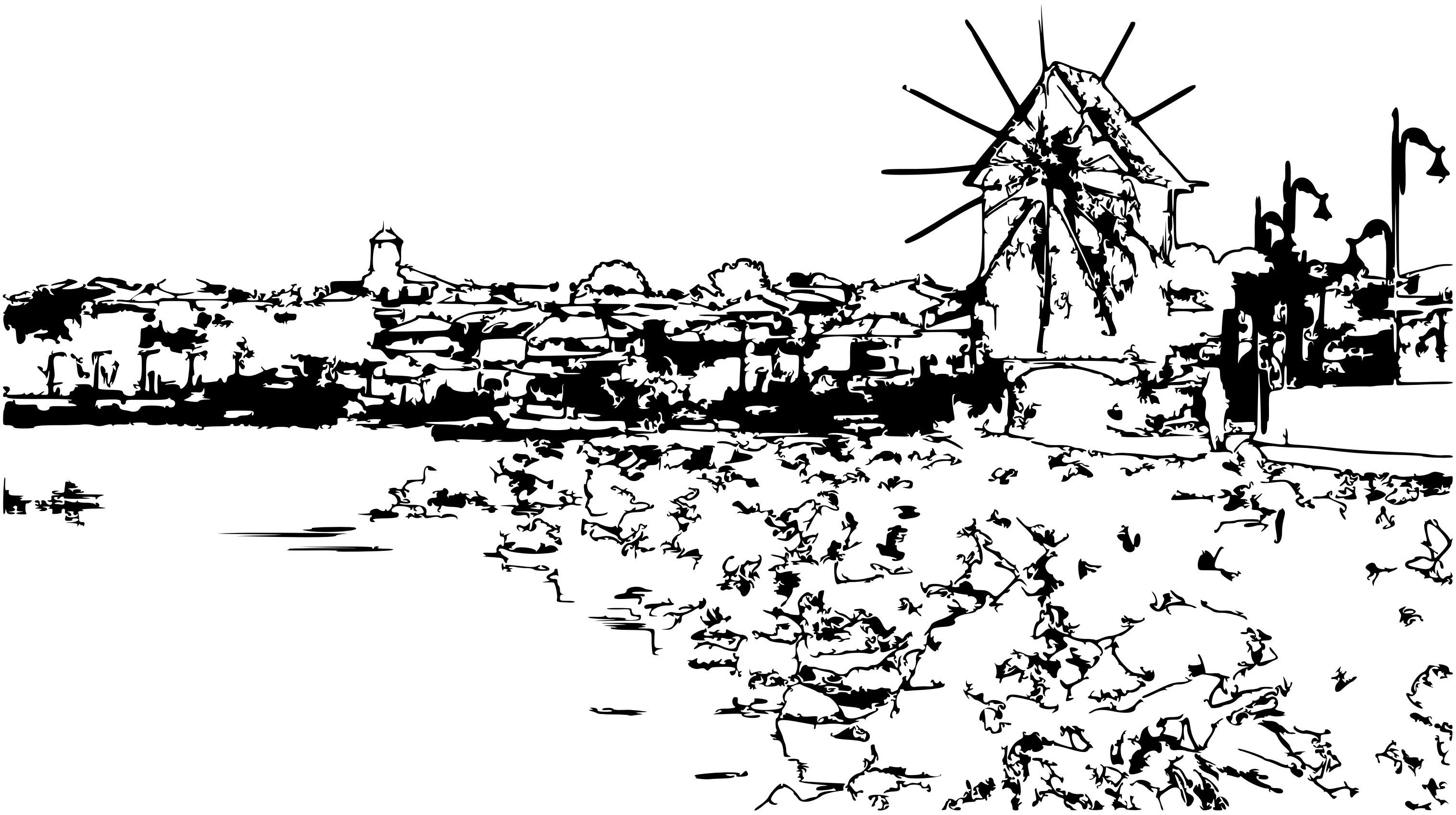
1ST INTERNATIONAL CONFERENCE ON LWR FUEL PERFORMANCE, MODELLING AND EXPERIMENTAL SUPPORT

LWR 2025

Book of Abstracts



14-19 September 2025, Nessebar, Bulgaria

1ST INTERNATIONAL CONFERENCE ON LWR FUEL PERFORMANCE, MODELLING AND EXPERIMENTAL SUPPORT

14-19 SEPTEMBER 2025, SOL NESSEBAR RESORT, NESSEBAR, BULGARIA
Hosted by the Institute for Nuclear Research and Nuclear Energy – BAS

AGENDA

Recommendation to the speakers: Keep 3-5 minutes of the time of your presentation for Q&A

Day 1: Sunday, 14 September 2025

10:00 – 17:00 **Registration**

19:00 – 20:00 **Welcome drink**

20:00 - 21:30 **Dinner – Italian Restaurant**

Day 2: Monday, 15 September 2025

10:00–11:00 **CONFERENCE OPENING SESSION**

Opening and greetings:

- *Ms. N. Kiselova - chairperson of the National Assembly of the Republic of Bulgaria,*
- *Mr. Lionel Gaiffe (Framatome) - Senior Executive Vice President,*
- *Ms. Kristina Ryttersson (Westinghouse) -Senior Director EMEA Fuel Engineering*
- *Dr. Emilia Georgieva (Studvik Scandpower GmbH)*

11:30 – 12:30 Session 1: Plenary Session

Chairpersons: ***Krasimir Kamenov, Dimitar Tonev***

<u>11:00–11:30</u> **Dimitar Tonev**
SCIENCE AND THE NUCLEAR INDUSTRY IN BULGARIA, D. Tonev[1], M. Manolova[1], A. Demerdjiev[1], E. Geleva[1], G.D. Dimitrova[1], V. Varijska[1], P. Vryashkova[1], D.M. Dimitrova[1], L. Nedelchev[1], Tz.K. Marinov[1], L. Pantaleev-Simeonova[1] ,V.N. Pavlova[1], N.N. Petrov[1], N. Goutev[1], B. Bonchev[1], M. Mitev[1,2], *[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, Sofia, [2]Technical University of Sofia, Sofia, Bulgaria*

<u>11:30–12:00</u> **Gwen Bolsee**
IMPROVING NUCLEAR ECONOMICS: TOWARDS LEU+ AND HIGH BURNUP FUEL FOR LIGHT WATER REACTOR APPLICATIONS,Gwen Bolsee[1]*, Steven Cole[2], Norm GARNER[2], *[1]Framatome SAS, Lyon – France,[2]Framatome Inc, Richland, WA, USA*

<u>12:00–12:30</u> **Krasimir Kamenov**
EXPERIENCE OF RWFA FUEL IMPLEMENTATION AT KOZLODUY NPP UNIT 5, K. Kamenov, D. Hristov, A. Kamenov, S. Kolev, M. Milchev, A. Avramov, T. Nikolova, V. Naev, *Kozloduy NPP, Bulgaria*

12:35 – 12:45 Conference photo

<u>12:30 – 14:00</u> Lunch Break

14:00 – 15:40 Session 2 Fuel performance and operational experience

Chairpersons: Krasimir Kamenov, Gwen Bolsee

<u>14:00–14:25</u> **Nicolas Lamorte**
FRAMATOME UNIQUE OPERATING EXPERIENCE ON 14FT PRODUCTS, N. Lamorte[1], G. Bolsee[1], *[1]Framatome, Lyon, France*

14:25–14:50 **Johann Plancher**
FUEL CYCLE OPTIMIZATION WITH FRAMATOME FARGO SOFTWARE, Johann Plancher[1]*, Greg Hobson[2], Hans-Wilhelm Bolloni[3], [1]*Framatome SAS, Lyon, France,* [2]*Framtome Inc., Lynchburg VA, USA,* [3]*Framatome GmbH, Erlangen, Germany*

14:50–15:15 **Luis Gonzalez**
VVER WATER CHEMISTRY COMPATIBILITY WITH WESTINGHOUSE FUEL, Luis Gonzalez[1]*, Mathilde Gaillard[1], Lena Oliver[1], Britta Helmersson[1]. [1] *Westinghouse Electric Sweden, Västerås*

15:15–15:40 **Michal Kuna**
WESTINGHOUSE LTA INSPECTION AT TEMELÍN NPP**,** Michal Kuna[1], Jan Höglund[2]**,** [1]*Westinghouse Electric Czech Republic, s.r.o., Czech Republic, Prague***,** [2]*Westinghouse Electric Sweden AB, Sweden, Västerås*

15:40- 16:00 **Coffee Break**

16:00 – 18:05 **Session 2 Fuel performance and operational experience (Cont.)**

Chairpersons: ***Akif Abdullayeu, Henri Rapeli***

16:00–16:25 **Johann Plancher**
FRAMATOME SOLUTIONS TO FLEXIBLE OPERATION, Johann Plancher[1]*, Alain Grossetete[1], Eric Corrieri[1], Lars Ackermann[2], [1]*Framatome SAS, Lyon, France,* [2]*Framtome GmbH, Erlangen, Germany*

16:25–16:50 **Johann Plancher**
EVOLUTION OF FRAMATOME CORE SIMULATORS, Johann Plancher[1], hilippe Bellanger[2], Bettina Hartmann[3], [1]*Framatome SAS, Lyon, France,* [2]*Framtome Inc., Lynchburg VA, USA,* [3]*Framatome GmbH, Erlangen, Germany*

<u>16:50–17:15</u> **Pavlo Biziuk**
MODELLING OF POWER DISTRIBUTION CONTROL PROCESSES IN THE CORES OF VVER-1000 REACTORS USING ANC-H REACTOR CORE SIMULATOR PROGRAM (ARCS), Pavlo Biziuk[1,] [1]*Branch "SS "Scientific and Technical Center "JSC "NNEGC "Energoatom", Ukraine*

<u>17:15–17:40</u> **Henri Rapeli**
LICENSING OF A NEW FUEL DESIGN AT LOVIISA NPP, Henri Rapeli[1]*, Olli Hyvönen[1], [1]*Fortum Power and Heat Ltd, Fortum, Finland*

<u>17:40–18:05</u> **Antonios Mylonakis**
THE POLCA8H NODAL CODE FOR VVER-1000, Antonios Mylonakis[1], Carl Adamsson[1], Sten-Örjan Lindahl[1], [1]*Westinghouse Electric Sweden AB, Västerås, Sweden*

<u>Day 3: Tuesday, 16 September 2025</u>

08:30 – 10:10 Session 3 Improvement of fuel design and operation

Chairpersons: ***Didier Bardel, Nicolas Vioujard***

<u>08:30–08:55</u> **Didier Bardel**
ADDITIVE MANUFACTURING OF FUEL COMPONENTS: FROM IMPLEMENTATION OF NON-CRITICAL 316L COMPONENTS TO HEAVILY LOADED APPLICATIONS WITH ALLOY 718,
D. Bardel[1]*, G. Badinier[1], E. Schweitzer[2], K. Sohn[2], N. Schuyler[3], C. Wiltz[3], A. Dufresne[1], [1]*Framatome SAS, Lyon, France,* [2]*Framatome GmbH, Germany,* [3]*Framatome Inc., Richland WA, United States*

<u>08:55-09:20</u> **Gwen Bolsee**
FRAMATOME FUEL PORTFOLIO FOR LIGHT WATER BASED SMRs, Alice Dufresne[1], David Breeding[2], Elisa Calvo Tone[2], [1]*Framatome SAS, Fuel, Lyon – France,* [2]*Framatome Inc, Fuel, Lynchburg VA – United States*

09:20-09:45 **Romain Borrossi**
PERFORMANCE OF $M5_{FRAMATOME}$ CLADDING DURING LOSS-OF-COOLANT ACCIDENT, Romain BORROSSI[1]*, Raphaël CHOSSON[1], Xavier HALLER[1], [1]*Framatome, Lyon, France*

09:45-10:10 **Romain Borrossi**
PERFORMANCE OF $M5_{FRAMATOME}$ FUEL ROD DURING REACTIVITY INITIATED ACCIDENT INFORMED BY ADVANCED MODELING, Romain BORROSSI[1]*, Raphaël CHOSSON[1], Xavier HALLER[1], [1]*Framatome, Lyon, France*

10:10- 10:30 **Coffee Break**

10:30 – 12:35 **Session 3: Improvement of fuel design and operation (Cont.)**

Chairpersons: ***Didier Bardel, Nicolas Vioujard***

10:30–10:55 **Romain Borrossi**
FRAMATOME’S FUEL ROD AND FUEL ASSEMBLY STRUCTURAL MATERIALS FOR PWR AND VVER REACTORS, R. Borrossi[1], C. Forot[1], P. Guillermier[1], P.-B. Hoffmann[2], E. W. Schweitzer[2], [1]*Framatome SAS, Lyon, France,* [2]*Framatome GmbH, Erlangen, Germany*

