IAEA Technical Meeting on Advances and Innovations in Fast Reactor Design and Technology
IAEA headquarter, Vienna, 29 September 2025

IAEA Activities on Fuels for Fast Reactors

Anzhelika Khaperskaia,

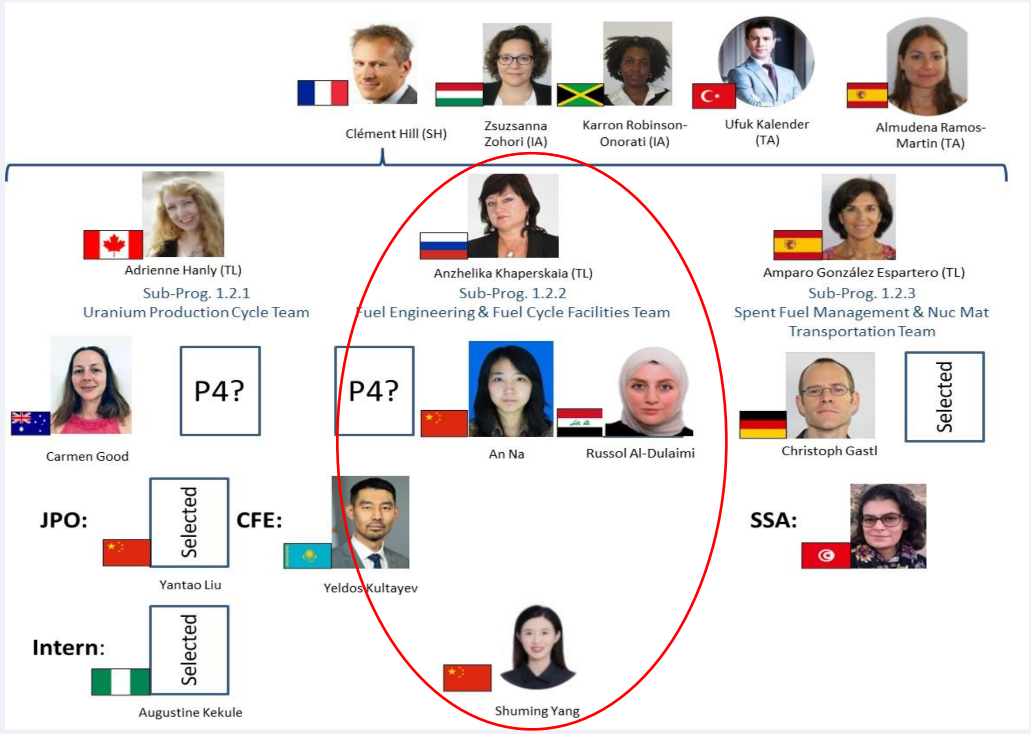
Technical Lead

NFCMS /NEFW, IAEA

OUTLINE

- Introduction (IAEA NFCMS structure and activities)
- Completed and future CRPs on Fast Reactors Fuel
- Other activities on FR fuel development (publications, e-learning etc.)

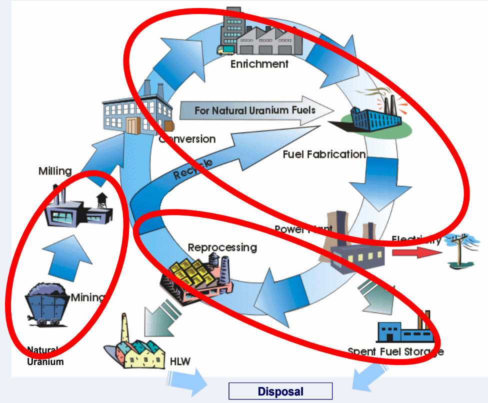
Nuclear Fuel Cycle and Materials Section (NFCMS)



NE | NS | NA | TC | SG | MT

Division of Nuclear Fuel Cycle and Waste Technology (NEFW)

- RRS
- WTS
- **NFCMS**
  - UPC Team
  - **NFE Team**
  - SFM Team
- DERS



- **Project 1.2.2.001 Nuclear Power Reactor Fuel Engineering and Operation:** Support Member States (MSs) to understand and address factors affecting the design, fabrication and in-pile behaviour of currently operating and innovative nuclear fuels and materials for power reactors.

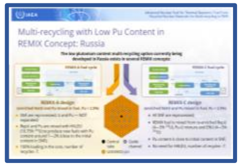
- **Project 1.2.2.002 Fuel Cycle Facilities Operation and Life Management :** Support MSs to technically implement IAEA Safety Standards when operating or upgrading existing nuclear fuel cycle facilities, and to understand and address factors affecting the ageing of these facilities

IAEA ongoing activities to support fuel & NFCFs development & operation

Water-cooled reactor fuels

- **Accident Tolerant Fuels**
    1. **CRP T12032** on "*Testing and Simulation of Advanced Technology and Accident Tolerant Fuels* (ATF-TS)" (2020-2024)
    2. TM on "Advanced Technology Fuels: Progress on their Design, Manufacturing, Experimentation, Irradiation, and Case Studies for their Industrialization, Safety Evaluation, and Future Prospects" (28-31 October 2025)
    3. **Proposed new CRP** on "*Testing and performance simulation of Advanced Technology Fuels (ATF- HBU)* (2026 -?)

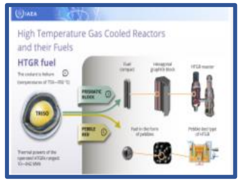
- **Fuels for recycling/multi-recycling**
    1. TECDOC on "*Mixed Oxide Fuels Design, Operation and Management*" (in preparation to publishing)
    2. IAEA publications on "*Challenges and Opportunities in Reprocessed Uranium Fuels*" (TECDOC in progress)
- **Conventional Water-Cooled Reactor fuels**
    1. NES on "*Review of Fuel Failures in Water Cooled Reactors (2016–2020)Rev.1*" (under internal review)
    2. TECDOC on "*Structural Behaviour of Fuel Assemblies in Water Cooled Reactors*" (in preparation to publishing)
    3. TM on "*Advances in Fuel Design, Manufacturing and Examinations for Pressurized Heavy Water Reactors*" (November 2024, Argentina)
    4. TM on "*Digitalization and the Use of Artificial Intelligence in Advanced Nuclear Fuel Manufacturing and Quality Control*", July 2026



Innovative GEN-IV and SMR fuels

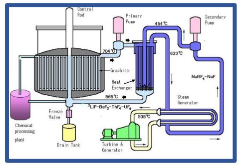
- **Water-cooled SMR fuels**
    1. *Workshop on "Core and Plant Simulation with an Emphasis on Fuel Behaviour in Light Water Reactor Based Small Modular Reactors"* (27-29 February 2024, TECDOC in progress)
- **Fast reactor fuels**
    1. ***CRP T12031** on "Fuel Materials for Fast Reactors (FMFR)"* (2019-2023: Final Report in preparation to publishing)
    2. *NES Technical Report on "Nuclear Fuel Technologies for Liquid Metal Cooled Fast Reactors (LMFRs)"* (in preparation to publishing)
    3. *Workshop on the "Behaviour of Liquid Metal Cooled Fast Reactors Fuels"* (30 June – 04 July 2025)
    4. **Proposed new CRP** on *"Benchmark Exercises on Testing and Performance Simulation of Advanced Fuels for Liquid Metal-Cooled Fast Reactors"* (2026-…)

- **Gas-cooled SMR fuels**
    1. ***CRP T12034** on "Fuel Modelling Exercises for Coated Particle Fuel for advanced reactors including SMR"*



Molten salt SMR fuels

1. *New Simulation tool module development for MSR with relevant fuel cycle*
2. *Workshop on "Molten Salt Reactor Fuel: Recent Development and Future Challenges"* (21-25 July 2025)
3. *Workshop on "Current Status of Structural Material Development for Molten Salt Reactors and Related Challenges"* (July 2026)



HALEU fuel

1. *Workshop on "Operational Aspects of Manufacturing High Assay Low Enriched Uranium Advanced Fuels"* (August 2026)

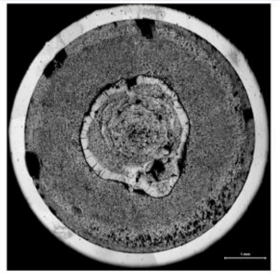
PIE for SMR fuels

1. ***CRP T12033** on "Standardization of Subsized Specimens for Post-Irradiation Examination and Advanced Characterization of Fuel and Structural Materials for Small Modular Reactor and Advanced Reactor Applications (PIE for SMR)"*

