Framatome VVER-440 Fuel Product Line Development

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ABSTRACT

To increase diversification and security of fuel supply for European utilities operating VVER, Framatome is developing fuel products for VVER-440 reactors ensuring 100% European sovereignty.

The development program was launched in 2024, aiming at readiness for regular fuel deliveries by the early 2030s. Since June 2024 this effort is backed by the EU-funded SAVE project and since April 2026 by a development agreement between Framatome, ČEZ, Fortum, MVM Paks NPP, and Slovenské elektrárne.

The contribution will outline this program and its main activities:

- Consolidation of requirements across all target utilities and reactors,
- Design and licensing of the VERA-440 fixed and follower fuel assemblies,
- Design and licensing of a VVER-440 container,
- Upgrade and qualification of manufacturing routes.

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Next Generation Fuel Development for VVER Type Reactors

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ABSTRACT

Westinghouse has established extensive operating experience with VVER fuel designs through the successful irradiation performance of the upgraded Robust Westinghouse Fuel Assembly designs (so-called RWFA and RWFA-T designs) in mixed and homogeneous VVER-1000 cores in Ukraine, at Temelín NPP in the Czech Republic, and at Kozloduy NPP in Bulgaria. Irradiation performance and operation experience of these designs has been highlighted by all Customers with a positive overall feedback. In parallel, the upgraded Westinghouse fuel designs NOVA E-5/E-6 for VVER-440 reactors have demonstrated excellent performance in mixed cores of Rivne and Loviisa NPP units.

As a part of the Westinghouse's long-term strategy for the VVER market, further product enhancements are being pursued based on operating experience, customer feedback, and identified opportunities for improved fuel economy, reliability, and

manufacturability. The ongoing Next Generation VVER Fuel Development activities for both VVER-1000 and VVER-440 fuel designs are presented herein.

For VVER-1000, Westinghouse initiated a research and development project in 2020 to develop a next-generation structural grid for the Westinghouse VVER-1000 fuel assembly design family. The development targets improved neutronic, mechanical, and thermal-hydraulic performance compared with existing grid designs. The design combines the neutronic benefits of zirconium-alloy spacer grids with a novel individual cell-based grid architecture developed to achieve lateral strength characteristics comparable to, or better than, those of the current Nickel Alloy 718 design while reducing hydraulic resistance. Preliminary analyses and testing have demonstrated the potential for both improved mechanical performance and reduced overall fuel assembly pressure drop.

For VVER-440, the NOVA E-7 development aims to improve neutron economy through an increased water-to-uranium ratio. Key design changes include an increased fuel rod pitch and a thinner shroud, providing improved moderation.

Additional features under evaluation include an extended pellet stack and optimized components to enhance fuel economy, reduce material usage, and improve manufacturability while maintaining fuel reliability and inspection capability.

The current development status, including results from analyses and scoping tests, is summarized together with an outlook on the remaining development activities.

Framatome VVER-1000 Fuel Assembly Development Status

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ABSTRACT

Recent geopolitical and fuel supply security considerations have reinforced the need for alternative fuel solutions for VVER reactors. To address this challenge, Framatome has been developing since 2019 a fully European sovereign VVER-1000 fuel assembly design, supported by dedicated design methods, qualification programs, European manufacturing capabilities, and transportation solutions. Lead Fuel Assembly (LFA) deployment is targeted for 2029, with reload implementation planned in the early 2030s.

The new fuel assembly has been developed to meet strict requirements for fuel reliability, mechanical robustness, transportability, and compatibility with VVER operating conditions. It combines proven Framatome technologies, including M5TM cladding and Q12 zirconium alloy structural components, with specific design such as an open skeleton architecture, advanced

spacer grids cell design, and enhanced debris protection features. Its qualification relies on an extensive verification and validation program combining advanced simulation methods and experimental testing. Mechanical, hydraulic, and thermal-hydraulic investigations have been performed to assess structural behaviour, flow performance, and critical heat flux margins. With more than 95% of the out of pile qualification program completed or underway, the results obtained to date confirm the maturity and readiness of the design for future lead fuel deployment in VVER-1000 reactors.

This paper presents the design basis of the fuel assembly, summarizes the ongoing qualification campaign, and discusses the development status of the associated licensing methods and transportation package. The results obtained to date demonstrate strong mechanical performance, effective hydraulic behavior, and good agreement between numerical predictions and experimental measurements, supporting the readiness of the design for future lead fuel deployment in VVER-1000 reactors.