10:55–11:20 **Dounia Lhachemi**
DEVELOPMENT OF EUROPEAN VVER-1000 FUEL ASSEMBLY BY FRAMATOME: PROJECT OBJECTIVES AND OVERVIEW OF THE DEVELOPMENT APPROACH, Dounia Lhachemi[1]*, Marco Messingschlager[2], [1]*Lyon – FRANCE,* [2]*Erlangen – Germany*

11:20–11:45 **Guy Gentet**
DEVELOPMENT OF EUROPEAN VVER-440 FUEL ASSEMBLY BY FRAMATOME, Guy Gentet[1]*, Pascal Blanc[2], [1]*Lyon – France,* [2]*Erlangen – Germany*

11:45–12:10 **Guy Gentet**

FRAMATOME SOLUTIONS TO SUPPORT UTILITIES FOR LONG TERM OPERATION, Guy Gentet[1]*, Lars Ackermann [2], [1] *rue Professeur Jean Bernard, 69007 Lyon – France,* [2] *Paul-Gossen-Str. 100, 91052 Erlangen – GERMANY*

12:10–12:35 **Guy Gentet**

FUEL RELIABILITY IMPROVEMENTS, Guy Gentet[1]*, Pierre-Henri Louf[1], Christophe Petit[1], [1]*Framatome SAS, Lyon, France*

12:35 – 14:00 **Lunch Break**

14:00 – 15:40 **Session 3: Improvement of fuel design and operation (Cont.)**

Chairpersons: ***Jan Höglund, Dounia Lhachemi***

14:00–14:25 **Karl Buchanan**

PROTECT CR: THE LEADING E-ATF SOLUTION BY FRAMATOME,K. Buchanan[1]*, T. Garnier[1], E. W. Schweitzer[2], N.Vioujard[1], [1]*Framatome SAS, Lyon, France,* [2]*Framatome GmbH, Erlangen, Germany*

14:25–14:50 **Nicolas Vioujard**

DEVELOPMENT OF INNOVATIVE MATERIAL FOR LWR FUEL, N. Vioujard [1]*, [1]*Framatome SAS, Lyon, France*

14:50 – 15:15 **Elmar W. Schweitzer**

IRRADIATION FEEDBACK OF FRAMATOME'S PROTECT-CR CLADDING TO A BURNUP OF >70 MWD/KGU, E. W. Schweitzer[1]*, T. Garnier[2], K. Buchanan[2], M. Aumand[3], [1]*Framatome GmbH, Erlangen, Germany,* [2]*Framatome SAS, Lyon, France,* [3]*Framatome Inc., Framatome Inc. Lynchburg,VA, USA*

15:15–15:40 **Jan Höglund**

MITIGATION OF LOCAL POWER PEAKING BY INTRODUCING HF SPACER PINS IN THE VVER-440 FOLLOWER FUEL ASSEMBLY, Akif Abdullayev[1]*, Douglas Hake[2], Jan Höglund[1], [1]*Westinghouse Electric Sweden AB, Västerås, Sweden,* [2]*Westinghouse Electric Company, PA, USA*

15:40- 16:05 **Coffee Break**

16:05 - 17:45 **Session 3: Improvement of fuel design and operation (Cont.)**

Chairpersons: ***Jan Höglund, Dounia Lhachemi***

16:05–16:30 **Alex Riznychenko**

NEXT GENERATION VVER-1000 FUEL DEVELOPMENT, Alex Riznychenko [1], Uffe Bergmann [1], Jan Höglund [1], [1]*Westinghouse Electric Sweden AB, Västerås, Sweden*

16:30–16:55 **Gwen Bolsee**

FRAMATOME FUEL RESEARCH CENTERS & LABS, G. Bolsee[1], V. Rebeyrolle[1]*, O. Calonne[2], D. Blavius[3], B. Stepnik[4], [1]*Framatome SAS, Lyon, France*, [2]*Framatome, France*, [3]*Framatome CERCA, France*, [4]*Framatome GmbH, Germany*

16:55–17:20 **Oleksandr Mazurok**

APPROACHES TO SUPPLY UKRAINIAN NPPS WITH CORE COMPONENTS, Valeriy Zuyok*[1], Oleksandr Mazurok[2], Oleg Zelenyy[3], Sergiy Lavrov[4], Oleg Godun[3], Anton Makarenko[5], [1]*National Science Center "Kharkov Institute of Physics and Technology", Kharkiv,* [2]*ES Group LLC, Kyiv,* [3]*Scientific and Technical Center, JSC "NNEGC "Energoatom", Kyiv,* [4]*Atomenergomash, JSC "NNEGC "Energoatom", Kyiv,* [5]*Scientific and Technical Complex "E.O. Paton Electric Welding Institute", Kyiv, Ukraine*

17:20–17:45 **Angel Demerdjiev**

RADIATION SHIELDING ASSESSMENT OF THE CYCLOTRON CENTRE AT INRNE-BAS, A. Demerdjiev[1]*, D. Tonev[1], G. Dimitrova[1], N. Goutev[1], E. Geleva[1], V. Variyska[1], [1]*Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, Bulgaria*

Day 4: Wednesday, 17 September 2025

08:30 – 10:10 Session 4: Fuel modelling and experimental support

Chairpersons: ***Arndt Schubert, Stanislav Linhart***

08:30–08:55 **Arndt Schubert**
TOWARDS MODELLING OF ADVANCED-TECHNOLOGY FUEL RODS WITH THE TRANSURANUS CODE, A. Schubert*[1], I. M. Paponetti[2], P. Aragón[1], D. Jaramillo Sierra[1], Zs. Soti[1], P. Van Uffelen[1], *[1]European Commission – Joint Research Centre, Karlsruhe, Germany, [2]University of Bologna, Italy*

08:55–09:20 **Johann Plancher**
GALILEE, FRAMATOME FUEL ROD PERFORMANCE CODE FOR LWR, Y. Wrtal[1]*, J. Plancher[1], G. Gentet[1], M. Brunmayr[2], T. Distler[2], O. Wieckhorst[2], *[1]Framatome SAS, Lyon, France*, *[2]Framatome GmbH, Erlangen, Germany*

09:20–09:45 **Veit Marx**
FRAMATOME ADVANCED SOLUTIONS AGAINST GRID-TO-ROD FRETTING WEAR RISK, Julien Pacull[1]*, Veit Marx[2], *[1]Framatome SAS, Lyon, France, [2]Framatome GmbH, Erlangen, Germany*

09:45–10:10 **Veit Marx**
FRAMATOME'S ADVANCED CORE DISTORTION SIMULATION CAPABILITIES, Veit Marx[1], *[1]Framatome GmbH, Erlangen- Germany*
Knotek J.[1], Pašta O[1]., [1]Centrum výzkumu Řež s.r.o., Czech Republic

10:10- 10:30 **Coffee Break**

10:30 – 12:35 Session 4: Fuel modelling and experimental support (Cont.)

Chairpersons: ***Arndt Schubert, Stanislav Linhart***

10:30–10:55 **Tommaso Carnicella**
FUEL ASSEMBLY BOW DNBR PENALTY FOR VVER-1000 REACTORS, Tommaso Carnicella [1], Akif Abdullayev [1], Ulf Lindelöw[1], [1]*Westinghouse Electric Company, Sweden, Västerå*

10:55–11:20 **Stanislav Linhart**
IRRADIATION-INDUCED GROWTH OF ADVANCED ZIRCONIUM ALLOYS, Stanislav Linhart[1]*, Josef Běláč[1], Radomír Řeháček[1], Alexander Y. Shevyakov [2], Anatoly Y. Gusev [2], Pavel Pešek[3], [1]*ALVEL, a.s., Brno, Czech Republic,* [2]*SC VNIINM, Moscow, Russian Federation,* [3]*ÚJV Řež, Husinec-Řež, Czech Republic*

11:20–11:45 **Michael Brunmayr**
FRAMATOME – IMPROVEMENT OF FUEL PRODUCT TH PERFORMANCE, Peter Pohl[1]*, M. Rehm[1], K. Marcinkiewicz[1], H. Gabriel[1], Fabien Filhol[2], [1]*Framatome GmbH, Erlangen, Germany,* [2]*Framtome SAS, Lyon, France*

11:45–12:10 **Michael Brunmayr**
VALIDATION OF FRAMATOME SUBCHANNEL CODE COBRA-FLX FOR VVER FUEL, Michael Brunmayr[1]*, Mira Pashtrapanska[1], Harry Gabriel[1], [1]*Framatome GmbH, Erlangen, Germany*

12:10–12:35 **Marcin Kopeć**
DIGITAL IMAGE PROCESSING FROM NUCLEAR FUEL INSPECTIONS IN CVR, Kopeć M[1]., Blažek J.[1], Knotek J.[1], Pašta O[1]., [1]*Centrum výzkumu Řež s.r.o., Czech Republic*

12:35 – 14:00 **Lunch break**

14:00 – 16:05 Session 4 Fuel modelling and experimental support (Cont.)

Chairpersons: ***Emiliya Georgieva, Katalin Kulacsy***

<u>14:00–14:25</u> Jürgen Michna

INSPECTIONS AND MEASUREMENTS ON FUEL ASSEMBLIES, Ch. WENDLAND[1], J. MICHNA[1]*, S. GARNIER[2], A. GUEDICHI[2], V. DUNN [3], J. OLIVER[3], *[1]Framatome GmbH, Erlangen – Germany, [2]Framatome SAS, Lyon – France, [3]Framatome Inc., LYNCHBURG VA – USA*

<u>14:25–14:50</u> Emiliya L. Georgieva

CMS5 BENCHMARKING AGAINST A VVER-1000 PLANT DATA, Emiliya L. Georgieva[1]* and Tamer Bahadir[2], *[1]Studsvik Scandpower GmbH, Norderstedt, Germany, [2]Studsvik Scandpower Inc., Idaho Falls, ID, USA*

<u>14:50–15:15</u> Fredrik Waldemarsson

WESTINGHOUSE FUEL THERMAL-HYDRAULIC TESTING CAPABILITIES**,** Fredrik Waldemarsson[1]*, Carrie Wood[2]**,** *[1]Westinghouse Electric Sweden***,** *[2]Westinghouse Electric Company*

<u>15:15- 15:40</u> Coffee Break

<u>15:40–16:05</u> Nadejda Rijova

ANALYSIS OF MAIN STEAM LINE BREAK AT HOT ZERO POWER USING ATHLET, COCOSYS and DYN3D, Nadejda Rijova[1]*, Alexander Yordanov[1], *[1]Enpro Consult, Sofia, Bulgaria*

16:05 – 16:55 Session 5: Fuel safety and QA

Chairpersons: ***Emiliya Georgieva, Katalin Kulacsy***

16:05–16:30 **Katalin Kulacsy**

FUEL PERFORMANCE ANALYSIS CAPABILITY DEVELOPMENT IN SURO, Katalin Kulacsy[1], *[1]National Radiation Protection Institute (SÚRO),Czech Republic*

16:30–16:55 **Maximilian Stark**

DEVELOPMENT OF VVER-1000 FUEL ASSEMBLY SHIPPING SOLUTION BY FRAMATOME: PROJECT OBJECTIVES AND STATUS, Dounia Lhachemi[1], Maximilian Stark, Johannes Von Morstein[2], *[1]Lyon, France, [2]Lingen –Germany*

19:00- 23:00 **SOCIAL DINNER**

Day 5: Thursday, 18 September 2025

09:00 – 10:15 Session 6: Spent fuel performance and management

Chairpersons: ***Roman Glushenkov, Radan Sedláček***

09:00–09:25 **Radan Sedláček**
FRAMATOME'S PERFORMANCE IN SPIZWURZ CREEP BENCHMARK, Radan Sedláček[1]*, Dietmar Deuble [1], [1]*Framatome GmbH, Erlangen,Germany*

09:25–09:50 **Roman Glushenkov**
SPENT NUCLEAR FUEL MANAGEMENT OF UKRAINIAN NUCLEAR OPERATOR, Roman Glushenkov[1], [1]*JSC "NNEGC "Energoatom", Kyiv, Ukraine*

09:50–10:15 **Mykhaylo Tretyakov**
STUDYING THE INTEGRITY OF SNF ROD CLADDING AT VARIOUS STAGES OF DRY STORAGE TECHNOLOGY IMPLEMENTATION,Valeriy Zuyok[1]*, Mykhaylo Tretyakov[1] , Roman Rud[1] , Volodymyr Zigunov[1], Yana Kushtym[1] , Vadym Hrudnytskyy [1], [1] *National Science Center "Kharkiv Institute of Physics and Technology"of the NAS of Ukraine, Kharkiv, Ukraine*