CRP T12031 (FMFR) on “*Fuel Materials for Fast Reactors*” (2019 - 2023)

Objective of the CRP

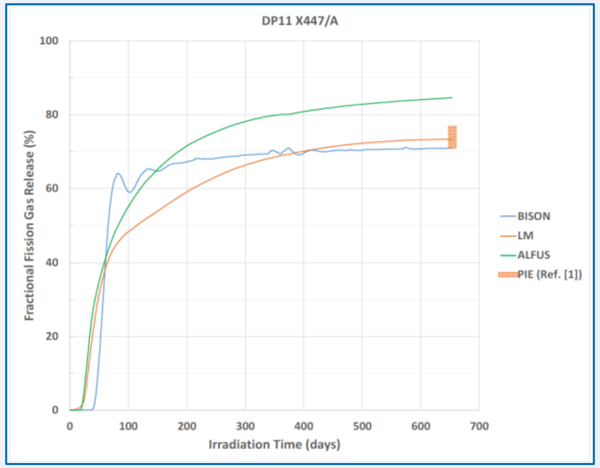
To support the performance assessment of fuel and cladding materials for sodium-cooled fast reactor technology, in line with Generation IV (Gen-IV) objectives, by enhancing the capabilities of fuel performance codes.



Optical metallographic images of METAPHIX-2 #1 test fuel pin

The focused areas of the CRP :

- Compile and share irradiation test data on fast reactor fuel materials, specifically mixed oxide (MOX) and metallic fuels (U/Pu-based alloys), as well as steel-based claddings; Optimize the use of limited experimental data by promoting joint utilization for mutual benefit.
- Perform simulations using different fuel performance codes based on the collected datasets; Compare, analyse, and share the results, develop recommendations on fuel performance codes enhancement and identification of gaps in irradiation data



Fission gas release for X447 DP11 calculated by ALFUS comparing with measured data and those calculated by BISON and LIFE-METAL(LM)

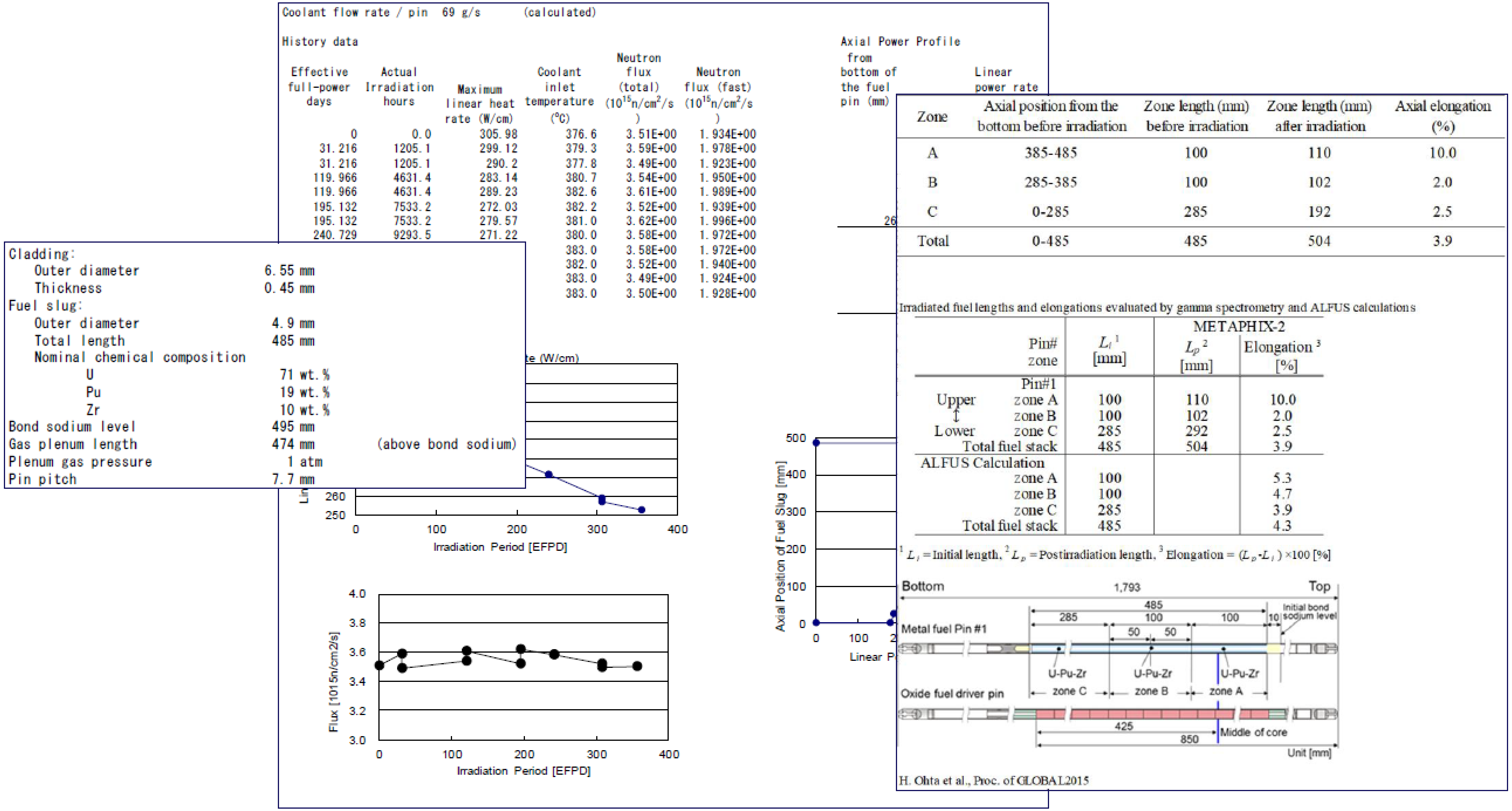
The CRP on **Fuel Materials for Fast Reactors (FMFR, 2019–2023)** was the first IAEA Coordinated Research Project to conduct a benchmark exercise for fuel performance codes for FR fuel

The CRP focused on **fuel behaviour under nominal operating conditions** and served as a complementary initiative to the **NEA/OECD Expert Group on Innovative Fuel Elements (EGIFE)**, which focuses on benchmarking fuel performance codes under **transient and accident conditions**

IRRADIATION DATA PROVIDED AND FUEL PERFORMANCE CODE USED BY EACH CRP PARTICIPANT

Country	Organization		Irradiation data	Code
India	IGCAR	Oxide	FBTR	CAMOX
Japan	JAEA	Oxide	B5D-2 (Joyo)	CEPTAR
	CRIEPI	Metal	METAPHIX-1#1 (Phenix) METAPHIX-2#1 (Phenix)	ALFUS
KOREA	KAERI		HT9 cladding (BOR-60)	MACSIS
FRANCE	CEA	Oxide	SANTENAY (Phenix) SUPERFACT#4-16 (Phenix)	GERMINAL
USA	ANL	Metal	X447 DP11 (EBR-II)	LIFE-4 LIFE-METAL BISON
	INL	Oxide	FO2 (FFTF)	BISON (Oxide)
European commission (EC)	JRC	Metal Oxide	METAPHIX-1#1 (Phenix) METAPHIX-2#1 (Phenix) SUPERFACT#4-16 (Phenix)	(TRANSURANUS)

Example of FMFR output


Coolant flow rate / pin 69 g/s (calculated)

History data

Effective full-power days	Actual Irradiation hours	Maximum linear heat rate (W/cm)	Coolant inlet temperature (℃)	Neutron flux (total) (10^{15}n/cm^2/s)	Neutron flux (fast) (10^{15}n/cm^2/s)
0	0.0	305.98	376.6	3.51E+00	1.934E+00
31.216	1205.1	299.12	379.3	3.59E+00	1.978E+00
31.216	1205.1	290.2	377.8	3.49E+00	1.923E+00
119.966	4631.4	283.14	380.7	3.54E+00	1.950E+00
119.966	4631.4	289.23	382.6	3.61E+00	1.989E+00
195.132	7533.2	272.03	382.2	3.52E+00	1.939E+00
195.132	7533.2	279.57	381.0	3.62E+00	1.996E+00
240.729	9293.5	271.22	380.0	3.58E+00	1.972E+00
			383.0	3.58E+00	1.972E+00
			382.0	3.52E+00	1.940E+00
			383.0	3.49E+00	1.924E+00
			383.0	3.50E+00	1.928E+00