Keywords: VVER-1000, fuel assembly, fuel qualification, energy sovereignty, CHF, hydraulic testing, mechanical testing, accident tolerant fuel, licensing, Framatome

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ATLAS™ Fuel Design and Licensing in the European Pressurized Reactor Olkiluoto Unit 3

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ABSTRACT

The loading of Westinghouse ATLAS™ lead test assemblies (LTAs) in the European

Pressurized Reactor (EPR), Olkiluoto Unit 3 (OL3), and the start of mixed core operation, marks the completion of the qualification and LTA licensing process of the ATLAS fuel design in Finland. The ATLAS LTA licensing program has demonstrated that the ATLAS fuel design can be loaded and operated safely in the OL3 core at a thermal power of 4300 MW. All fuel related conditions have been analytically or experimentally substantiated, which include normal conditions, Anticipated Operational Occurrences (AOO), and accident conditions. Normal conditions are comprised of operating conditions including transients and shipping and handling. Analytical models, methods and methodologies are supported and verified by tests and validations from the ATLAS development program. The success of the ATLAS LTA program was achieved by building the ATLAS fuel design on proven features from Westinghouse's existing and well-established fuel product portfolio and the excellent cooperation between all involved organizations, including Westinghouse,

TVO and the Finnish nuclear regulatory body, STUK.

The qualification and licensing process to enable operation of the ATLAS fuel design in OL3 started in 2022 with technical meetings between TVO and Westinghouse aimed at elaborating the technical and financial justification for the licensing and loading of the Westinghouse ATLAS fuel design in OL3. After completion of the technical assessments between TVO and Westinghouse, it was decided to qualify the ATLAS fuel design for operation of fuel reloads in OL3, and a joint LTA supply and licensing program was launched by TVO and Westinghouse.

The OL3 ATLAS fuel design licensing program was carried out as a split scope in close cooperation between TVO, its partners, and Westinghouse. The licensing scope addressed by TVO, and its partners, was to handle AOOs, accidents, dynamic stability, decay heat, criticality safety and radiological consequences. Westinghouse was responsible for verifying the mechanical behavior of the ATLAS fuel design in

the OL3 core with support from TVO and its partners regarding the mechanical impact from a seismic and LOCA event. Westinghouse responsibilities also included benchmark of the nuclear and core design, thermal mechanical fuel rod design, thermal-hydraulic compatibility and subchannel Critical Heat Flux analyses.

The project has positioned Westinghouse as a viable fuel supplier to the EPR and contributed to increasing the security of supply of nuclear fuel in Europe, which is increasingly important in the current geopolitical situation.

GAIA Fuel Assembly: Proven Robustness & Performance for 17x17 Reactor fleet

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ABSTRACT

The GAIA™ fuel assembly was developed by Framatome as a standardized fuel platform for 17×17 Pressurized Water Reactors (PWRs), addressing increasing utility expectations regarding fuel reliability, operational flexibility, extended fuel cycles, and higher burnup capability.

The design objective was to consolidate several design principles and learnings from legacy fuel assembly families into a single platform while retaining proven technologies and integrating targeted improvements in fuel assembly performance, debris protection, manufacturability, and operational robustness. Key design features include an enhanced structural architecture, low-creep structural material, advanced spacer grid designs, an optimized debris-filtering bottom nozzle, and the implementation of high performance M5TM fuel rod cladding. GAIA can be delivered with PROtect, our EATF solution.

Framatome will present the rationale behind the main design evolutions and summarizes the qualification results obtained during the development program. Operational feedback from the reactor deployments is also considered. Results demonstrate improved structural robustness, effective debris filtration performance, excellent dimensional stability, and reliable in-reactor behavior, confirming the suitability of the GAIA platform for current and future PWR operating strategies, in particular, in the frame of Advanced Fuel Management (AFM) programs.

The successful industrial deployment of GAIA illustrates the benefits of a standardized fuel product capable of supporting a broad range of reactor requirements while maintaining high levels of safety, reliability, and fuel performance.

Keywords: PWR fuel, fuel assembly, GAIA, fuel qualification, debris filtration, grid-to-rod fretting, fuel reliability, Framatome.

Performance of Fuel Technology on AP1000™ Reactors

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Westinghouse Electric France

ABSTRACT

Westinghouse continues to deliver proven and high-performing AP1000®

Pressurized Water Reactor nuclear fuel to the expanding global fleet of AP1000 units.

This presentation will highlight the key fuel and core design features that have

consistently met performance expectations over nearly a decade of operational

excellence, earning strong customer satisfaction. It will also showcase advanced

core operation technologies—including BEACON™ Core Monitoring Software,

MSHIM™ Operation and Control Strategy, and the Advanced First Core design—

that have enabled reliable and efficient reactor performance. The session aims to

foster discussion around how these technologies address current challenges and

unlock opportunities in new nuclear plant development.