09:50–10:15 **Valeriy Zuyok**
DEVELOPMENT OF NEUTRON-ABSORBING MATERIALS AND RCCA WITH INCREASED PERFORMANCE FOR VVER-1000 REACTOR**,** Volodymyr Krasnorutskyy[1], Viktor Grytsyna[1], Ihor Chernov[1], Volodymyr Zigunov[1], Valeriy Zuyok[1], Anton Kushtym[1]**,** [1]*The Science and Technical Establishment "Nuclear Fuel Cycle" of National Science Center Kharkov Institute of Physics and Technology, Ukraine*

10:15–10:40 **Conference overall discussion and conclusionss**

12:00 – 13:30 **Lunch Break**

14:00 – 18:00 **Excursion**

Day 6: Friday, 19 September 2025

09:00 – 10:30 Poster Session

1. **Vítězslav Matocha**
EXTENDED VALIDATION OF TRANSURANUS CODE FOR THE PWR CLADDING TYPES CONDUCTED AT UJV REZ, Vítězslav Matocha[1], Martin Dostál[1], Jan Klouzal[1], *[1]ÚJV Řež, a. s., Czech Republic*

2. **Ondrej Pašta**
ADVANCED DIGITAL IMAGE PROCESSING FOR MATERIAL EVALUATION IN NUCLEAR APPLICATIONS, Pašta O[1]., Knotek J.[1], Blažek J.[1], Kopeć M., *[1]Centrum výzkumu Řež s.r.o., Czech Republic*

3. **Mladen Mitev**
FROM OPEN TO CLOSED FUEL CYCLE – ADVANTAGES AND DRAWBACKS OF THE PROCESS, Mladen Mitev [1,2]*, Kalin Filipov[1], *[1]Technical University – Sofia*, *[2]Institute for Nuclear Research and Nuclear Energy of the Bulgarian Academy of Sciences*

4. **Petya Vryashkova**
STUDY OF FUEL BEHAVIOUR DURING LBLOCA WITH PARTIAL DAMAGING OF REACTOR CORE, Petya Vryashkova*[1], Petya Petrova[1], Pavlin Groudev[1], *[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences*

5. **Galina Dimitrova**
MONTE CARLO-BASED STUDIES FOR RADIATION SAFETY IN THE DESIGN OG BGNCC, G.D. Dimitrova[1]*, A. Demerdjiev[1], D. Tonev[1], N. Goutev[1], E. Geleva[1], V. Variyska[1], N.N. Petrov[1], B. Bonchev[1], *[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences*

6. **Elena Geleva**
POST-FUKUSHIMA STRESS TESTS AT KOZLODUY NPP: ASSESSING RESILIENCE TO EXTREME CLIMATIC EVENTS, D. Tonev[1], E. Geleva[1]*, N. Goutev[1], A. Demerdjiev[1], G. Dimitrova[1], V. Variyska[1], N.N. Petrov[1], B. Bonchev[1],*[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences*

ABSTRACTS

SESSION 1: PLENARY SESSION

SCIENCE AND THE NUCLEAR INDUSTRY IN BULGARIA

D. Tonev[1]*, M. Manolova[1], A. Demerdjiev[1], E. Geleva[1], G.D. Dimitrova[1], V. Varijska[1], P. Vryashkova[1], D.M. Dimitrova[1], L. Nedelchev[1], Tz.K. Marinov[1], L. Pantaleev-Simeonova[1], V.N. Pavlova[1], N.N. Petrov[1], N. Goutev[1], B. Bonchev[1], M. Mitev[1,2]

[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, 72 Tsarigradsko chaussee Blvd, 1784 Sofia, Bulgaria
[2]Technical University of Sofia, 8 Kliment Ohridski Blvd, 1000 Sofia, Bulgaria
**E-mail of the Corresponding Author: dimitar.tonev@inrne.bas.bg*

Abstract

The Institute for Nuclear Research and Nuclear Energy at the Bulgarian Academy of Sciences was among the first organizations in Bulgaria to initiate activities in the nuclear field, both in the country and in South-East Europe. In Bulgaria, the development has followed a characteristic path: beginning with fundamental science, and then applying the acquired knowledge, experience, and expertise to advance activities in the nuclear domain—namely nuclear energy, nuclear medicine, radiation protection, and radioactive waste management.

As a result, world-class scientific schools were established in reactor physics, neutron physics, accelerator physics, experimental and theoretical nuclear physics, and detector development. Over time, new interdisciplinary fields also emerged, such as radiochemistry, radiopharmacy, and radiobiology.

The key milestone for Bulgaria in the nuclear field was the commissioning of the IRT-2000 research reactor in 1961. The first

chain reaction at INRNE-BAS placed Bulgaria on the map of nuclear nations. This achievement laid the foundation for the safe operation of six units at the Kozloduy Nuclear Power Plant.
At present, Bulgaria operates two WWER-1000 reactors, while a new challenge for the nuclear community is the introduction of nuclear fuel from different suppliers. Looking forward, Bulgaria maintains its nuclear ambitions, with plans to construct and operate two additional units at Kozloduy. Scientific organizations will play a pivotal role in supporting these developments, particularly in the education and training of a new generation of nuclear engineers and experts.

Currently, a central challenge for INRNE-BAS is the commissioning of the TR-24 cyclotron facility. Meeting this and other strategic goals will ensure Bulgaria's continued progress in its modern nuclear era. The activities of Bulgarian scientific organizations in close collaboration with the nuclear industry will be presented.

Acknowledgment: This research was funded by the Ministry of Education and Science of the Republic of Bulgaria, through the National Program D01-99: Qualification improvement in the field of nuclear technologies and nuclear engineering.

Keywords: Research reactor; Nuclear fuel; TR-24 Cyclotron.

IMPROVING NUCLEAR ECONOMICS: TOWARDS LEU+ AND HIGH BURNUP FUEL FOR LIGHT WATER REACTOR APPLICATIONS

Gwen BOLSEE[1]*, Steven COLE[2], Norm GARNER[2]

[1]Framatome SAS, rue Professeur Jean Bernard, 69007 Lyon – FRANCE
[2]Framatome Inc., 2101 Horn Rapids Road, Richland, WA 99354, USA

E-mail of the Corresponding Author: gwen.bolsee@framatome.com

Abstract

Access to fuel assemblies containing uranium enriched above 5 wt% 235U and extending peak rod licensed burnup levels above 62 GWd/MTU has been identified by US reactor operators and fuel fabricators as a significant means of economically extracting more energy from the existing LWR fleet.

Enabling access to uranium enriched to a level between 5 and 10 wt% 235U, referred to as LEU+ material, requires significant upgrades to fuel fabrication processing lines to assure satisfactory margin of subcriticality will be maintained. Extending licensed burnup levels requires acquisition of benchmarking data and use of this data to extend the modelling of fuel into a higher burnup range and securing regulatory approval for these methods.

This presentation will provide an overview of the scope and status of Framatome's project, which is on track to meet its primary objective to be ready to supply reloads of LEU+ / high burn-up fuel assemblies from 2027.

Keywords: LEU+, High burnup, Economics

EXPERIENCE OF RWFA FUEL IMPLEMENTATION AT KOZLODUY NPP UNIT 5

K. Kamenov[1]*, D. Hristov[1], A. Kamenov[1], S. Kolev[1],
M. Milchev[1], A. Avramov[1], T. Nikolova[1], V. Naev[1]
[1]Kozloduy NPP, Bulgaria
E-mails: K_Kamenov@npp.bg; DVHristov@npp.bg;
ALKamenov@npp.bg; STKolev@npp.bg;
MSMilchev@npp.bg; APAvramov@npp.bg; TVNikolova@npp.bg;
VINaev@no-mail.kz

Abstract

The loading of Westinghouse RWFA VVER-1000 fuel in Kozloduy NPP and start of mixed core operation with RWFA and TVSA fuel, marks the completion of the qualification and licensing process for RWFA in Bulgaria.

The advanced RWFA assembly distinguishes itself with increased uranium content, enrichment up to 4.7wt% and using of 12 Gd fuel rods. This enables the safety and effectiveness of the fuel cycles to be increased.

After successful completion of the feasibility study and full licensing process of RWFA for unit 5, on April 22nd BNRA issued a license for RWFA implementation on Unit 5, and in May 2024 the first 43 fresh RWFA were loaded in a mixed core with the resident TVSA for the cycle 31.

This paper contains information about the experience of the first and second RWFA fuel batch loadings at the Kozloduy NPP Unit 5. Neutron-physics characteristics for cycles 31st and 32nd of Unit 5, calculated by APA-H and HELHEX code packages have been compared with relevant measured/reconstructed data.

A very good agreement and acceptable differences of the results, both between the measured and calculated and between the calculated ones are demonstrated.

SESSION 2: FUEL PERFORMANCE AND OPERATIONAL EXPERIENCE

FRAMATOME UNIQUE OPERATING EXPERIENCE ON 14FT PRODUCTS

N. Lamorte[1]*, G. Bolsee[1]
[1]Framatome, 2 rue du Professeur Jean-Bernard 69007 LYON, France
E-mail of the Corresponding Author: nicolas.lamorte@framatome.com

Abstract

Fuel assemblies (FA) under light water reactor (LWR) operational conditions are subject to deformations during their life cycle. FA lateral deformations or bow result from numerous complex and coupled phenomena: creep under irradiation, hydraulic forces, axial stresses, reactor design, and operational history. Excessive bow can be detrimental towards the safe operation of nuclear reactors as it could ultimately cause incomplete control rod insertion (IRI) or lead to difficulties during core loading and unloading during outages. Consequently, preventing excessive FA bow is necessary to ensure a safe operation of these reactors. In particular, the 14ft FA are slender structures subjected to high flow environment and FA bow is the consequence of a highly multi-physics and multiscale phenomenology which requires understanding the fluid structure interaction over long time periods under irradiation.

In this context, assessing the complex phenomenology of FA distortion is significantly facilitated by the extensive and diverse operating experience gained by Framatome and its clients over several decades. This experience consists of a wide range of core designs and operational conditions. In parallel, Framatome's FA designs have been incrementally improved to maximize their performance regarding FA distortion and suppress any of its potential negative impact on plant operations. These evolutions are the result of Framatome's expertise in designing robust 14ft FA and assessing a unique range of operating experience.

In parallel, high-fidelity models have been developed and validated to support Framatome’s capability to secure and enhance FA product performance. This overall expertise opens the perspective to actively re-enforce our support to utilities in operability and/or risk assessment in case of FA distortion or IRI considerations by proposing advanced FA designs and, when needed, new efficient pre-emptive actions and short-term countermeasures.

Keywords: FA bow, FA distortion, Operating experience, FA performance

FUEL CYCLE OPTIMIZATION WITH FRAMATOME FARGO SOFTWARE

Johann Plancher[1]*, Greg Hobson[2], Hans-Wilhelm Bolloni[3]

[1]Framatome SAS, Lyon, France
[2]Framtome Inc., Lynchburg VA, USA
[3]Framatome GmbH, Erlangen, Germany
E-mail of the Corresponding Author: johann.plancher@framatome.com

Abstract

The task of designing a fuel loading pattern is determinant to the efficiency of the core. It is key to improving fuel cycle cost or for transition to longer cycles, for power uprate or for flexible operation. FARGO is a set of Framatome software tools for integrated fuel assembly and loading pattern optimization (LPO). Fuel assembly optimization relies on the APOLLO2-A physics solver combined with DAKOTA's Single Objective Genetic Algorithm (SOGA) and exhaustive search methods. Enrichment zoning and burnable poison placement is optimized with a user-defined objective function from the APOLLO2-A results. The optimization is performed in core regions which contain different patterns of fresh and burned fuel assemblies using either single or multiple assembly geometries. LPO is based on the core simulator with methods developed by NCSU as part of the MIDAS system, including Genetic Algorithm (GA), Parallel Simulated Annealing (PSA) and Reinforcement Learning (RL). Feedback from core calculations can be used to re-rank the lattice optimization results.

The use of FARGO enables Framatome designing performant fuel loading pattern either increasing design margin or increasing fuel cycle performance, therefore reducing fuel cycle cost.

Keywords: Nuclear Fuel, Loading Pattern Optimization

VVER WATER CHEMISTRY COMPATIBILITY WITH WESTINGHOUSE FUEL

Luis Gonzalez[1]*, Mathilde Gaillard[1], Lena Oliver[1], Britta Helmersson[1].
[1] *Westinghouse Electric Sweden, Bränslegatan 1, SE-72163 Västerås*

E-mail of the Corresponding Author:
luis.gonzalezfonseca@westinghouse.com

Abstract

The coolant chemistry can have a major impact on fuel performance and as such is important to keep within existing guidelines and experience. Significant experience exists for Westinghouse fuel and materials in Pressurized Water Reactors (PWR) but has also gained several operating years in Voda-Vodyanoi Energetichesky Reaktor (VVER) plants. VVERs are based on the same concept as PWRs. However, there are differences and similarities between VVER and PWR fuel operating chemistry conditions that must be evaluated to assess their impact on fuel performance, corrosion, and the risk of crud deposition.