Axial Power Profile

from bottom of the fuel pin (mm)	Linear power rate
26	

Cladding:		
Outer diameter	6.55 mm	
Thickness	0.45 mm	
Fuel slug:		
Outer diameter	4.9 mm	
Total length	485 mm	
Nominal chemical composition		
U	71 wt.%	
Pu	19 wt.%	
Zr	10 wt.%	
Bond sodium level	495 mm	
Gas plenum length	474 mm	(above bond sodium)
Plenum gas pressure	1 atm	
Pin pitch	7.7 mm	

Zone	Axial position from the bottom before irradiation	Zone length (mm) before irradiation	Zone length (mm) after irradiation	Axial elongation (%)
A	385-485	100	110	10.0
B	285-385	100	102	2.0
C	0-285	285	192	2.5
Total	0-485	485	504	3.9

Irradiated fuel lengths and elongations evaluated by gamma spectrometry and ALFUS calculations

Pin# zone		L_i [1] [mm]	METAPHIX-2 L_p [2] [mm]	METAPHIX-2 Elongation [3] [%]
Pin#1				
Upper	zone A	100	110	10.0
↕	zone B	100	102	2.0
Lower	zone C	285	292	2.5
	Total fuel stack	485	504	3.9
ALFUS Calculation				
	zone A	100		5.3
	zone B	100		4.7
	zone C	285		3.9
	Total fuel stack	485		4.3

[1] L_i = Initial length, [2] L_p = Postirradiation length, [3] Elongation = $(L_p - L_i) \times 100$ [%]

H. Ohta et al., Proc. of GLOBAL2015

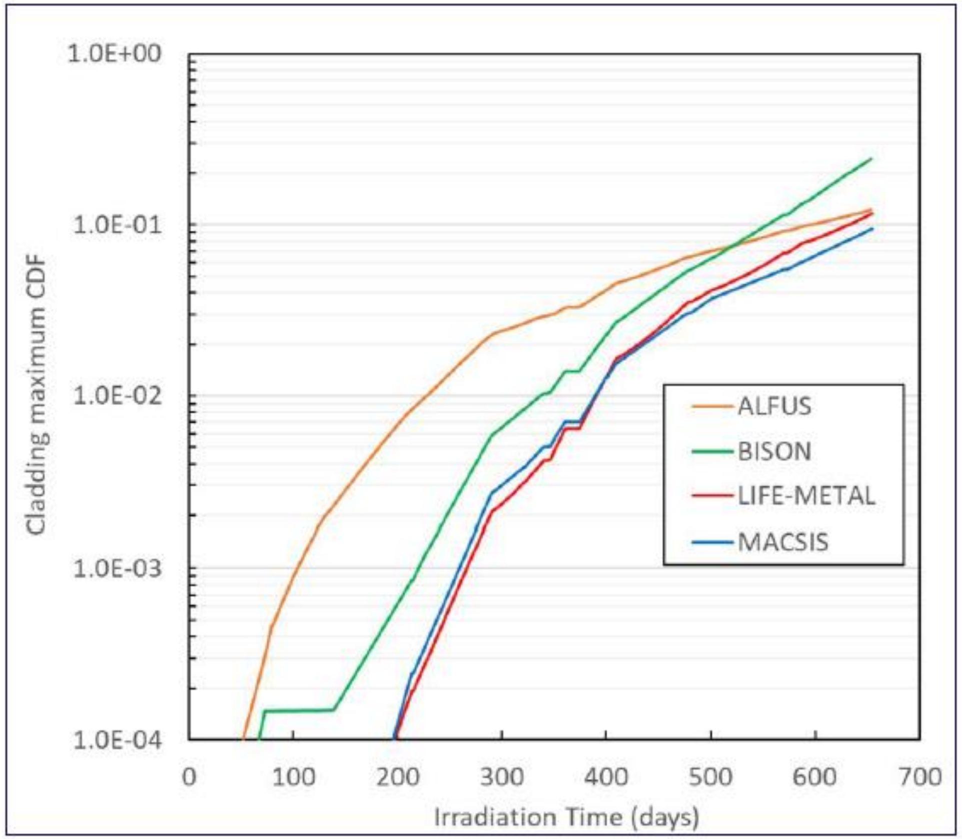
The irradiation test data sets on fuel materials, including oxide & metallic fuels and steel-based claddings, have been shared among the participants

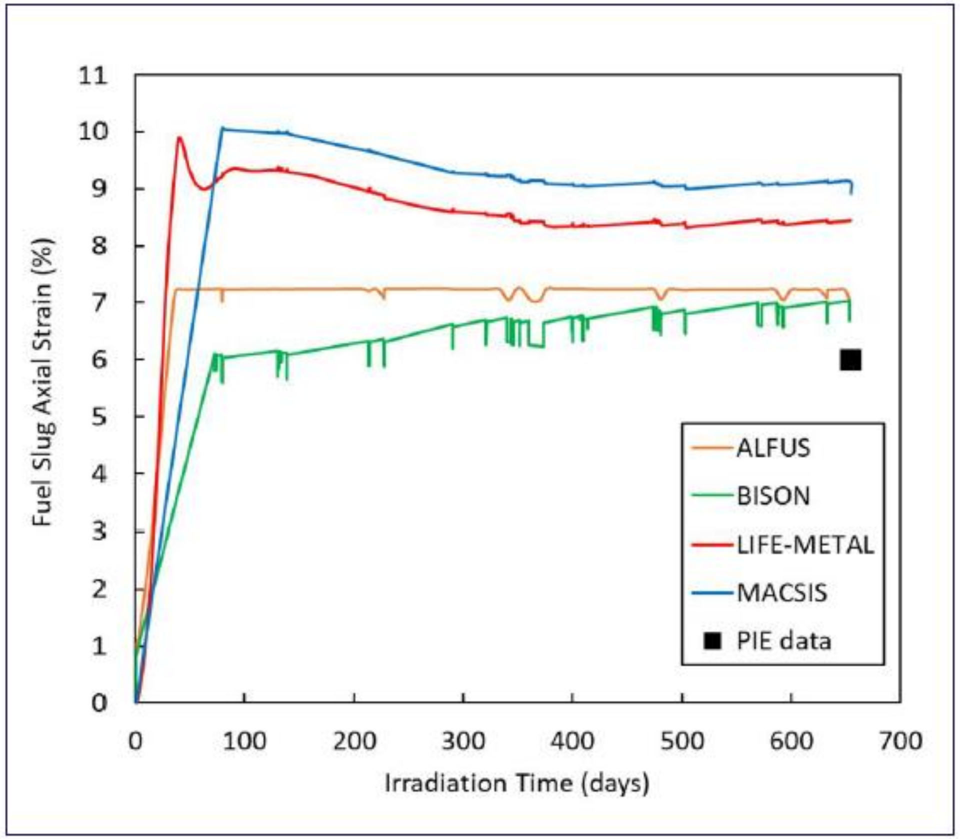
PARTICIPATION OF FUEL PERFORMANCE CODES IN EXERCISES

Oxide fuel	**India**	**Japan**	**France**	**USA**	
	IGCAR	JAEA	CEA	ANL	INL
	CAMOX (Oxide)	CEPTAR (Oxide)	GERMINAL (Oxide)	LIFE-4 Rev.1 (Oxide)	BISON (Oxide)
FBTR MOX	+	+	+	-	-
Joyo B5D-2	+	+	+	+	+
Phenix SANTENAY	+	+	+	+	-
FFTF FO2	+	+	+	+	+
Phenix SUPERFACT	+	+	+	+	-

Metallic fuel	**Japan**	**Korea**	**USA**	
	CRIEPI	KAERI	ANL	INL
	ALFUS (Metal)	MACSIS (Metal)	LIFE-METAL BISON (Metal)	BISON (Metal)
Phenix METAPHIX 1	+	+	+	+
EBR-II X447 DP11	+	+	+	+
Phenix METAPHIX 2	+	+	+	+