VERA-440: Verification and Validation Strategy and its Application to Selected Tests and Benchmark Calculation

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ABSTRACT

This presentation summarizes the Verification and Validation (V&V) plan for the mechanical and thermal-hydraulic assessments of VERA-440 Fuel Assembly design. Emphasis is placed on the application of the V&V plan to two mechanical assembly tests, the endurance test and the follower fuel assembly (FA) drop test, as well as to a thermal-hydraulic subchannel benchmark, demonstrating model accuracy, reliability, and predictive capability.

The endurance test is conducted in cooperation with CEA in the HERMES-P loop, simulating thermal-hydraulic conditions of VVER-440, including temperature, pressure, and flow rate, over extended operating periods. The test demonstrates the wear-resistance under simulated operating conditions. In addition, follower FA drop tests are performed at ŠKODA JS VVER-440 test facility. This campaign is intended to demonstrate follower FA performance under realistic boundary conditions.

The presented benchmark study assesses the thermal-hydraulic subchannel codes COBRA-FLX and ALTHAMC12 for VVER-440 fuel assembly applications using 10 publicly available critical heat flux (CHF) data sets. Based on both code-to-data and code-to-code comparisons, the results show that the two codes provide generally comparable predictions of CHF . In comparison with experimental data, both codes achieved better agreement when the PG-S correlation was applied, whereas larger deviations were observed with the OKB3 correlation. Sensitivity analyses indicate that the spacer grid pressure loss coefficient has only a minor effect, while the thermal diffusion coefficient exerts a more noticeable influence, particularly in calculations using OKB3. Overall, the study confirms the applicability of both codes to VVER-440 thermal-hydraulic analyses and supports their use in future validation and safety assessment activities.

Overall, the presented activities represent an important milestone in the validation of the design, demonstrating its robustness and maturity. The achieved results provide valuable input and a solid basis for future qualification and safety assessment processes.

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Core-level analysis of ATF Fuel Rods Performance under LBLOCA conditions in a PWR using TRACE and TRANSURANUS

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ABSTRACT

This work presents the results of a collaborative research project conducted between the Spanish regulatory body (CSN) and the private company NFOQUE Advisory Solutions (NFQ). The study focuses on the application of a core-level Fuel Safety Evaluation Methodology (FSEM) to assess the potential benefits of implementing Accident Tolerant Fuels (ATFs), also referred to as Advanced Technology Fuels, compared to conventional fuel rod designs under Large-Break Loss-of-Coolant Accident (LBLOCA) conditions in a generic three-loop Pressurized Water Reactor (PWR).

The developed FSEM consists of three main stages. The first stage involves the characterization of pre-transient fuel rod conditions through irradiation period simulations. The second stage focuses on the calculation of thermal-hydraulic boundary conditions across the reactor core during the LBLOCA scenario. Finally, the third stage integrates the previous results to simulate fuel rod performance under transient conditions. The computational tools employed in this work include the fuel performance code TRANSURANUS and the system thermal-hydraulics code TRACE.

A best-estimate modeling approach is applied in the first and second stages, while the third stage incorporates a Best Estimate Plus Uncertainty (BEPU) methodology based on Monte Carlo sampling techniques. In addition, sensitivity analyses are performed to evaluate the correlation between uncertain input parameters and selected Figures of Merit (FoMs).

The application of the FSEM enables the determination and statistical characterization of key safety metrics, such as burst probability and the number of failed fuel rods, for various fuel rods concepts. The configurations analyzed include:

- UO_2 / Zr-4
- UO_2 / Cr-coated Zr-4
- Cr-doped UO_2 / Zr-4
- Cr-doped UO_2 / Cr-coated Zr-4
- UO_2 / FeCrAl
- U_3Si_2 / FeCrAl

For cases involving FeCrAl cladding, an additional configuration is examined in which the cladding thickness is reduced from 570 µm to 385 µm to compensate for the associated neutron absorption penalty.

Overall, the results demonstrate that the incorporation of ATF materials provides significant benefits both during the irradiation phase and under transient conditions. In particular, the core-level analysis highlights the influence of evolving thermal-hydraulic boundary conditions and fuel loading patterns on burst probability and rod failure rates. A substantial reduction in these Figures of Merit is observed when ATF materials are used in both fuel and cladding, confirming their potential to enhance fuel safety performance under irradiation conditions and LBLOCA conditions.