Westinghouse has recently compiled and contrasted the differences between PWR and VVER chemistry control to assure the compatibility with Westinghouse fuels. The behavior of various control parameters in VVER-440 and VVER-1000 units has been compared with Westinghouse and EPRI guidelines in this study. As can be expected, a few minor differences exists in the guidance.

Regardless, the data showed that most control parameters have been measured within the range of Westinghouse operational experience for all the plants to which Westinghouse supplies with fuel.

Keywords: VVERs, PWRs, boiling duty, crud, fuel failure

WESTINGHOUSE LTA INSPECTION AT TEMELÍN NPP

Michal Kuna[1], Jan Höglund[2]

[1] *Westinghouse Electric Czech Republic, s.r.o., Czech Republic, 140 00 Prague*

[2] *Westinghouse Electric Sweden AB, Sweden, 721 63 Västerås*

E-mail of the Corresponding Author: michal.kuna@westinghouse.com

Abstract

From 2019 to 2022 six (6) LTAs were loaded in Temelin unit 1 together resident fuel design TVSA- T.mod.1. After three (3) cycles an LTA was found to be leaking. First visual examinations carried out in 2022 didn't show any anomalies. More detailed inspections were planned for the outage in year 2024.

This inspection included top nozzle removal and extraction of rods in order to perform single rod visual inspections, single rod eddy current measurements. In addition, UT search to find the leaking rod was planned. However, extraction of the leaking rod was prohibited by NPP management, so no single rod inspection of the leaking rod was planned. Thoughtful preparations were made to consider all possible risks and VVER-1000 specifics. In order to mitigate the risks an upgrade of Temelin's Fuel Repair and Inspection Equipment was performed, emergency procedures written and needed tools to deal with abnormal scenarios prepared. The most severe risk by fuel rod extraction is based on breaking rod with breakaway pellet(s).

Forty (40) rods were extracted from a non-leaking assembly for inspection. No grid to rod fretting marks were found. In the leaking assembly the leaking rod was discovered in the third row. The leaking rod didn't show increased growth or any other abnormal behavior although it has been leaking for a whole cycle.

Based on the visual and eddy current inspection of 40 rods grid to rod fretting has been ruled out as a cause of leakage but the root cause of the leakage remains unknown. Currently the same design of fuel assemblies is being delivered to Temelin NPP and it is scheduled

to be operated in 2026. Westinghouse recommended to repeat similar inspection after 2 or 3 18 months cycles.

Keywords: RWFA-T, fuel inspection, grid to rod fretting, Temelin NPP

FRAMATOME SOLUTIONS TO FLEXIBLE OPERATION

Johann Plancher[1]*, Alain Grossetete[1], Eric Corrieri[1], Lars Ackermann[2]

[1]Framatome SAS, Lyon, France
[2]Framtome GmbH, Erlangen, Germany
E-mail of the Corresponding Author: johann.plancher@framatome.com

Abstract

In many regions of the world, the constant increase of solar and wind power in the energy mix has impacted electricity generation. The electricity storage capacity is currently not sufficient to level the production from these inconstant energy sources. For the next decades

ahead, it is of interest for utilities to adapt the operation of nuclear power plant and optimize it: load follow, frequency control, adaptation to unexpected grid request, and extended low power operation.

Over the years Framatome has provided solutions to many utilities, especially in Europe, to provide their plants the needed operational flexibility. Framatome Fuel products (Fuel assemblies and Control rods) have demonstrated their ability to facilitate various operational modes.

Strong with this experience, Framatome has also further developed software solutions such as the ARGOS Core Monitoring System to facilitate flexible operation. ARGOS provides features such as the Predictor, to simulate and predict operational transient and therefore enables anticipating such scenario. Framatome also provides the OAPS software for automatic optimization of transient scenario including short term prediction. For example, the nuclear reactor operator can find the optimum balance between the use of control rod insertion and the use of boron injection/dilution in order to minimize the impact on core parameters (xenon transient) and minimize the cost of transient (transient duration and use of boron). OAPS system also provides optimal real-time recommendations during load-following transients. Human Factors Engineering (HFE) studies have demonstrated a reduction in operator workload as a result of these functionalities, which also address an emerging need for instant response to grid requests.

ARGOS and OAPS are trademarks or registered trademarks of Framatome or its affiliates, in the USA or other countries.

Keywords: Flexible Operation, Load Follow, Core Monitoring System

EVOLUTION OF FRAMATOME CORE SIMULATORS

Johann Plancher[1]*, Philippe Bellanger[2], Bettina Hartmann[3]

[1]*Framatome SAS, Lyon, France*
[2]*Framatome Inc., Lynchburg VA, USA*
[3]*Framatome GmbH, Erlangen, Germany*
E-mail of the Corresponding Author: johann.plancher@framatome.com

Abstract

Over the years, Framatome has developed, licensed and implemented full suites of modern codes and methodologies to maximize design margins for a wide range of Pressurized Water Reactor as well as Boiling Water Reactor.

In France, the well-established SCIENCE code suite covers the full EDF fleet of nuclear reactors. It benefits from the extensive use over more than 30 years for various fuel products, core designs and operational mode, including flexible operation.

For the export market such as the USA and other customers in Europe, Framatome has now been using for some years the ARCADIA code suite. The package consists of five codes (APOLLO2-A, ARTEMIS, COBRA-FLX, GALILEO, and S-RELAP5) and five evaluation models for analysis of fuel rod thermal-mechanical behavior, Small Break Loss-Of-Coolant-Accidents (SBLOCA), Realistic Large-Break LOCA (RLBLOCA), Rod Ejection Accident (AREA), and Transient non-LOCA (ARITA). The combination of these advanced codes and methods releases design margin for the customer facilitating potential power uprate or flexible operation.

Framatome is also preparing the future. Together, Framatome and EDF, have been developing the ODYSSEE code chain which is being reviewed by the ASNR in France. ODYSSEE combines the experience and know-how from both companies as well as benefits from the expertise in R&D of both companies.

Keywords: Advanced Codes and Methods

MODELLING OF POWER DISTRIBUTION CONTROL PROCESSES IN THE CORES OF VVER-1000 REACTORS USING ANC-H REACTOR CORE SIMULATOR PROGRAM (ARCS)

Pavlo Biziuk[1]*
[1]*Branch "SS "Scientific and Technical Center "*JSC "NNEGC "Energoatom"
E-mail of the Corresponding Author : p.biziuk@direkcy.atom.gov.ua

Abstract

Due to the conversion of Ukrainian NPPs to Westinghouse fuel and the difficult situation in the Ukrainian power system characterized by periodic shelling of critical infrastructure by the Russian Federation, it became necessary to quickly unload NPPs during shelling and, accordingly, to issue a power unit unloading forecast to the control room operational personnel taking into account current restrictions on nuclear fuel operation.

For TVEL fuel, the IR program (Reactor Simulator developed by RSC KI, Moscow) was used at some NPPs to issue a forecast. At present, when switching to Westinghouse fuel assemblies, the above program cannot work due to the absence of constants for this fuel.

For these purposes, a graphical program was developed for computational modelling and forecasting of power management processes in VVER core using the ANC-H code - ARCS (ANC-H Reactor Core Simulator)

The ARCS program is designed for computational modelling of power control processes in VVER-1000 core using the ANC-H code. The ARCS software is a graphical user interface that allows performing control actions (power change, position of the CR etc.) in a convenient mode and performing neutronic calculations forecast using the ANC-H code and displaying information in text and graphical form. The ARCS program is similar in functionality to the IR program.

Validation calculations of transient processes for Ukrainian NPPs using the ARCS program showed that the maximum value of the RMS for axial offset is no more than 2.4%, and the maximum value of the RMS for boric acid concentration is no more than 0.2 g/kg, which is a good convergence of the calculated data with the experimental values and confirms the correctness of the selected approaches to modelling transient processes and the correctness of the input data formation and display of the calculation results.

Thus, the ARCS program can be used to perform a computational forecast of the state of VVER-1000 cores both for fuel assemblies consisting of fuel assemblies from JSC TVEL, mixed fuel assemblies TVS-A and TVS-W/WR (transient fuel assemblies) and for fuel assemblies consisting entirely of fuel assemblies from Westinghouse.

Keywords: VVER-1000, power distribution

LICENSING OF A NEW FUEL DESIGN AT LOVIISA NPP

Henri Rapeli[1]*, Olli Hyvönen[1]

[1]Fortum Power and Heat Ltd
POB 100, FI-00048 Fortum, Finland
E-mail of the Corresponding Author: henri.rapeli@fortum.com

Abstract

Fortum prepared for the possible extension of Loviisa NPP operating lifetime beyond 2030 in 2020-2021. In February 2023, the Finnish Government granted a new operating licence until the end of 2050. In November 2022, Fortum signed an agreement with Westinghouse Electric Company (WSE) for the design, licensing and supply of a new fuel type for Loviisa NPP. The new fuel type is called NOVA E-6 (fixed fuel assembly) and NOVCD (fuel follower). WSE's fuel design is based on BNFL's NOVA E-3 and NOVCA fuel design which was used in Loviisa in the early 2000s.

During the years 2022-2025, Fortum assessed and had the new supplier and fuel type approved by the authorities. As a part of the approval process, a dummy assembly (stainless steel bars instead of uranium pellets) was manufactured and loaded into Unit 2 (2023 – 2024) and Unit 1 (2024-2025). The dummy assembly replaced one shielding assembly in the reactor periphery and it was used to obtain mechanical operating experiences of the new fuel type.

The dummy assembly was visually inspected after the first cycle with Loviisa NPP's poolside inspection equipment (ATULA) and will be inspected again after the second cycle. Feasibility study of WSE fuel was approved by STUK in summer 2024. Seismic analyses of WSE fuel were completed in 2025. WSE LTAs (fixed fuel assemblies and follower fuel assemblies) were loaded into Unit 2 in 2024 and almost full reload batch is planned to be loaded into Unit 2 reactor in 2025 fuel outage. Fortum plans on inspecting one or several fuel assemblies after three and four irradiation cycles.

Fortum has also participated in the APIS program (Accelerated Program for Implementation of secure VVER fuel Supply) which is

co-funded by the European Union and led by Westinghouse. Within APIS program, Fortum inspected irradiated NOVA E-3 fuel assemblies to gain insight on its performance, inspected the WSE dummy assembly, performed CFD analyses to assess the PLC-performance of WSE's grid designs and assessed the economic feasibility of WSE's ADOPT-pellets in VVER-440 fuel.

Acknowledgment: Westinghouse, APIS Program

Keywords: VVER-440, NOVA E-6, NOVCD, licensing, APIS

THE POLCA8H NODAL CODE FOR VVER-1000

Antonios Mylonakis[1]*, Carl Adamsson[1], Sten-Örjan Lindahl[1]

[1]Westinghouse Electric Sweden AB 721 63 Västerås, Sweden
E-mail of the Corresponding Author:
antonios.mylonakis@westinghouse.com

Abstract

Westinghouse Electric Sweden introduces POLCA8H 1.0, a new nodal code for VVER-1000 reactors, with planned extension to VVER-440 cores. The code solves the neutron diffusion equation in hexagonal nodes using the Fourier expansion method. Each hexagonal assembly is modelled either as one column of hexagonal nodes or as six columns of triangular ones. For treating axial heterogeneities, on-the-fly axial homogenization is performed. The cross-section representation model accounts for all significant physical phenomena influencing reactor behaviour. A coupled thermal-hydraulics module computes the coolant density and temperature distributions for each assembly by solving the conservation equations for mass, momentum, and energy. The distribution of coolant in the core is estimated in an iterative manner from the requirement that the pressure drop over all channels must be equal in a core steady state. Pin power reconstruction is carried out using the conventional two-step approach, where the nodal flux solution is combined with precomputed pin power form factors obtained from lattice physics calculations.

The code is implemented in modern Fortran and parallelized using OpenMP to ensure computational efficiency on shared-memory architectures. POLCA8H is intended to be used for steady-state design and licensing applications as well as for the class of safety calculations that can be performed with steady-state methods. Selected initial V&V results are presented, providing encouraging evidence of the code's accuracy and performance in representative reactor physics applications.