Example of FMFR output



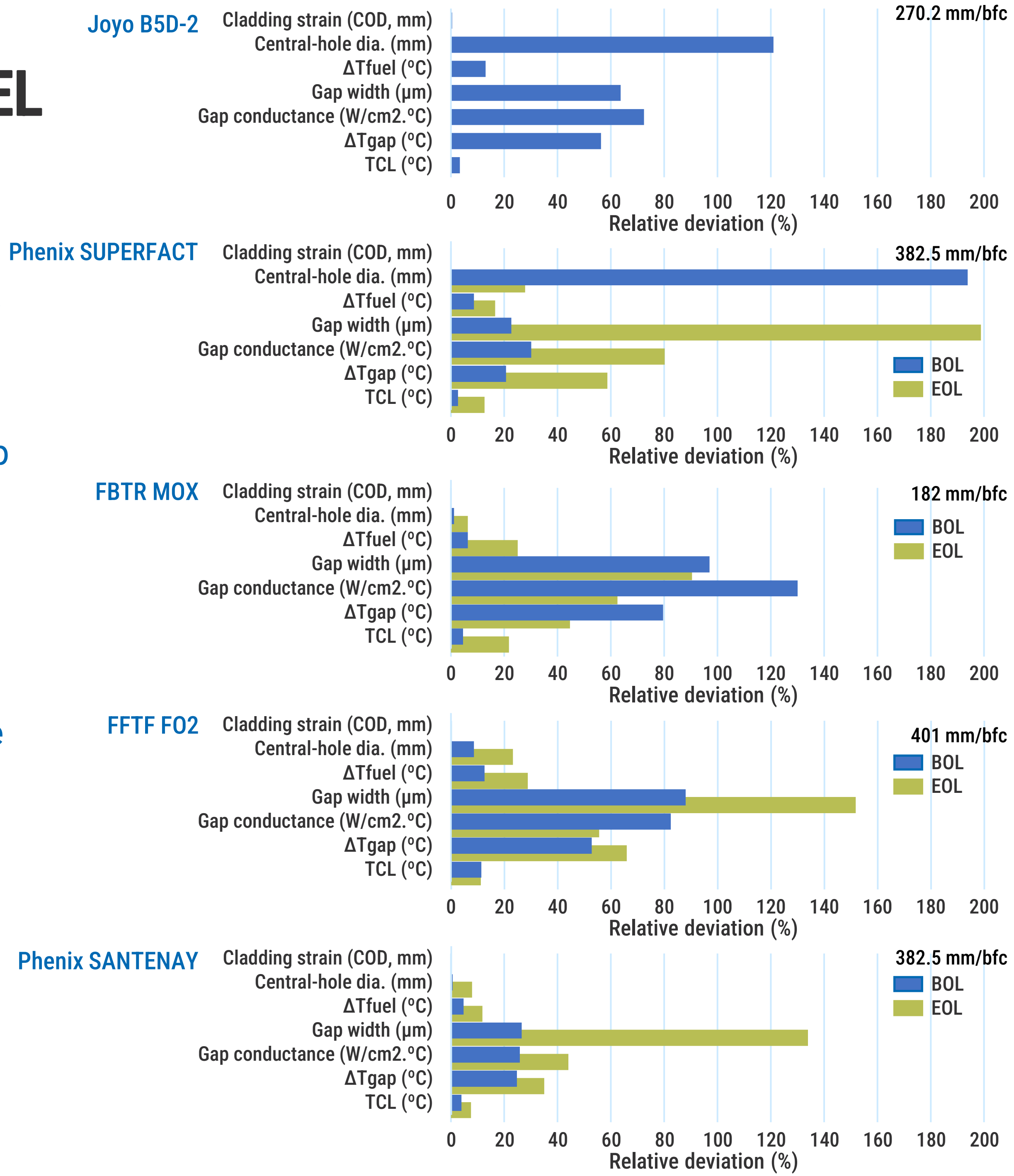


The results of code-benchmark exercises have been compared and discussed among the participants

BENCHMARK EXERCISES FOR OXIDE FUEL

code-to-code comparison results

- There are **some deviations among the calculation results** by the codes, probably **due to differences in the models installed in the codes**.
- Deviations for **gap performance calculations,** i.e. ΔTgap, gap conductance and gap width, **were comparably greater** in all cases. In addition, deviations for central-hole calculations at BOL were greater in solid fuels for Joyo B5D-2 and Phenix SUPERFACT.
- Since these deviations, if assimilated to uncertainties, have an impact on the safety margins of these fuels but also have an impact on the fuel performances, it appeared the need to work on the reliability of certain models which have a strong impact on the dispersions between the results.
- The **models with a strong impact** are the following:
  - ✓ **Gap closure** (pellet relocation, swelling and JOG formation) model
  - ✓ **Fuel restructuring model**

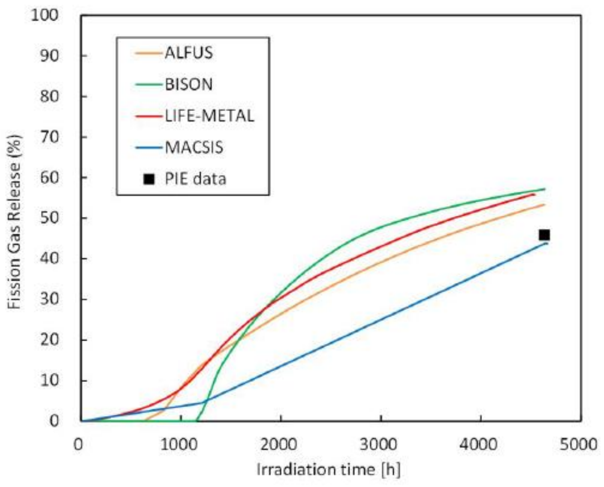


BENCHMARK EXERCISES FOR ME FUEL

code-to-code comparison results

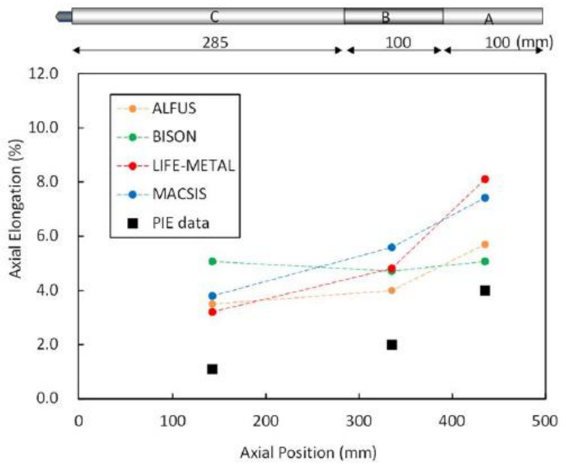
- **ALFUS, BISON, LIFE-METAL, and MACSIS,** in general, can properly calculate the irradiation behaviour characteristic to metal fuel, including fission gas release, fuel slug swelling (axial elongation), gas plenum volume decrease, FCCI, cladding CDF.
- But there are some differences among the calculation results by the codes, probably due to differences in the models installed in the codes. For future codes development, it would be useful to compare model assumptions and assess how calculation outcomes depend on irradiation conditions and fuel specifications.