Digital Image Processing from Nuclear Fuel Inspections

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ABSTRACT

Visual inspection of nuclear fuel assemblies is a critical activity for assessing fuel integrity, oxidation behaviour, and the safe operation of nuclear reactors. However, conventional inspections generate thousands of video frames while providing operators with only a limited view of the fuel, making engineering evaluation labor-intensive and difficult to standardize. This presentation introduces a digital inspection workflow that transforms raw inspection videos into standardized engineering context data through automated image processing and the creation of OIO (One Image Overview) representations. The resulting digital outputs enable quantitative measurements, historical comparison of individual fuel assemblies across operating campaigns, and the development of searchable lifetime fuel records. By reducing manual engineering review, the approach shortens outage evaluation activities and supports faster operational decision-making. Ultimately, digitalization shifts fuel inspection from isolated visual documentation to a strategic knowledge platform that improves plant efficiency, preserves engineering expertise, and establishes a foundation for future Digital Twin applications.

Future Prospects for the Transition to a Closed Fuel Cycle

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ABSTRACT

Although most nuclear power reactors are currently operated in an open fuel cycle, this is not the case for countries with advanced nuclear power programs and nuclear industries, which have either already closed their fuel cycles, are working toward closing them, or are at least planning to do so. The Czech Republic has not yet officially planned any transition to a closed fuel cycle, even though it is preparing for further development of its nuclear power sector.

However, in connection with the planned construction of new reactors meeting EUR criteria, the Czech Republic also has the opportunity to close its fuel cycle by recycling plutonium and uranium in the new reactors, with the aim of making better use of nuclear fuel and significantly reducing the amount of spent fuel destined for a deep geological repository.

A key prerequisite for this future transition to a closed fuel cycle is that newly constructed reactors will meet EUR criteria for the use of mixed uranium-plutonium (MOX) fuel.

This paper analyzes the technical, legislative, and economic possibilities and conditions for closing the fuel cycle through the use of reprocessing and MOX and RepU fuel fabrication services within the European Union.

Diversified and Sovereign Fuel Cycles: Securing Reprocessing Pathways for Fleet Longevity

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ABSTRACT

As nuclear fleets expand globally, optimizing the back-end fuel cycle is critical for long-term operational sustainability. Globally, nuclear operators and regulatory bodies face a complex problem governed by finite wet storage capacities and the structural timelines required for the implementation of Deep Geological Repositories (DGR). This technical equilibrium is further compounded by current geopolitical shifts within the nuclear fuel market, which necessitate a strategic restructuring of industrial supply chains and an accelerated diversification of back-end risks to guarantee national energy sovereignty.

While dry storage serves as a reliable operational buffer, long-term technical assessments highlight constraints, including the thermal evolution of fuel assemblies, structural degradation mechanisms of containment materials over multi-decade timelines, and the requirement for continuous, capital-intensive monitoring. Deep Geological Repositories (DGR) disposal is widely recognized and established as the reference long-term pathway for Intermediate & High-Level Waste (ILW & HLW) elimination. However, as illustrated by leading projects in Finland, Sweden or France, the implementation of DGR remains complex, capital-intensive, subject to delays and some financial uncertainties.

By enabling the separation of valuable materials (Uranium and Plutonium) and the safe and long term conditioning of waste, reprocessing significantly reduces both the volume by a factor of 5 and the radiotoxicity of ultimate waste by a factor of 10. This directly alleviates and delays constraints on national strategies planning for “short-term” storage systems and DGRs.

Reprocessing also contributes to the desaturation of spent fuel pools in the short to medium term, providing operators with additional time to align infrastructure development with actual needs.

From a financial perspective, it allows for an optimization of the CAPEX timing. By smoothing capital expenditures over time and leveraging the dynamics of dedicated provision funds, countries can strengthen their capacity to finance future DGR projects under more favorable

conditions while reducing the risks of SNF management, replaced by optimized waste forms. Such “reprocessed waste forms” (i.e., vitrified and compacted waste canisters) benefit from well-characterized thermal and radiotoxic behavior, preventing any scattering of fission waste into the environment and offering a high level of predictability for long-term management.

In a context where uncertainty is becoming the main driver of risk in spent fuel management, the ability to introduce flexibility and improve predictability is essential.

Reprocessing should not be seen as an alternative to geological disposal, but as a complementary and enabling solution.

Furthermore, a comprehensive evaluation of material accessibility underlines the strategic value of recycling uranium and plutonium. While immediate domestic recycling infrastructure may vary across different national programs, the technical capability to supply Mixed Oxide (MOX) and Reprocessed Uranium (ERU) fuels represents a proven vector for long-term resource circularity and strategic fuel diversification.

By establishing early technical cooperation focused on fuel data and waste package integration, nuclear stakeholders can transition from a reactive storage posture toward safer, diversified and more resilient fuel cycle management.