Keywords: nodal code, VVER-1000, neutron diffusion

SESSION 3: IMPROVEMENT OF FUEL DESIGN AND OPERATION

ADDITIVE MANUFACTURING OF FUEL COMPONENTS: FROM IMPLEMENTATION OF NON-CRITICAL 316L COMPONENTS TO HEAVILY LOADED APPLICATIONS WITH ALLOY 718

D. Bardel[1]*, G. Badinier[1], E. Schweitzer[2], K. Sohn[2], N. Schuyler[3], C. Wiltz[3], A. Dufresne[1]

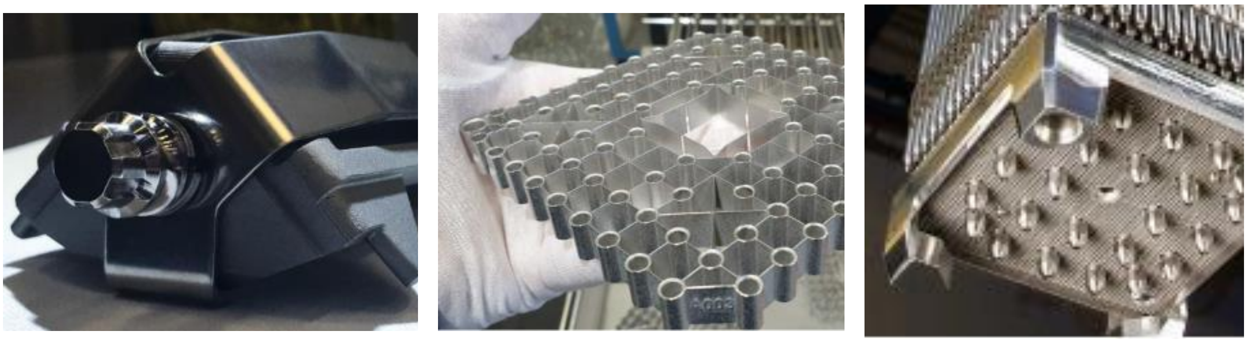
[1] Framatome SAS, 2 rue Professeur Jean Bernard, 69007 Lyon, France
[2] Framatome GmbH, Karlstein Am Main, Germany
[3] Framatome Inc., 2101 Horn Rapids Road 99354 Richland WA, United States
E-mail of the Corresponding Author: didier.bardel@framatome.com

Abstract

Additive manufacturing (AM) is becoming an increasingly accepted manufacturing route in the nuclear industry because it allows for more efficient designs and can lead to significant gains in the global supply chain compared to conventional manufacturing routes. In this framework, Framatome has established a global roadmap for the deployment of AM for Light Water Reactors (LWR) fuel components, which aims at tackling key challenges such as material/process qualification, in-service material behavior, component design and production at the industrial scale. In this presentation, a specific focus will be made on the Laser Powder Bed Fusion (L-PBF) technique with 316L stainless steel and alloy 718, as this process and those two materials are a perfect match for advanced-design nuclear fuel components.

First, a global overview of the opportunities for AM for nuclear fuel components is presented. Then, a description of some 316L

stainless steel products will be highlighted to show that, despite the exotic microstructures inherent of the L-PBF technique, the overall materials results are well on target. This made possible successful implementation of lead components in commercial cores. Technical data such as out and in-pile tensile results obtained on this alloy will be put forward to briefly discuss AM behavior compared to a conventional 316L material.



a) b) c)

Figure 1: Pictures of lead components implemented in LWR commercial reactors. a) channel fastener, b) upper tie plate, and c) anti-debris filter.

Heavily loaded nuclear fuel components must be made with stronger material, e.g. alloy 718. However, one must also ensure that critical material performances such as the resistance to stress-corrosion cracking (SCC) are at the right level for those applications. In that regard first results obtained on L-PBF alloy 718 samples show that SCC resistance is at least similar to conventional alloy 718, opening the way to future applications of the L-PBF technique for critical nuclear fuel components.

Keywords: Material irradiation, Additive Manufacturing, Stress Corrosion Cracking, Advanced design, Qualification

FRAMATOME FUEL PORTFOLIO FOR LIGHT WATER BASED SMRs

Alice Dufresne[1]*, David Breeding[2], Elisa Calvo Tone[2]

[1]Framatome SAS, Fuel, Lyon – France
[2]Framatome Inc, Fuel, Lynchburg VA – United States
E-mail of the Corresponding Author: alice.dufresne@framatome.com

Abstract

In the context of 31 countries' commitment to triple the installed nuclear capacity, the deployment of Light Water Small Modular Reactors (SMRs) is expected to play a key role in decarbonizing not only electricity production but also hard-to-abate sectors needing low temperature heat. The new reactor developments are bolstering innovation in the nuclear industry. As a major fuel designer and manufacturer, Framatome is leveraging its 60+ years of expertise in fuel design, development, fabrication, and licensing, to develop fuel portfolio solutions for SMRs.

This paper describes Framatome's development of the fuel portfolio for light water-based SMRs. Light water SMRs indeed benefit from their familiarity with the existing nuclear industry while encapsulating innovations in their designs and targeted market segments. Framatome supports the ambitious SMRs timelines thanks to fast-track development and industrialization of fuel assemblies and core components with a portfolio of solutions to adapt fuel products to address specific light water based SMRs requirements.

Framatome leverages its readily available technological bricks from existing qualified and licensed fuel designs to secure time to market while maximizing added value. Framatome is also engaged with SMR designers to propose and develop specific value adding features to the fuel products life cycle and reactor economics.

Keywords: SMR, fuel solutions.

PERFORMANCE OF M5FRAMATOME CLADDING DURING LOSS-OF-COOLANT ACCIDENT

Romain Borrossi[1]*, Raphaël Chosson[1], Xavier Haller[1]

[1]*Framatome, 2 rue Professeur Jean Bernard, 6900, Lyon, France*
E-mail of the Corresponding Author: romain.borrossi@framatome.com

Abstract

In recent years, Safety Authorities worldwide have undertaken comprehensive reviews and updates of the historical methodologies and requirements governing the safety demonstration of nuclear reactors under accident conditions. This initiative aims to incorporate advancements in fuel materials and designs, evolving reactor operating conditions, as well as significant scientific progress and enhanced understanding of fuel behavior since the establishment of the original requirements in the 1970s. Among the various postulated accident scenarios, the Loss-Of-Coolant Accident (LOCA) has been a particular focus, with regulatory methods and criteria evolving notably in the United States and France.

A key aspect of these updated regulations is the consideration of cladding embrittlement due to in-service hydrogen absorption. An additional consideration is the potential for early onset of breakaway oxidation—which could severely compromise the mechanical integrity of the cladding.

M5Framatome*, the reference alloy developed by Framatome for Pressurized Water Reactor (PWR) cladding applications, is a Zr-1%Nb-O alloy with controlled iron and sulfur contents and a fully recrystallized microstructure. This study presents experimental evidence of the excellent performance of this material under LOCA conditions. The alloy's performance is primarily attributed to its low corrosion rates and minimal hydrogen uptake under nominal operating conditions. These properties have been validated through extensive operational experience in PWRs, including high burnups and demanding irradiation environments. Furthermore, recent autoclave

tests conducted under VVER 1000 standard chemistry have confirmed the alloy's excellent corrosion resistance, with favorable outcomes observed for exposure durations up to 1400 days.

At elevated temperatures, both unirradiated and irradiated **M5$_{\textbf{Framatome}}$** claddings exhibit oxidation rates lower than those predicted by the Baker-Just and Cathcart-Pawel correlations. Combined with low pre-transient hydrogen content, this results in excellent mechanical performance during quench in semi-integral tests (strength-based approach used in France and Japan) and high residual ductility after quench (as required by the US NRC methodology). Additionally, no evidence of breakaway oxidation was observed in either pristine or surface-damaged samples after 10,000 seconds at 1000 °C.

Acknowledgment: Several tests presented in this paper have been performed by CEA, in the frame of the GAINE project from the French nuclear tripartite institute CEA EDF Framatome, JAEA and UJP Praha.

Keywords: LOCA, **M5$_{\textbf{Framatome}}$**, high temperature oxidation, breakaway oxidation, post- quench behavior.

PERFORMANCE OF M5FRAMATOME FUEL ROD DURING REACTIVITY INITIATED ACCIDENT INFORMED BY ADVANCED MODELING

Romain Borrossi[1]*, Raphaël Chosson[1], Xavier Haller[1]

[1]*Framatome, 2 rue Professeur Jean Bernard, 69007, Lyon, France*
E-mail of the Corresponding Author: romain.borrossi@framatome.com

Abstract

M5Framatome* is the reference alloy of Framatome for PWR cladding applications. It is a zirconium-based alloy containing 1% niobium, and oxygen, iron, and sulfur. It has a fully recrystallized microstructure. The absence of tin in **M5Framatome** results in a very low corrosion rate and hydrogen pick-up during normal operation, even at high burnup.

Extensive Reactivity-Initiated Accident (RIA) testing, conducted in pulse reactors, has demonstrated the excellent performance of **M5Framatome** fuel rods with respect to Pellet-Cladding Mechanical Interaction (PCMI) induced failure, up to very high burnup. This low susceptibility to PCMI-induced failure is primarily attributed to the alloy's corrosion resistance, and crucially to the low hydrogen content, which is widely recognized as the principal factor governing cladding embrittlement under RIA conditions.

Over the past decade, several regulatory frameworks have evolved worldwide, particularly in France and in the US. These changes included updates to RIA fuel safety criteria, reflecting the degradation of the cladding mechanical performances in accident scenarios due to operational hydrogen uptake. Within these new regulatory contexts, **M5Framatome** cladding continues to provide substantial safety margins against the risk of PCMI-induced failure during RIA events.

Following the PCMI phase of a RIA, a boiling crisis may occur if the injected energy is sufficiently high. Recent RIA test programs have focused on the behavior of fuel rods during the post-Departure

from Nucleate Boiling (DNB) phase, where cladding temperatures can exceed 700 °C. Under these conditions, a potential risk of cladding ballooning and burst due to rod internal overpressure exists. Available test results indicate that the behavior of $\mathbf{M5_{Framatome}}$ segments still meet the safety requirements after DNB, (i.e., no loss of coolable geometry, no fuel-coolant interaction due to fuel dispersal), even in case of cladding burst. Because the post-DNB behavior involves several complex phenomena, advanced fuel performance codes are useful tools to support tests interpretation and assess their transposability to anticipated in-reactor RIA transients.

This work presents a review of the RIA testing, highlighting the robust performance of the $\mathbf{M5_{Framatome}}$ cladding during both the PCMI and post-DNB phases. Additionally, the interpretation of high enthalpy tests exhibiting cladding ballooning and burst is supported by advanced simulations conducted with the ALCYONE fuel performance code, which is developed by CEA in partnership with EDF and Framatome.

Acknowledgment: Several tests presented in this paper have been performed by JAEA, with support from NRA, under cooperative research agreements between JAEA and Framatome. The simulations presented in this paper have been conducted with the ALCYONE fuel performance code developed by CEA with the constant financial and technical support of EDF and Framatome.

Keywords: RIA, M5Framatome, PCMI, DNB, Fuel performance modelling

FRAMATOME'S FUEL ROD AND FUEL ASSEMBLY STRUCTURAL MATERIALS FOR PWR AND VVER REACTORS

R. Borrossi[1]*, C. Forot[1], P. Guillermier[1], P.-B. Hoffmann[2], E. W. Schweitzer[2]

[1]Framatome SAS, 2 rue Prof. Jean Bernard, 69007 Lyon, France
[2]Framatome GmbH, Paul-Gossen-Str. 100, 91052 Erlangen, Germany
E-mail of the Corresponding Author: romain.borrossi@framatome.com

Abstract

For PWR and VVER reactors, Framatome mainly proposes its two flagship zirconium materials for fuel assembly design: M5$_{\textbf{Framatome}}$* for fuel rod claddings and Q12* for structural components.

M5$_{\textbf{Framatome}}$ is a zirconium-base alloy containing 1% niobium, no tin and well-controlled oxygen, iron, and sulfur contents, combined with a stable, fully recrystallized (RXA) microstructure. The absence of tin in **M5$_{\textbf{Framatome}}$** results in a very low corrosion rate and hydrogen pick-up at high burnup and in high duty irradiation conditions. The choice of the fully recrystallized condition ensures high creep resistance and a high stress-relaxation rate, which provides excellent Pellet-Cladding Interaction (PCI) behavior.

The quaternary Zr alloy Q12 with Nb, Sn, and Fe has become the standard structural material for Framatome's PWR fuel assemblies. In comparison to conventional Sn-alloyed material derived from Zry-4, Q12 features lower corrosion kinetics and thus reduced hydrogen uptake combined with good creep and growth properties which make it particularly suited for high burnup applications.

Both M5$_{\textbf{Framatome}}$ and Q12's excellent performance in PWR have been demonstrated up to high burnups (above 75 GWd/tU fuel rod average burnup) thanks to dedicated irradiation programs in

several reactors in the world followed by post irradiation characterization and testing.

For VVER reactors, in 2020, Framatome launched long-term corrosion testing programs to support the licensing of Framatome's VVER fuel assembly design. The long-term corrosion testing of $M5_{\mathbf{Framatome}}$ and Q12 materials in a static autoclave at 360 °C with VVER water chemistry is now near 1400 days of exposure. The operating experience and long-term corrosion testing demonstrate the excellent performance of Framatome's alloys in both PWR and VVER conditions.