FMFR CRP`s benchmark exercises lack uncertainty quantification for both PIE data and code predictions. PIE uncertainties arise from irradiation condition estimates and analytical errors; while code uncertainties are linked to model assumptions and material properties. ***Future benchmark work should include systematic evaluation of these uncertainties, as they are essential for comparing experimental and calculated results.***



Fission gas release

METAPHIX-1#1



Fuel slug axial elongation

CRP FMFR results : TECDOC

TECDOC

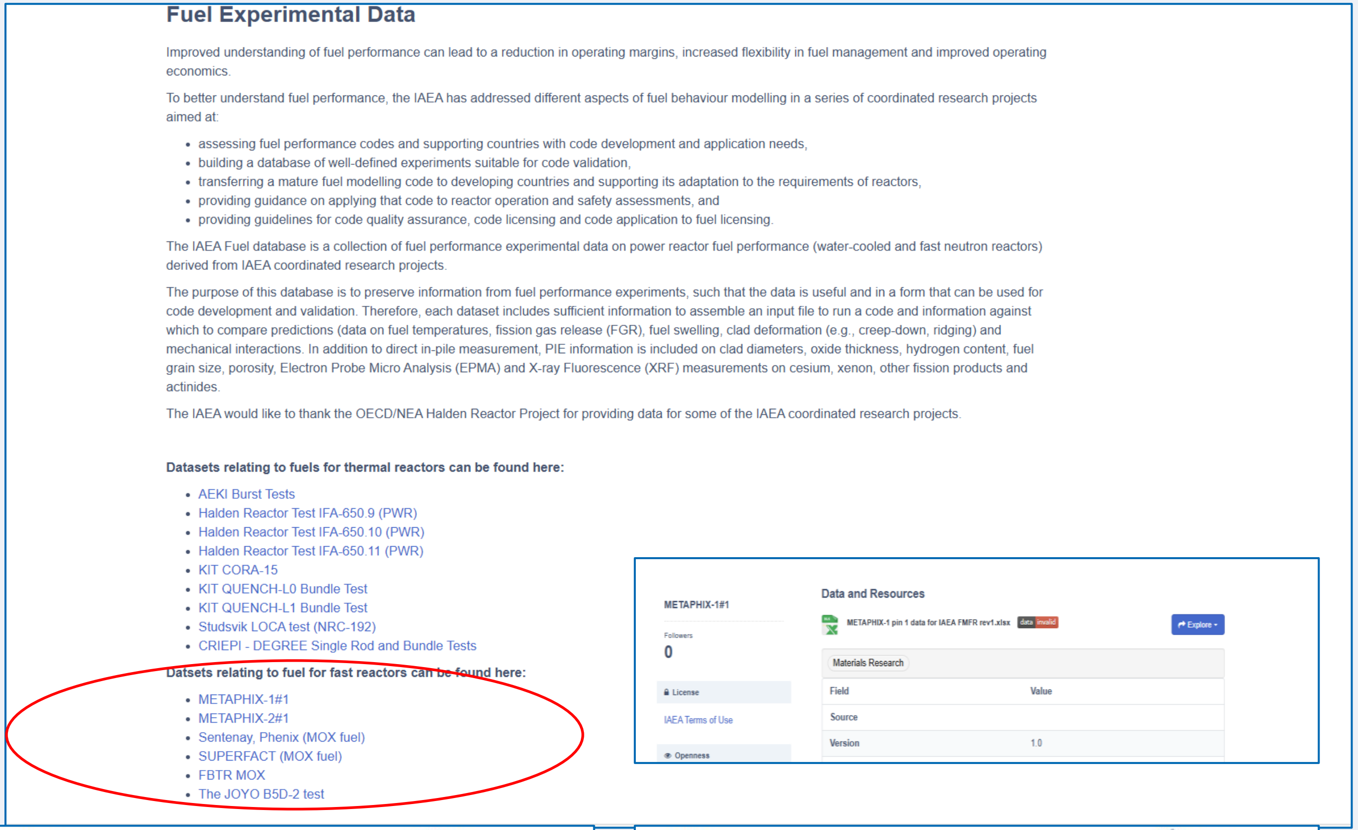
"Fuel Materials for Fast Reactors (FMFR) (2019-2023). Final report of a Coordinated Research Project"

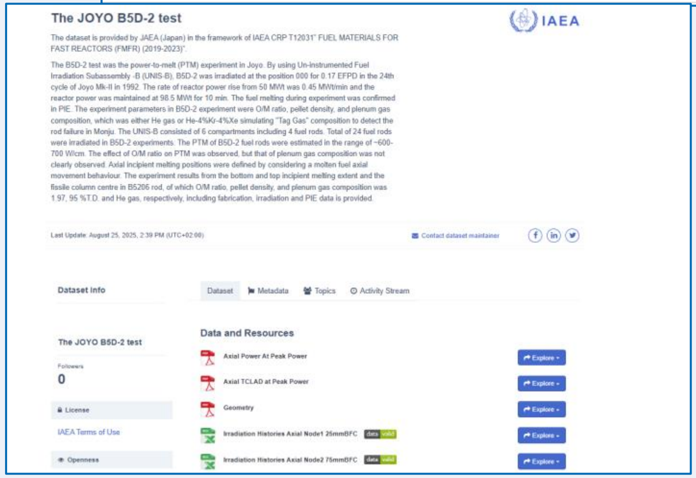
is under preparation to publication

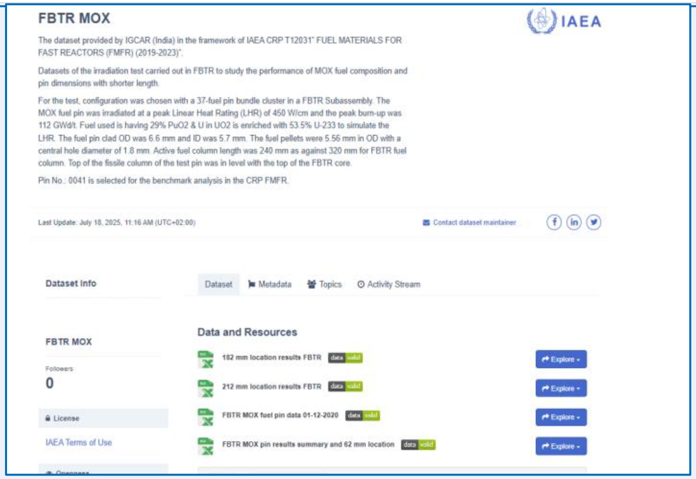
CRP FMFR results: experimental data

Experimental data have been uploaded to the **IAEA Fuel and materials Database**

[The IAEA Fuel and Material Database - IAEA Data Platform]







[Fuel Experimental Data - IAEA Data Platform]

Front End of the Nuclear Fuel Cycle Webinar Series

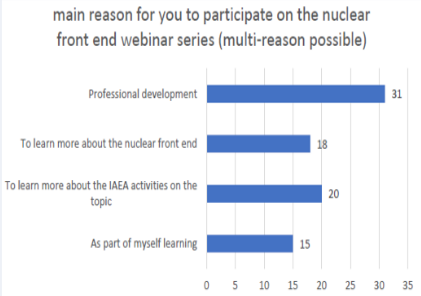
Nuclear Fuel Engineering and Nuclear Fuel Facilities Topics

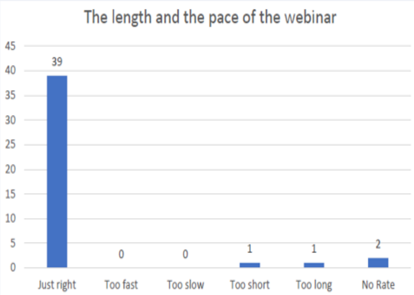
- **Webinar 1:** Quality and Reliability Aspects in Nuclear Power Reactor Fuel Engineering. Guidance and Best Practices to Improve Nuclear Fuel Reliability and Performance in Water-Cooled Reactors (Recording), 05 February 2025
- **Webinar 2:** Results of the Coordinated Research Project on "Testing and Simulation for Advanced Technology and Accident Tolerant Fuels" (CRP ATF-TS), 26 March 2025
- **Webinar 3:** Results of the Coordinated Research Project on "Fuel Materials for Fast Reactors (CRP FMFR), 9 April 2025

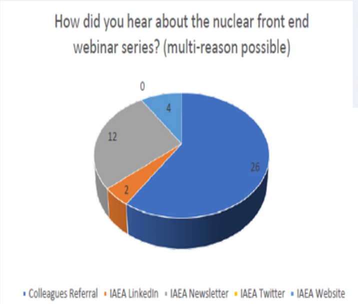
Webinar : Nuclear Fuel Reliability and Performance in Water-Cooled Reactors

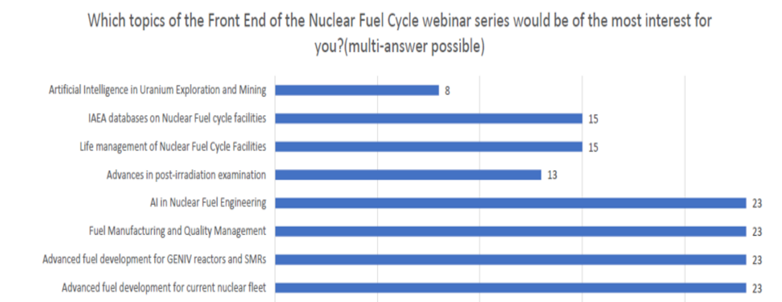
(2025-02-05)

Preliminary registered : more than 250 people
Participated: 135 from 46 MSs









Dr. Jinzhao Zhang

Global Expert and Technical Director at Tractebel (ENGIE)

- Ph.D. from UCLouvain (1993).
- 40+ years in nuclear reactor R&D, engineering, and consulting.
- Leads nuclear fuel design and safety analysis at Tractebel.
- Active member or chair in international working groups on fuel performance and safety at IAEA (TWGFPT) and OECD/NEA (EGRFP, WGFS, WGAMA).
- Co-chair and coordinator of the CRP ATF-TS.

