensk NPP experiments. Benchmark results were compared to prior FEMAXI-6 calculations by the Lithuanian Energy Institute and proprietary code results from the experiment designers. The ramp test included a 16-hour steady-state operation followed by a 65-second linear power variation. Power and neutron flux varied across axial fuel rod segments. Coolant temperature and pressure remained constant at 299.85°C and 7.4 MPa, with 663 kg of coolant circulated per hour. Reactor start-up occurred at 20°C with plenum pressure of 0.587 MPa.



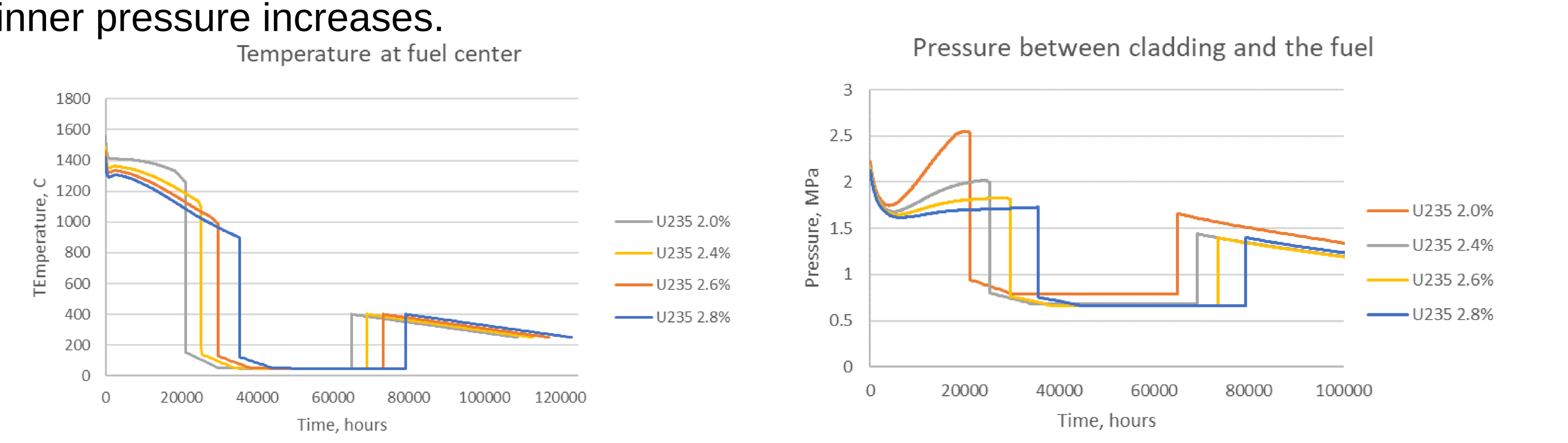
RBMK-1000 validation

Fuel rod irradiation scenario was used in accordance to the prior SCALE code calculations, which were also used to obtain decay heat evolution for the interim storage period.

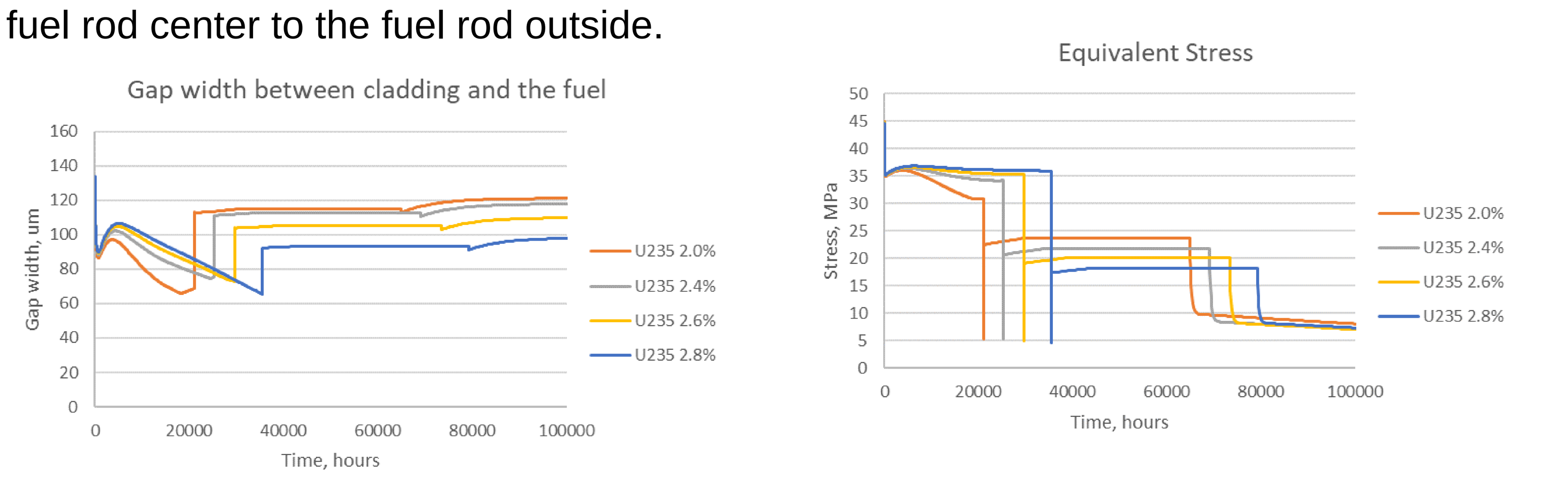


Linear heat rate for the operation and shutdown (represents decay heat)

Pellet does not interact with the cladding during irradiation. It relaxes after being put in the spent fuel pool and later expands even more while in the dry storage cask. The reason for this might be the change in pressure from 0.22 MPa to 0.1 MPa. Likewise, inner pressure increases.



From the mechanical standpoint, Equivalent stress in cladding is relatively low both during the operation scenario and storage stages. It drops by around half when the fuel rod is put into the spent fuel pool and drops even further when it is placed in the dry storage. After irradiation and wet storage, the stress vector changes direction from the fuel rod center to the fuel rod outside.



As for strains in the cladding, cumulative creep strain is the highest at 2.8%, as it mostly corresponds to the irradiation time. Plastic strain is not noticeable as there was no contact with the fuel pellet and cladding or other phenomena that could have caused significant deformation.

5. Conclusions

TRANSURANUS work mostly covers the effects of the different operation scenarios with specific irradiation histories. Some extra configurations were also investigated with different decay heat calculation configurations available in SCALE such as structural differences in FA, graphite, coolant materials used in TRITON sequence. A ~10% increase in decay heat led to only a 0.1% change in most of the thermomechanical parameters analyzed, dropping to 0.003% or less for more enriched fuels — differences too minor for detailed analysis.

Acknowledgments

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