Keywords: Zirconium alloy, High Burn up, $\mathbf{M5}_{\mathbf{Framatome}}$, Q12

DEVELOPMENT OF EUROPEAN VVER-1000 FUEL ASSEMBLY BY FRAMATOME: PROJECT OBJECTIVES AND OVERVIEW OF THE DEVELOPMENT APPROACH

Dounia Lhachemi[1]*, Marco Messingschlager[2]

[1] rue Professeur Jean Bernard, 69007 Lyon – FRANCE
[2]Paul-Gossen-Str. 100, 91052 Erlangen – GERMANY
E-mail of the Corresponding Author: Dounia.lhachemi@framatome.com

Abstract

Framatome, a leading fuel designer and manufacturer, is actively developing new fuel assemblies for VVER-1000 and VVER-440 reactors to provide European utilities with diversified and fully

European-controlled fuel solutions. The company began work on VVER- 1000 fuel assemblies in 2019, prioritizing safety, reliability, and an optimal balance between performance and robustness.

These projects leverage Framatome's extensive expertise in fuel development, collaborations with key VVER stakeholders, advanced modelling tools, and a comprehensive range of prototyping and testing capabilities. This approach positions Framatome to meet its goals for delivering high-performance, robust fuel assemblies that adhere to the highest safety and quality standards.

From the beginning of the project, requirements are defined using a holistic lifecycle management approach (System Engineering methodology) and advanced numerical tools are used to optimize the design for mechanical, thermal, and hydraulic performance. Now the project has entered a phase of components and full-scale testing to assess performance. Tests Experimental validation program cover hydraulic, mechanical, and endurance tests under representative conditions.

In addition, from the early stage of the project, manufacturing process development is also conducted to ensure the feasibility and industrialisation of proposed designs by involving manufacturing experts.

The presentation will also highlight major design achievements and provides timelines for the qualification of this new fuel assembly.

Keywords: FA bow, FA distortion, Operating experience, FA performance

DEVELOPMENT OF EUROPEAN VVER-440 FUEL ASSEMBLY BY FRAMATOME

Guy Gentet[1]*, Pascal Blanc [2]

[1] rue Professeur Jean Bernard, 69007 Lyon – FRANCE
[2] Paul-Gossen-Str. 100, 91052 Erlangen – GERMANY
E-mail of the Corresponding Author: Guy.gentet@framatome.com

Abstract

To increase diversification and security of fuel supply for European utilities operating VVER reactors, Framatome is developing VVER-440 and VVER-1000 fuel assemblies. Designed from the ground up, these products ensure 100% European sovereignty.

These projects leverage Framatome's extensive expertise in fuel development, collaborations with key VVER stakeholders, advanced modelling tools, and a comprehensive range of prototyping and testing capabilities. This approach positions Framatome to meet its goals for delivering high-performance, robust fuel assemblies that adhere to the highest safety and quality standards.

The development of the VVER-1000 Fuel Assembly started in 2019 and was joined in 2024 by the development of VVER-440 fuel assemblies (fixed and follower).

The presentation will focus on the VVER-440 fuel assembly development project for which Framatome relies on its extensive know-how in fuel product development, partnerships with European Utilities, key stakeholders, state-of-the art codes & methods, innovative design, as well as prototyping, testing, and manufacturing technologies ranging from the conventional to leading-edge.

The presentation will introduce the key enablers for a fast-paced development (Systems Engineering Methods, Verification and validation Plan, Rapid Prototyping) as well as outlines the main achievements in term of design and the timelines for the qualification of these new VVER-440 fuel assemblies.

Keywords: VVER-440, European sovereignty, Fuel assembly design

FRAMATOME SOLUTIONS TO SUPPORT UTILITIES FOR LONG TERM OPERATION

Guy Gentet[1]*, Lars Ackermann [2]

[1] rue Professeur Jean Bernard, 69007 Lyon – FRANCE
[2] Paul-Gossen-Str. 100, 91052 Erlangen – GERMANY
E-mail of the Corresponding Author: Guy.gentet@framatome.com

Abstract

As the demand for electrical power continues to rise, extending the operational lifetime of nuclear power plants becomes increasingly crucial. Traditionally, nuclear plants have been designed for a 40-year operational life, but with advancements in technology and safety, these lifespans are now being extended, often to 60 years and potentially beyond 80 years. However, the lifetime of a nuclear power plant is largely constrained by the design limitations of its components. While many components can be refurbished or replaced during scheduled outages, the reactor pressure vessel (RPV) is an exception, making it the most critical and vulnerable component when it comes to lifetime extension and Long-Term Operation (LTO).

To extend the RPV's operational lifetime, reducing the neutron flux that it is exposed to, is essential. A key strategy for reducing neutron fluence at the RPV involves the implementation of either shielding fuel assemblies in the reactor core or the insertion of flux depressor assemblies in the fuel assemblies at the periphery of the core.

The shielding fuel assemblies can be customized to provide adjustable, in general medium to very high, reduction of the neutron fluence at any elevation or at specific axial location. The flux depressor fuel assemblies can be considered when a medium level of reduction of the neutron fluence is requested at all axial locations. The impact of these two solutions on the core can be evaluated through detailed simulation.

This presentation will summarize the evaluation of a shielding fuel assembly to optimize the shielding with the most effective fuel

assembly features and of a flux depressor assembly which provides a homogenous and medium flux attenuation. Both examples demonstrate the capability of Framatome to find customer specific solutions, ensure their licensing, and gain maximum benefits in the operation of nuclear power plants.

Keywords: Long Term Operation, Shielding fuel assembly, Flux depressor assembly

FUEL RELIABILITY IMPROVEMENTS

Guy Gentet[1]*, Pierre-Henri Louf[1], Christophe Petit[1]

[1]Framatome SAS
2 rue Professeur Jean BERNARD
69007 LYON, FRANCE
E-mail of the Corresponding Author: Guy.gentet@framatome.com

Abstract

Since the 1970s, the reliability of fuel assemblies developed by Framatome has seen remarkable improvement, as evidenced by a significant reduction in failure rates and enhanced control over the mechanisms leading to loss of loss of fuel rod integrity. Initially, fuel rod failure rates were on the order of several hundred failures per million rods, mainly due to the presence of foreign bodies (metallic debris), mechanical failures, wear phenomena or manufacturing quality issues.

To address these challenges, Framatome has progressively introduced major innovations in the design of its assemblies. The integration of debris filters in the lower end fittings represented a decisive advancement, intercepting foreign materials likely to cause cladding perforations ([1]). In addition, the optimization of grid designs — through the adoption of specific rod support geometries and

high-strength materials — has significantly reduced wear from friction between the rods and the grid supports. In manufacturing, various improvements, in particular in welding processes, further allowed significant improvements in reliability ([2]).

Over the decades, these advances have reduced the failure rate to less than 10 failures per million rods, positioning Framatome at the leading edge in terms in fuel reliability. Analysis of the remaining causes of loss of tightness now shows a predominance of debris fretting.

The latest generation of assemblies, embodied by the GAIA design, illustrates the culmination of this innovation process. GAIA incorporates next-generation debris filters, grids with optimized geometry, and advanced materials, offering increased robustness under core operating conditions and further enhanced reliability.

Looking ahead, improvement prospects are focused on the use of additive manufacturing to produce filters with complex geometries, enabling increased debris retention efficiency while minimizing hydraulic pressure losses and rod failure. This approach paves the way for a new generation of tailor-made components, adapted to the increasing expectations for fuel reliability in nuclear reactors.

GAIA is a trademark or registered trademark of Framatome or its affiliates, in the USA or other countries.

Keywords: Reliability, Innovation, Quality

References:

1. S.M. Mazurkiewicz, C. Bretting, C. Petit. Addressing the Negative Industry Trend of Debris- related Failures in Commercial Nuclear Fuel, TopFuel, 2019

2. Thomas S. Wilkerson, John Strumpell, Jean-Pierre Devort, Pierre Mollard, How AREVA improves Fuel Quality & Reliability: Zero Tolerance for Failure and Fuel Excellence Plan outcomes, TopFuel, 2010

PROTECT CR: THE LEADING E-ATF SOLUTION BY FRAMATOME

K. Buchanan[1]*, T. Garnier[1], E. W. Schweitzer[2], N.Vioujard[1]

[1]Framatome SAS, 2 rue Prof. Jean Bernard, 69007 Lyon, France
[2]Framatome GmbH, Paul-Gossen-Str. 100, 91052 Erlangen, Germany
E-mail of the Corresponding Author: karl.buchanan@framatome.com

Abstract

Following the Fukushima event, the global nuclear industry accelerated research into Enhanced Accident Tolerant Fuel (E-ATF) with the intent of improving safety characteristics and performance and adding value for commercial light water reactors (LWR). Framatome's E-ATF near-term strategy is based on chromium-coated **M5$_{\textbf{Framatome}}$** cladding and Cr_2O_3-doped UO_2 fuel as an evolutionary solution and silicon carbide composite (SiC_f/SiC) cladding as our long-term breakthrough solution. Their implementation proceeds through a step-by-step process that moves from materials irradiation to reload deliveries, facilitated by key partnerships with US and French national research labs.

The first irradiation of Cr-coated **M5$_{\textbf{Framatome}}$** claddings under representative PWR conditions began in 2016 at the Gösgen nuclear power plant and six 12-month cycles have since been completed successfully. Full-length Lead Fuel Rod (LFR) insertions in Vogtle, Gösgen, Arkansas Nuclear One and Blayais followed from 2019 and a Lead Fuel Assembly (LFA) was inserted in Calvert Cliffs in 2021. With a burn-up of 73 GWd/tU reached in 2025, these programs have demonstrated the capability of Framatome's PROtect Cr-coated **M5$_{\textbf{Framatome}}$** fuel rods to withstand a full lifecycle in commercial LWRs.

The data obtained from irradiation programs and out-of-pile characterizations in both normal operation and accident conditions is being used to update licensing codes and methods (neutronics, thermal-mechanics, system thermal-hydraulics), and licensing

documents are in preparation for submittal to the US NRC (Nuclear Regulatory Commission) and to a European Safety Authority in the next six months.

In parallel, Framatome's manufacturing capability is ramping up with the installation in 2025 of an industrial-scale coating pilot equipment at our Paimboeuf tubing plant. Our Cr-coated cladding pilot, at full production, will be able to coat up to one hundred thousand (100,000) cladding tubes per year and thus have the capacity to support initial batch programs (several reloads per year).

These achievements are the result of an ambitious roadmap and driven by a decade long partnership with the US DOE, US and French national labs, and Framatome affiliate organizations in the US, France, and Germany. The anticipated near-term result will be batch readiness for the PROtect Cr-coated rods, for both the US and EU markets, by 2027.

PROtect and $\mathbf{M5_{Framatome}}$ are trademarks or registered trademarks of Framatome or its affiliates, in the USA or other countries.

DEVELOPMENT OF INNOVATIVE MATERIAL FOR LWR FUEL

N. Vioujard [1]*

[1]Framatome SAS, 2 rue Prof. Jean Bernard, 69007 Lyon, France
E-mail of the Corresponding Author: nicolas.vioujard@framatome.com

Abstract

In some countries, the existing fleet of commercial nuclear power plants faces economic pressure from low natural gas prices and the emergence of utility scale renewable generation. In some others, increased electricity demand to reduce carbon emission are incentives to build new reactors and/or to make power uprates. The accident at Fukushima Daiichi in 2011 highlighted the value of fuel margin in the case of a severe accident. These major trends in the power generation industry, among others, provoke questions as to where innovations in the fuel technology are useful for responding to the challenges faced by the nuclear power generation industry.

This presentation explores the broad array of advanced fuel concepts to assess the relative merits of each solution to the nuclear plant operators when compared to the reference technology of UO_2. The advanced technologies review includes UO_2 fuel with additives and dense fuels that will be interesting to the industry in the future.

Several criteria serve as the guidepost for the assessment. These criteria correspond to the key considerations for implementing any new technology including: economic aspects, manufacturing aspects, or physical properties and their impact in normal operations, or in the unlikely event of an accident.

The results show that the key benefits of advanced technologies are largely dependent on the industry trends and pressures. This systematic review of the innovative technologies in today's development pipeline will provide a key reference for decision makers as they face the questions of which technologies to implement and when these technologies will be available in large quantities.

IRRADIATION FEEDBACK OF FRAMATOME'S PROTECT-CR CLADDING TO A BURNUP OF >70 MWD/KGU

E. W. Schweitzer[1]*, T. Garnier[2], K. Buchanan[2], M. Aumand[3]

[1]*Framatome GmbH, Paul-Gossen-Str. 100, 91052 Erlangen, Germany*
[2]*Framatome SAS, 2 rue Prof. Jean Bernard, 69007 Lyon, France*
[3]*Framatome Inc., Framatome Inc. 3315 Old Forest Rd, Lynchburg, VA 24501-2912*

E-mail of Corresponding Author: elmar.schweitzer@framatome.com

Abstract

Framatome's research activities on Enhanced Accident Tolerant Fuel (E-ATF) with improved safety characteristics and added performance and value for commercial light water reactors lead to the development of PROtect-Cr fuel, which is based on chromium-coated M5$_{\mathbf{Framatome}}$ cladding and Cr_2O_3-doped UO_2. The implementation of PROtect-Cr fuel is proceeding step-by-step going from materials irradiation to reload deliveries facilitated by key partnerships.