Nicolas WAECKEL

Consultant, France

PHD and a post-PHD Thesis on Structural Mechanics - INSA Lyon 1978

40+ years in nuclear R&D and engineering
Led nuclear fuel design and safety at EDF and EPRI
EDF Corporate Expert, Fuel & Core safety analysis and international R&D activities (IAEA, NEA, NFIR, HALDEN, Studsvik, etc.)

Retired in 2020

Dr. Martin Ševeček

Czech Technical University and UJP Praha

Dr. Martin Ševeček is a nuclear engineer educated at the Czech Technical University in Prague, National Tsing Hua University, and Massachusetts Institute of Technology.
He has broad international professional experience in nuclear fuel development, testing, qualification, and licensing from academia, industry, and national regulator.
In his position at Czech Technical University and UJP Praha, he leads several national and international nuclear materials-related projects and teaches fuel performance- and heat transfer-related Master and Bachelor courses.
He is also active in IAEA and OECD/NEA expert groups, he's been a vice chair of the Working Group on Fuel Safety at OECD/NEA since 2021.

Dr. Antoine Bouloré

Senior Expert (CEA)

- Ph.D. from Ecole des Mines, Saint-Etienne (2001).
- 25+ years in LWR nuclear fuel modelling and simulation at CEA (Fuel Research Department)
- Active in IAEA projects (FUMAC and ATF-TS)
- Participant in OECD/NEA expert groups (EGMUP and EGRFP)

Dr. Marco Cherubini

Head of Core behavior at NINE (Nuclear and Industrial Engineering)

- Ph.D. from University of Pisa (2008).
- 20+ years in nuclear reactor safety, licensing , R&D, engineering, and consulting.
- Lead nuclear fuel analysis at NINE.
- Active in IAEA and OECD/NEA working groups and related sponsored projects.
- Contributor to IAEA's TECDOC and NEA reports notably on nuclear fuel safety

Dr. Juri Stuckert

Head of Nuclear Safety Research Group at KIT

Ph.D. from Russian Academy of Sciences (2003).
- 30+ years in nuclear reactor safety, experimental and modelling research, and consulting.
- Responsible for the QUENCH facility at KIT.
- Active in IAEA and OECD/NEA working groups and numerous accident research projects.
- Contributor to IAEA's TECDOC and NEA reports.
  - https://orcid.org/0000-0002-3974-2592
  - https://www.iam.kit.edu/awp/english/163.php

Dr. Takanari Ogata

Research Advisor,
Central Research Institute of Electric Power Industry

- 1987-present: CRIEPI
- PhD: Nuclear Engineering, Kyoto University, 2000
- More than 35 years experience of metal fuel R&D

- Fellow, Atomic Energy Society of Japan
- Chair of Nuclear Fuel Division, Atomic Energy Society of Japan
- American Nuclear Society
- OECD/NEA/NSC/WPFC Expert Group of Innovative Fuel Element
- Japan representative of IAEA Technical Working Group on Fuel Performance and Technology (2017-2023)
- IAEA Coordinate Research Project "Fuel Materials for Fast Reactors" (2019-2023)

Mr. Takayuki Ozawa

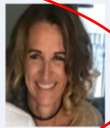
Principal Engineer,
Japan Atomic Energy Agency (JAEA)

Hokkaido University (1988).
More than 35 years in fast reactor fuel design & development, and fuel performance modeling.
Project coordinator in the field of Advanced Fuels in Fuel Cycle R&D and Waste Management Sub-Working Group of the Civil Nuclear Energy Research and Development Working Group (CNWG) between Japan and US.
Lead of fuel technology tasks in the SFR development program collaboration among CEA, Framatom, JAEA MHI and MFBR.
OECD/NEA/NSC/WPFC Expert Group of Innovative Fuel Element.
IAEA Consultancy Meeting on "the Status and Trends of Nuclear Fuel Technology for Fast Reactors" (2021-2023).
IAEA Coordinate Research Project "Fuel Materials for Fast Reactors" (2019-2023).

Ms. Nathalie Chauvin

Fuel Studies Department International Expert (CEA , France)

N. Chauvin has extensive experience in the Minor Actinides transmutation program, contributing to fuel design optimization, irradiation experiments, and synthesis reports.
She served as a project manager for the development of fuels for the Gas-Cooled Fast Reactor, focusing on oxide/carbide fuels and refractory cladding, including ceramic composites for both pin and plate-type fuel elements.
Currently, she leads international collaborations on fast reactor fuel development, holding key roles such as:
Chair of the Working Party on the Fuel Cycle (WPFC) at the OECD Nuclear Science Committee (NSC).
Chair of the Expert Group on Innovative Fuel Elements at OECD/NSC/WPFC.
Coordinator of the PUMMA project under EURATOM H2020.
CSI in the IAEA Coordinated Research Project (CRP) on Fuels and Materials for Fast Reactors.
Additionally, she actively participates in various scientific committees of international conferences, including IEMPT, Fast Reactor, and GLOBAL. She also serves as the CEA counterpart in multiple bilateral collaborations dedicated to MOX fuel research with international scientific organizations.



Workshop on Fuel Performance Assessment and Behaviour for Liquid Metal Cooled Fast Reactors: 30 June – 04 July 2025

1. The outcomes of the Coordinated Research Project (CRP) T12031 on Fuel Materials for Fast Reactors (2019-2023)
2. The current situation in MSs on liquid metal cooled FRs (including SMRs) fuel behaviour testing and modelling. Status of FR fuel performance codes
3. Properties of FR fuels, claddings, and coolants
4. Qualification of FR fuels: status and future needs
5. Future activity on FR fuel

37 presentations in total

39 experts	7 IAEA Staff	9 MSs	China
France	India	Italy	Japan
Korea	Russian Federation	Sweden	USA

Scope of the New CRP (Preliminary Title: Benchmark Exercises on Testing and Performance Simulation of Advanced Fuels for Liquid Metal-Cooled Fast Reactors)

Fuel Types to be Considered:

- Oxide fuels (UOX and MOX)
- HALEU
- Metallic fuels
- Nitride fuels
- Wide range of Pu content (including high Pu content)
- Minor actinide (MA)-bearing fuels, La-bearing fuel (as recycled fuel)
- Different fuel element design
- Diverse isotopic compositions of U and Pu

Materials Focus:

- Advanced cladding materials
- Cladding behaviour, including creep performance and response to transient conditions

Reactor Operating Conditions:

- High burnup operation
- Low linear heat rates
- Long operational fuel cycles
- Both nominal and off-normal (transient) scenarios
- Evaluation of margins to fuel melting
- Assessment of fuel failure safety margins

Analytical Focus Areas:

- Sensitivity and uncertainty analysis
- In-depth comparison of modelling approaches and parameters used in various fuel performance codes

Accelerated Testing Methods (Proposed as a Dedicated Work Package)

To support faster development and validation of advanced FR fuel systems, the CRP could integrate accelerated testing methods, including:

- Use of small specimens
- Ion irradiation techniques as surrogates for neutron irradiation to simulate radiation-induced damage
- Application of diverse Post-Irradiation Examination (PIE) techniques for material characterization

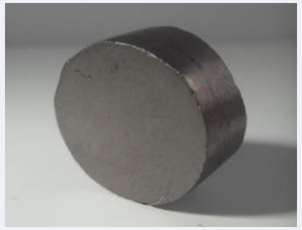
Planned Timeline for the New CRP (preliminary title: “Performance Simulation of Advanced Fuels for Liquid Metal Cooled Fast Reactors”