The PROtect-Cr solution improves performance through the deposition of a protective chromium coating on the external surface of M5$_{\mathbf{Framatome}}$ cladding tubes using physical vapor deposition. After first irradiation tests of unfueled specimens in the OSIRIS test reactor and with the IMAGO irradiation program in the Gösgen-Däniken Nuclear Power Plant, several lead test rod programs were launched in the U.S. and Europe. In 2025, several programs in the US have reached end-of-life, and a maximum average fuel rod burnup of 73 MWd/kgU has been reached at Gösgen-Däniken with the irradiation continuing for its sixth and final cycle.

This presentation covers the irradiation feedback from Framatome's PROtect-Cr E-ATF solution showing poolside measurements and visual examinations for up to 5 cycles and non-destructive and destructive examinations. The characterizations and tests have confirmed the good in-pile performance of Framatome's Cr-coated M5$_{\mathbf{Framatome}}$ cladding. These results are complemented with an

outlook on the industrialization of the coating process to reach reload readiness in 2027.

PROtect and M5$_{\mathbf{Framatome}}$ are trademarks or registered trademarks of Framatome or its affiliates, in the USA or other countries.

Keywords: E-ATF, Cr-coated cladding, irradiation experience

MITIGATION OF LOCAL POWER PEAKING BY INTRODUCING HF SPACER PINS IN THE VVER-440 FOLLOWER FUEL ASSEMBLY

Akif Abdullayev[1]*, Douglas Hake[2], Jan Höglund[1]

[1]Westinghouse Electric Sweden AB, 72163 Västerås, Sweden
[2]Westinghouse Electric Company, PA 16066 USA
E-mail of the Corresponding Author: *akif.abdullayev@westinghouse.com*

Abstract

A VVER-440 core consists of 312 fixed Fuel- and 37 follower Fuel Assemblies (FA), in total 349 FA. The fixed FAs function as fuel assembly types in western NPPs and VVER-1000 plants. The follower assembly is an extension of the control assembly and is attached to the absorber part by a mating feature incorporated in the top nozzle. The follower is intended to take the place of the control assembly when it is withdrawn.

Enhanced moderation conditions in the fuel-absorber connection region right above the follower fuel rods and low absorption of thermal neutrons in the gas plenum region at the top of the follower fuel rods induces an undesired local power peak in the

nearby fuel rods of neighbouring fixed FA when the control assembly is partially inserted in a core. This power peaking in the region of the fuel-absorber connection has two undesirable effects on fuel operation: 1) It increases the neutron flux and linear heat generation and 2) it increases the power ramp when the control rod is moved. For better fuel performance and thermal margins, it is desirable to mitigate this power peak as much as possible.

In Westinghouse VVER-440 follower FAs a neutron absorbing feature is added to the peripheral fuel rods in the form of a hafnium (Hf) spacer to mitigate the power peak in neighbouring fixed FAs. In the 36 peripheral fuel rods, the stainless-steel spacers between the plenum spring and the pellet stack are replaced with Hf spacers which are also relocated to a position between the top end plug and the spring.

In this paper, the results from analyses of the Hf spacers effectiveness to mitigate the undesired power peaks are presented. At first a partially inserted follower FA was modelled in MCNP5 and the pin powers in a neighbouring (fixed) fuel assembly were calculated. Results show that the suppression effect is very strong and that the local excess from the Hf pins is around 15%. However, the axial power distribution from MCNP is not fully realistic and the MCNP calculation also does not consider doppler feedback. These effects are only possible to model in a 3D core simulator.

A more realistic evaluation of the power ramp was obtained by modelling the operation of the follower FAs with Hf pins in the Westinghouse ALPHA-H/PHOENIX-H/ANC-H (APA-H) 3D core simulation code package. A series of calculations was performed for a representative VVER-440 operating cycle with the lead bank moving from an all-out (250 cm) to an all-in (0 cm) position at hot full power. The axial surface power was collected for every fixed FA neighbouring each lead bank. The conclusion that the Hf spacers in the upper non-fuel region of the follower FA do suppress local power in neighbouring assemblies as predicted.

In summary, the results from the local power peak study in MCNP5 and the 3D core simulator modelling in APA-H, clearly demonstrate that Hf pins in the Westinghouse VVER-440 follower fuel

is an effective way of mitigating the power peak in surrounding fuel assemblies.
Keywords: VVER-440, Follower, Power Peak, Hafnium

Co-funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or Euroatom. Neither the European Union nor the granting authority can be held responsible for them

APIS

NEXT GENERATION VVER-1000 FUEL DEVELOPMENT

Alex Riznychenko [1]*, Uffe Bergmann [1], Jan Höglund [1]

[1] *Westinghouse Electric Sweden AB, 72163 Västerås, Sweden*
E-mail of the Corresponding Author: riznyco@westinghouse.com

Abstract

The upgraded Robust Westinghouse Fuel Assembly designs (RWFA/RWFA-T) have demonstrated an excellent irradiation performance in mixed and homogeneous cores of VVER-1000 reactors in Ukraine, Temelín NPP in the Czech Republic and Kozloduy NPP in Bulgaria with a positive overall feedback from all Customers on the fuel performance.

Further enhancement of products for the VVER-1000 market considering the voice of the Customers for potential design improvements is a part of Westinghouse long-term strategy. The ongoing Next Generation VVER-1000 Fuel Development is presented here.

In 2020 Westinghouse initiated a Research and Development project focused on developing a Next Generation Grid Design for Westinghouse VVER-1000 fuel assembly designs family. The

objective of the development is to develop a structural grid with improved neutronic, mechanical and thermal-hydraulic properties versus the existing Westinghouse grid designs for VVER-1000 fuel. Zr-alloy grid material is used to enhance the neutron economy.

To compensate for the expected reduced grid strength characteristics with use of zirconium material, the grid mechanical design is optimized for increased lateral strength by transitioning from a strap-based design to individual grid cells design. It is expected that this design approach together with optimized to minimize flow resistance cell shape and grid spring will also result in a sufficient overall fuel assembly pressure drop reduction to meet the design objectives. Results of the performed Computational Fluid Dynamics (CFD) calculations, scoping mechanical and comparative tests utilizing prototype grids made by additive manufacturing demonstrated superior lateral strength characteristics and significantly lower pressure drop for the new grid design. Confirmatory mechanical testing is planned to be completed in 2026/2027. Lead Test Assemblies with the Next Generation Grid Design are expected in 2028.

Keywords: VVER-1000, Development, Grids, Pressure Loss

Co-funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or Euroatom. Neither the European Union nor the granting authority can be held responsible for them

APIS

FRAMATOME FUEL RESEARCH CENTERS & LABS

G. Bolsee[1], V. Rebeyrolle[1]*, O. Calonne[2], D. Blavius[3], B. Stepnik[4]

[1]Framatome SAS, 2 rue professeur Jean Bernard, 69007 Lyon – France
[2]Framatome, Avenue Paul Girod, 73400 Ugine – France
[3]Framatome CERCA, 54 Avenue de la déportation, 26100 Romans – France
[4]Framatome GmbH, Seligenstädter Str 100 63791 Karlstein am Main, Germany
E-mail of the Corresponding Author: veronique.rebeyrolle@framatome.com

Abstract

Research dedicated to Fuel is one of the key pillars to offer utilities reliable and high-performance advanced fuel and to develop innovative features providing added value. The Fuel Business Unit of Framatome has 3 main labs dedicated to research:

The first one is the Component Research Center (CRC) located in Ugine, focusing on zirconium alloys manufacturing and metallurgy, with 2 smaller labs close to the manufacturing plants in Jarrie for extractive metallurgy and in Paimboeuf for tube manufacturing. The Component Research Center activities encompass development of advanced Zirconium alloys, microstructural characterization and out-of-pile properties and optimization and simulation of manufacturing processes. Process simulation covers a large variety of processes, such as VAR melting, forging or pilgering.

The second one is the prototyping lab at Karlstein in Germany, specialized in the manufacturing and testing of advanced fuel assembly designs. This lab is the center of excellence to support new Fuel designs and to manufacture spacer grids including advanced processes development and Fuel Assemblies cages for all reactor types. Recently welding processes have been developed for adaptation to the VVER 1000 spacer grid manufacturing.

Finally, the CERCA Research & Innovation Lab (CRIL) is dedicated to advancing uranium-based fuel development, contributing

significantly to research more particularly on fuel for Research Reactors and for Advanced Modular Reactors. The lab is designed to offer flexibility and reactivity enabling adaptation to a wide range of fuel types. It was recently the case with metallic Uranium plates made with Additive Manufacturing.

Beside these labs dedicated to Fuel, the development and qualification of advanced fuel designs takes benefit from the Framatome Technical Centers located in France and in Germany. Among them, the Kathy loop in Karlstein is dedicated to Critical Heat Flux tests on Fuel bundles for the assessment of thermal-hydraulic performance and licensing of advanced fuel designs.

Keywords: Fuel Assembly, Zirconium, Research, Facilities

APPROACHES TO SUPPLY UKRAINIAN NPPS WITH CORE COMPONENTS

Valeriy Zuyok*[1], Oleksandr Mazurok[2], Oleg Zelenyy[3], Sergiy Lavrov[4], Oleg Godun[3], Anton Makarenko[5]

[1]National Science Center “Kharkov Institute of Physics and Technology”, Ukraine, Kharkiv, 61108
[2]ES Group LLC, Ukraine, Kyiv
[3]Scientific and Technical Center, JSC “NNEGC "Energoatom”, Ukraine, Kyiv
[4]Atomenergomash, JSC “NNEGC “Energoatom”, Ukraine, Kyiv,
[5]Scientific and Technical Complex "E.O. Paton Electric Welding Institute", Ukraine, Kyiv
E-mail of the Corresponding Author: valeriyz@kipt.kharkov.ua

Abstract

Ukrainian National Nuclear Energy Generating Company Energoatom has completely refused from russian nuclear fuel and intends to switch to fuel of its own fabrication as well as fuel fabricated by the Swedish plant of Westinghouse. In addition to fuel, an integral part of the VVER-1000 core are rod cluster control assemblies (RCCAs), and for VVER-440 reactors – follower assemblies and shield assemblies (RNPP-1), the sole manufacturer of which is the RF. Therefore, Ukraine faces an acute issue of providing these alternative core components.

Taking into account the limited reserve of “fresh” RCCAs, a set of activities was performed to temporarily meet the needs of Ukrainian NPPs with VVER-1000 reactors to extend the service life of the existing ones. The justification for extending the service life of RCCAs is based on the mechanical reliability of control rods and RCCA design in general, meeting the criteria for physical efficiency of RCCAs, thermal characteristics of control rods, taking into account changes in the characteristics of their structural materials during the entire previous period of operation in a reactor core. The results of the calculations enabled to extend the service life of almost all RCCAs

operating in VVER-1000 NPPs of Ukraine from 25500 hours to 38000 hours in the Automatic Regulation group and from 75600 hours to 113500 hours in the Scram group.

The extra time enabled JSC NNEGC Energoatom to start fabrication of new nationally developed RCCAs at one of the Ukrainian enterprises. The first batch of the Ukrainian RCCAs was fabricated in 2023. The new RCCAs do not differ from the standard RCCAs manufactured in the RF in terms of external geometric dimensions, weight, neutron absorption efficiency and dimensions of the connection joints, but only materials produced by the technologies developed and mastered at Ukrainian enterprises are used in their fabrication. A set of activities was performed to implement the national RCCAs and pilot operation in the reactor core was started.

In order to ensure continuous operation of VVER-440 reactors, a set of activities was started to justify the safe operation of the nationally designed follower assembly and the shield assembly developed by Energoatom. The new assemblies do not differ from the russian assemblies in terms of external geometric dimensions, dimensions of connection joints and hydraulic characteristics, but they are also fabricated using only materials that have been produced under the technology developed and mastered at Ukrainian or EU enterprises.

The report presents the results of a comparative analysis of RCCA, absorber aasembly and shield assembly designs developed by JSC NNEGC Energoatom and standard designs manufactured in the RF, provides some test results and calculations for justification of mechanical reliability of the design in general, meeting the criteria for physical efficiency of the RCCAs, as well as the results of thermotechnical calculations, which enabled justification of safe operation of these core components.