- **Early 2026** – Convening of a Consultancy Meeting to shape the new CRP, including:
  - Detailed discussions on available experimental datasets and codes for benchmark exercises
  - Defining the scope of the CRP and specific tasks
  - Drafting the CRP Proposal
  - Formulating the CRP Logical Framework
- **Mid 2026** – Finalization of the CRP Proposal
- **Q3 2026** – Submission of the CRP Proposal for review and approval by the Committee for Coordinated Research Activities (CCRA)
- **Q4 2026** – Official launch of the CRP on fast reactor fuels

IAEA ongoing activities to support the development of Fast reactor fuels . NES report “Nuclear Fuel Technologies for Liquid Metal Cooled Fast Reactors (LMFRs)”

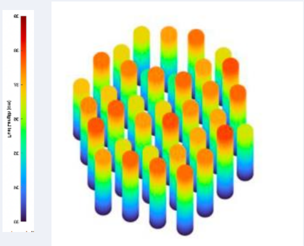
The state-of-the-art report with new technical information on the design, fabrication, and operation of fuels for Liquid Metal cooled FRs (LMFRs), including SMRs:

- Current situation with FR, including SMRs, FR core and fuel assemblies;
- Member Stats's activities in LMFRs and its fuel cycle;
- Oxide fuels for LMFRs (fabrication technologies and experience, irradiation experience, fuel failure and irradiation behaviour in LMFRs, advanced fuels with minor actinides);
- Nitride fuels for LMFRs;
- Metallic fuels for LMFRs;
- Cladding materials for LMFRs;
- Fuel performance evaluation codes;
- Summary and recommendations.

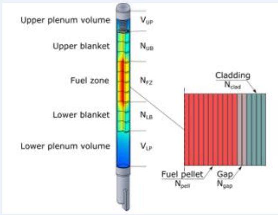


UN pellet produced by spark plasma sintering. Courtesy of KTH, Sweden.

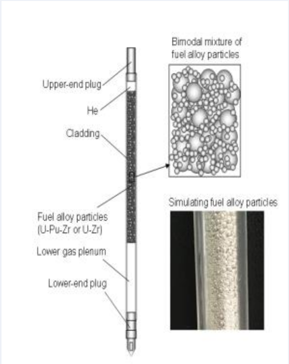
BN-600 Fuel assemblies with (U Pu) N Courtesy of VNIINM, Russia



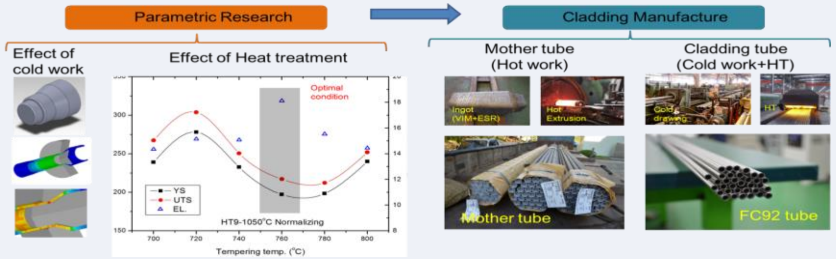
Example BISON Simulation of Axial Fuel Growth in an EBR-II Experiment



Fuel rod discretization scheme of the BERKUT-U code



He bonded particulate metallic fuel pin concept



Manufacture of FC92 cladding tube

E-Learning coure on Nuclear Fuel Engineering, Fabrication and Operation Behaviour

The course comprises 4 Modules with 17 Lectures in total


IAEA
International Atomic Energy Agency
Atoms for Peace and Development

CERTIFICATE of COMPLETION

This is to certify that

Anzhelika Khaperskaia

has successfully completed

Nuclear Fuel Engineering, Fabrication and Operation Behaviour E-learning course

Module 1. General fuel overview, design, fuel fabrication, surveillance, and qualification process

Each Lecture has a duration of about 45 minutes + Quiz (Q&A)

A certificate is awarded for the successful completion of each module.

[OPEN-LMS: All courses (iaea.org)]

- **Module 1. General Fuel Overview**
  - 1.1 General Overview
  - 1.2 Nuclear Fuel Description
  - 1.3 Nuclear Fuel Fabrication
  - 1.4 Thermal-mechanical and Irradiation Behavior of Fuel Rods
  - 1.5 Fuel Surveillance
  - 1.6 Fuel Qualification Aspects
- **Module 2. Conventional Nuclear Fuel for Thermal Reactors**
  - 2.1 Commercial Nuclear Fuel for Thermal Reactors (Overview)
  - 2.2 PWR Fuel
  - 2.3 BWR Fuel
  - 2.4 PHWR Fuel
- **Module 3. Nuclear Fuel for Advanced Reactors**
  - 3.1 Nuclear Fuel for GEN-4 Reactors and SMRs (Overview)
  - 3.2 FR Fuel
  - 3.3 HTGR Fuel
  - 3.4 MA fuel for Transmutation
- **Module 4. Advanced Nuclear Fuel for Thermal Reactors**
  - 4.1 Advanced Nuclear Fuel for Thermal Reactors (Overview)
  - 4.2 Fuels for Multi-recycling
  - 4.3 ATF

2023 – all e-learning lectures were published on the IAEA website
2024 - Translated into Chines (published)
2025 - Translation into Russia (ongoing)
Preparation for translation into Arabic

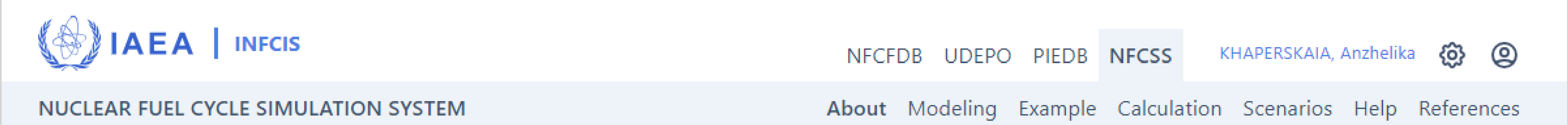
TWG FPT recommendation: NFE Network Task forces on specific areas

Recognizing the importance of advancements in nuclear fuel technologies and the renewed interest among IAEA MSs, the IAEA NFE&NFCFs team aims to effectively support MSs by leveraging a broader range of expertise and experiences. The TWG-FPT members have recommended the establishment of dedicated Task Forces (TFs) in specific technical areas:

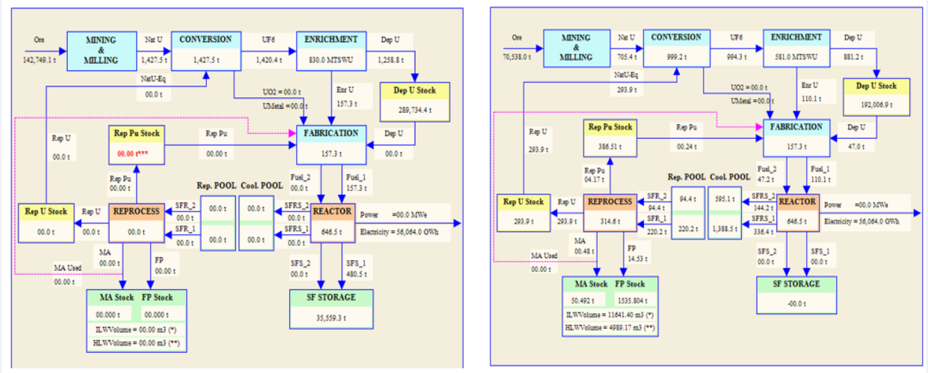
- **Light Water Reactor** (LWR) **Fuels**: Pressurized Water Reactor (PWR), Water-Water Energetic Reactor (WWER), Boiling Water Reactor (BWR).
- **Heavy Water Reactor** (HWR) **Fuels**: Pressurized Heavy Water Reactor (PHWR), CANada Deuterium Uranium (CANDU).
- **Fast Reactor (FR) Fuels**: Lead-cooled Fast Reactor (LFR), Sodium-cooled Fast Reactor (SFR).
- **Artificial Intelligence/Machine Learning** (AI/ML) in Nuclear Fuel Engineering.
- **Fuel Manufacturing and Quality Management.**
- **Molten Salt Reactor** (MSR) **Fuel**.

Task Force activities will be carried out through online working meetings (via Webex or MS Teams), or dedicated IAEA events such as Consultancy Meetings, Technical Meetings, and Workshops, and through the Nuclear Fuel Engineering (NFE) Network platform to enhance information exchange and collaboration.

Simulation tool (NFCSS)



- NFCSS is a scenario-based publicly available computer model (web-based tool) for the estimation of nuclear fuel cycle material and service requirements
- Reactors types: PWRS, BWRs, PHWRs, RMBKs, AGRs, GCRs, WWERs , FRs
- UOX, MOX and ThOX fuel cycles
- Calculates the requirements for Nat U resources, enrichment and fuel fabrication services, etc. SF inventory, Minor Actinide Inventory, FP inventory, Decay Heat and Radio-toxicity with material Flow Diagrams up to 200 years



Technical Features

- Only long-term actinides are calculated UOX, MOX fuel
- ORIGEN II (PWR-UO2-33G, PWR-MOX and BWR-UO2-27.5G, BWR-Pu) fuel libraries

  PWR (Pu-Th MOX) : ORIGEN II library (211) Pu-Th fuel (with modifications)

  BWR (Th fuel cycles) : ORIGEN II PWR Th- library (214)

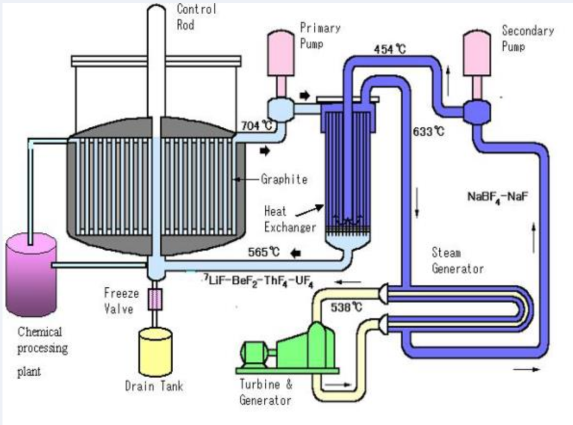
  Other reactors : Libraries provided by Consultant experts

Nuclear Fuel Cycle Simulation System (NFCSS)

	Open cycle	Closed Cycle	
The total natural uranium (2025 to 2110)*	337831 tones	202973 tones	
The total spent fuel (or HLW) accumulated in the end of life cycle	37228 tones of SNF	High-level waste	5405 tones
		Plutonium	478 tones
		Minor actinides	54.7 tones

* Example for calculation for NES with 10 PWRs (45GWd/t burn up) having a 60-years lifespan

NFCSS: on going activity



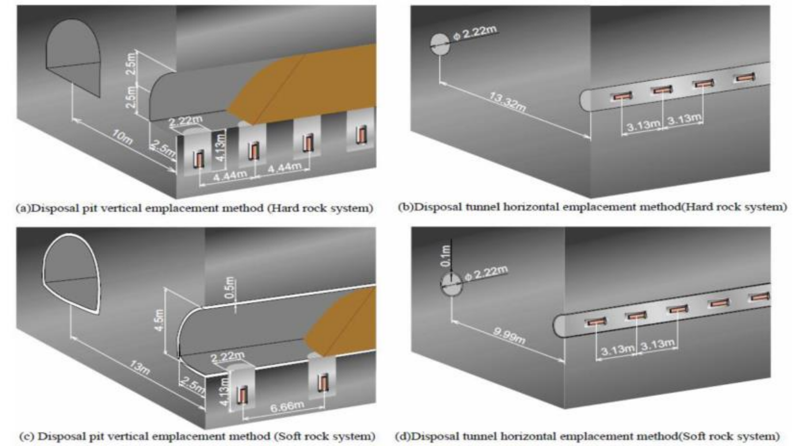
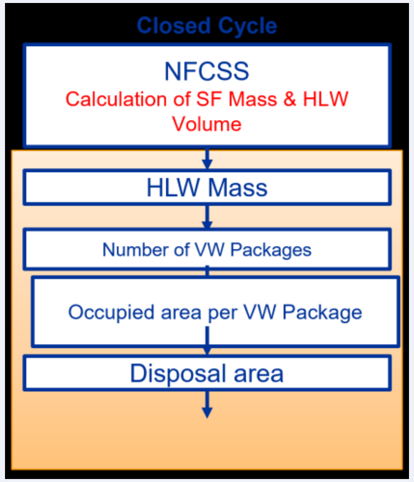
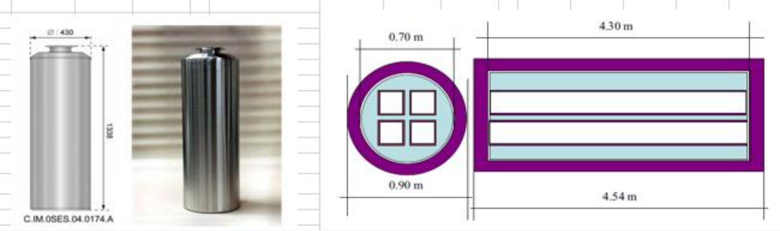


Figure 4.2.2-6 Specifications of the disposal tunnel and the waste package pitch

Upgrade recently completed:

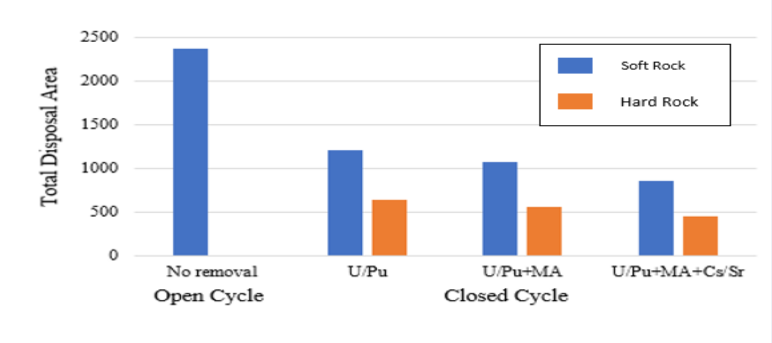
- Migration to new IT platform (using modern software development standards and principles, eliminate security vulnerabilities) during 2021-2023
- TECDOC-1868 on Thorium Fuel Cycle scenario Simulation





Ongoing:

- Development of simulation model for MSR fuel cycles
- Calculation for DGR sizing
- Development of fuel cycle cost calculation in progress



International Conference on Fuel Supply Chain for Sustainable Nuclear Power Development, Vienna, from 13 to 15 October 2026

Topic 1. Industry Prospects and Challenges Facing Raising Fuel Supply Demand:

Challenges in supply and front-end services to meet the increasing infrastructure requirements for conversion, enrichment and fuel fabrication

- Market fluctuations impacting supply chains
- Political support: policies and strategies
- Expansion of conversion enrichment and fuel fabrication facilities to meet growing demand
- Technological advancements improving processes' efficiency and sustainability

Topic 2. Supply and demand for raw materials for nuclear fuel supply:

Innovations in the front end of the nuclear fuel cycle, from exploration to mining:

- New uranium exploration and mining projects
- Innovative advancements in uranium exploration and mining
- Uranium and thorium resources, processing and mining and the circular economy

Topic 3. Advanced nuclear fuels for innovative reactor technologies:

Advanced technology fuels and fuels for advanced reactors:

- Design, qualification and operation of ATFs, LEU+ and HALEU fuels, TRISO fuels, fuels for Fast Reactors, MSRs and multiple recycling in all types of reactors
- Advances in nuclear fuel fabrication processes and quality control (automation, additive manufacturing and use of artificial intelligence)

Topic 4. Industrial and Innovative technologies for recycling nuclear materials:

Industrial operating experience and lessons learned in reprocessing for recycling:

- Experience of uranium (U) and plutonium (Pu) recycling and requirements for multirecycling U and Pu in thermal and Fast Reactors
- Prospects for reprocessing of spent fuels from advanced reactors, including minor actinides management
- Infrastructure development and implementation for reprocessing and managing nuclear materials to be recycled, including nuclear materials transport
- Life cycle of reprocessing and recycling and impacts on the final wastes to be disposed of

Concluding remarks

- The IAEA has long played a crucial role in supporting the development of advanced nuclear fuel technologies for many decades by providing platforms to exchange information, to coordinate research activity with international partners, documentation, etc.
- IAEA Member States are strongly encouraged to participate in topical meetings and CRPs, which provide valuable opportunities to engage with cutting-edge advanced nuclear fuel technologies that are critical to the future of nuclear energy

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