Keywords: RCCA, control rod, absober assembly, follower assembly, shield assembly, VVER

RADIATION SHIELDING ASSESSMENT OF THE CYCLOTRON CENTRE AT INRNE-BAS

A. Demerdjiev[1]*, D. Tonev[1], G. Dimitrova[1], N. Goutev[1], E. Geleva[1], V. Variyska[1]

[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, blvd. Tzarigradsko Chaussee 72, Sofia, BG-1784, Bulgaria
**E-mail of the Corresponding Author: angeld@inrne.bas.bg*

Abstract

At the Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, a new Cyclotron Centre is currently under development. As the project is in its design phase, a comprehensive assessment of the radiation shielding is required. We are employing the Monte Carlo code FLUKA [1, 2] to simulate the distribution of the radiation fields and to evaluate specific activity of the long-lived radioisotopes (^{154}Eu, ^{152}Eu, ^{134}Cs, ^{60}Co, ^{54}Mn) produced in the bunker walls. In the present study, the geometries of the cyclotron and the beamlines have been implemented, and the presence of reinforcement steel rods within the shielding walls has been taken into account. Results for the distribution of prompt and residual radiation fields have been obtained for different irradiation scenarios, along with the in-depth profiles of the specific activity of long-lived radioisotopes accumulated during the lifetime of the cyclotron.

Furthermore, the study also examines the impact of substituting portions of the shielding bunker walls with low-activation concrete [3], aiming to reduce long-term radiological hazards. The results show that selecting construction materials with low concentration of impurities decreases significantly the radiation levels and the specific activity of the accumulated long-lived radioisotopes. The results of this study will serve as a guideline for the final design of the facility's radiation shielding.

Acknowledgment: This research was funded by the Ministry of Education and Science of the Republic of Bulgaria, through the National Program D01-99: Qualification improvement in the field of nuclear technologies and nuclear engineering.

Keywords: TR-24 cyclotron, Radiation shielding, Monte Carlo simulations, FLUKA

References:

1. A. Ferrari, P.R. Sala, A. Fassó, J. Ranft. FLUKA: A Multi-Particle Transport Code., CERN-2005-010, INFN-TC-05-11, SLAC-R-773, 2005.

2. T.T. Böhlen, F. Cerutti, M.P.W. Chin, A. Fassò, A. Ferrari, P.G. Ortega, A. Mairani, et al. The FLUKA Code: Developments and challenges for high energy and medical applications. *Nuclear Data Sheets* **2014**, 120, pp. 211-214.

3.Stichelbaut F., Pí erard J., Kupers L. Compositions of low activation concrete and use thereof. *Eropean Patent,* 3266754A1, 2018.

SESSION 4 FUEL MODELLING AND EXPERIMENTAL SUPPORT

TOWARDS MODELLING OF ADVANCED-TECHNOLOGY FUEL RODS WITH THE TRANSURANUS CODE

A. Schubert*[1], I. M. Paponetti[2], P. Aragón[1], D. Jaramillo Sierra[1], Zs. Soti[1], P. Van Uffelen[1]
[1]*European Commission – Joint Research Centre, Karlsruhe, Germany*
[2]*University of Bologna, Italy*
E-mail of the Corresponding Author: Arndt.Schubert@ec.europa.eu

Abstract

In the frame of the Euratom APIS project (Accelerated Program for Implementation of secure VVER fuel Supply), properties for advanced technology fuel (ATF) have been implemented in the TRANSURANUS fuel rod performance code of the European Commission's Joint Research Centre (JRC). The APIS project includes the application and qualification of fuel performance simulations, making use of datasets from the International Fuel Performance Experiments (IFPE) database of the OECD/NEA, as previously applied in Co-ordinated research projects (CRP's) of the IAEA (FUMEX-II and FUMEX-III).

This paper demonstrates the implementation and application of near-term advanced-technology fuel in the TRANSURANUS code – by the examples of Cr-doped UO_2 fuel, Zirlo cladding as well as Cr-coated cladding. For 4 transient-irradiated VVER fuel rods compiled in the IFPE database (Kola3-MIR dataset), the outcomes of the earlier IAEA projects were reproduced and complemented by means of the latest version of TRANSURANUS. Starting from standard UO_2 fuel and Zr1Nb cladding and assuming identical irradiation histories, the simulated impact of the above-mentioned hypothetical ATF fuel and cladding configurations was analysed for: the fuel-to-cladding gap sizes, the fuel centre temperatures, the integral fission gas release as

well as the final radial profiles of the local Xe concentration. The results are consistent with those published earlier in the frame of the APIS project but show in more detail the impact on fission behaviour in doped fuels with larger grains and initial density. The simulated performance of ATF fuel rods meets expectations when compared to the standard options.

Keywords: fuel performance, modelling, Advanced Technology Fuel

GALILEE, FRAMATOME FUEL ROD PERFORMANCE CODE FOR LWR

Y. Wrtal[1]*, J. Plancher[1], G. Gentet[1], M. Brunmayr[2], T. Distler[2], O. Wieckhorst[2]

[1]Framatome SAS, 2 rue Professeur Jean Bernard, 69007 Lyon, FRANCE
[2]Framatome GmbH, Paul-Gossen-Straße 100, 91052 Erlangen, GERMANY
**E-mail of the Corresponding Author: yannick.wrtal@framatome.com*

Abstract

The simulation of fuel rod behaviour in a nuclear reactor requires reliable and accurate fuel performance codes. GALILEE is a fuel rod performance code developed by Framatome, which models the thermal and mechanical behaviour of fuel rods during normal operation and transient scenarios. The R&D around the code relies on the expertise of recognized research centers such as CEA in France, but also European programs such as the OperaHPC project.

This presentation exhibits first the extended experimental database of GALILEE in terms of variability of LWR products (focusing on UO_2, UO_2-Gd_2O_3 and MOX fuels with Zy-4 and M5 claddings), ranges of operating conditions, and diversity of experimental results either from commercial irradiation or international programs.

The main models and characteristics of GALILEE are then briefly described, and some validation plots are presented to illustrate the predictive capability of the code on a few integral physical quantities of interest such as fuel central temperature, fission gas release or helium balance.

Benchmarks on Halden and OECD/NEA data are also available to demonstrate that there are no significant biases for VVER application.

In parallel to GALILEO, the US certified fuel rod performance code, GALILEE is intended to be the reference fuel rod performance code to cover Framatome's current and future LWR fuel products in the European and export markets, including the VVER market.

M5$_{Framatome}$, GALILEE and GALILEO are trademarks or registered trademarks of Framatome or its affiliates, in the USA or other countries.

Keywords: Fuel rod performance code, GALILEE

FRAMATOME ADVANCED SOLUTIONS AGAINST GRID-TO-ROD FRETTING WEAR RISK

Julien Pacull[1]*, Veit Marx[2]

[1]Framatome SAS, 2 rue du Professeur Jean Bernard, 69007 Lyon, France

[2]Framatome GmbH, Paul-Gossen-Straße 100, 91052 Erlangen, Germany

Email: julien.pacull@framatome.com

Abstract

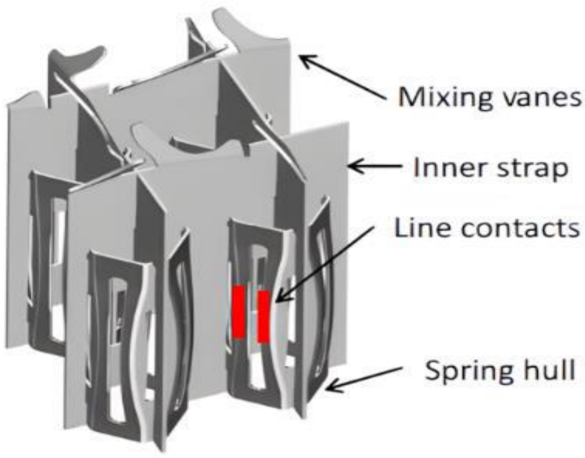
In operation, nuclear fuel assemblies are subjected to hydraulic excitations induced by the primary coolant flow. These excitations can trigger a large variety of flow-induced vibration mechanisms, from the ever-present turbulence-induced vibrations to resonance phenomena, e.g. fluid-elastic coupling or self-induced excitation. Excessive vibrations may result in grid-to-rod fretting wear, which has been widely reported as one of the main causes of fuel leakage in PWR worldwide.

Leveraging its extensive experience in nuclear fuel design, Framatome has developed a set of solutions to address grid-to-rod fretting and ensure the highest level of performance of its products.

These solutions include advanced experimental protocols which are used for testing actual fuel assembly components in severe conditions, that bounds the hydraulic loads in operation. In addition, state-of-the-art coupled fluid-structure simulation tools are used to gain a deeper understanding with respect to the physical phenomena at play, and to support the analysis of both test results and operational experience feedback.

The advanced methodologies developed by Framatome have been key to develop hardware designs which have demonstrated high resistance to grid-to-rod fretting. Namely, the unique ‘line contact’ type of rod support offered by HTP and GAIA designs has been operated successfully in a wide range of reactors, worldwide. Bottom

nozzle optimization, resulting in a uniform distribution of inlet flow, and to a reduction of turbulent excitation, has also been proven feasible.



Detail of GAIA mixing grid design

This presentation will provide an overview of the solutions developed by Framatome to mitigate the risk of excessive grid-to-rod fretting, and their benefits for the operation of the nuclear power plants. These solutions will be leveraged to support VVER 440 and 1000 fuel design development.

Keywords: grid-to-rod fretting, flow-induced vibrations, design, experiment, simulations

FRAMATOME'S ADVANCED CORE DISTORTION SIMULATION CAPABILITIES

Veit Marx[1]*

[1]Framatome GmbH
Paul-Gossen-Str. 100
91052 Erlangen- Germany
E-mail of the Corresponding Author: veit.marx@framatome.com

Abstract

Fuel assemblies (FA) in PWR are subjected to dimensional changes during reactor operation. These dimensional changes also comprise lateral deformation or FA bow along the vertical axis. Increased fuel assembly bow can negatively influence the handling of the fuel. During core loading and unloading, increased fuel assembly bow can cause higher handling loads and time delays of the outage. Excessive bow can also cause delayed or incomplete control rod insertion. Preventing excessive bow is necessary to guarantee the safe operation of the reactor.

To prevent such consequences, Framatome has developed over the last years advanced prediction tools which are based on a multiphysics and multiscale approach. These tools can be used to predict e.g. core-wide bow and help to mitigate the effects of bow and to guarantee the safe operation of fuel.The bow prediction tools are also applied to define fuel designs with improved resistance against bow. Additionally, Framatome has created the COLOSS software which calculates the interaction forces during core loading and unloading. With the COLOSS software the risk of core loading delays can be reduced. Another tool calculates the drop time of distorted fuel assemblies.

An overview on Framatome's FA distortion related prediction tools and their validation in existing reactors - mainly 14ft - will be presented. It will be shown how the suite of tools is used to support the licensing and the safe and efficient operation of the fuel.

With this experience, the tools are currently in the process to be extended to VVER-1000. The tool extension supports Framatome’s VVER-1000 fuel development project. The status of the new developments on VVER-1000 will be presented and an outlook will be provided. These new developments will be the basis to offer bow related engineering services to VVER-1000 customers. Adaptation to other reactor types is also possible.

Keywords: core distortion, fuel assembly bow, prediction and validation, COLOSS, VVER-1000

FUEL ASSEMBLY BOW DNBR PENALTY FOR VVER-1000 REACTORS

Tommaso Carnicella[1]*, Akif Abdullayev[1], Ulf Lindelöw[1]

[1]*Westinghouse Electric Company, Sweden, Västerås, 72136.*
E-mail of the Corresponding Author:
Tommaso.Carnicella@westinghouse.com

Abstract

The fuel assembly bowing is a phenomenon characterized by the axial bending of the fuel assembly. A direct consequence of this deformation is the increase (or decrease) of the inter-assembly gap, which exposes the peripheral region of the fuel assembly to over-moderated (or under-moderated) conditions. This leads to two distinct effects:

- An increase of the relative power of the peripheral rod which are directly facing the bowed region.
- An increase of coolant flow rate in those regions affected by the increase of the gap.

The present analysis combines the effects of those two phenomena in order to evaluate the impact on the Departure from Nucleated Boiling (DNBR) for the peripheral sub-channels, and consequently the impact on the enthalpy rise factor for the peripheral rods.

The evolution of the inter-assembly gap is a complex process, influenced by overall core distortion, fuel assembly design, Burnup history, and core location. Additionally, it should be noted that enthalpy rise factor tends to decrease with increasing Burnup, but for conservative design purposes, a fixed value has been adopted without accounting for this reduction.

The Westinghouse Thermal and Hydraulic Design 3D subchannel VIPRE-W code was employed. The full-core homogeneous model of the VVER-1000 reactor was discretized into 208 subchannels. A finer mesh was applied in the region between the three hottest fuel assemblies, which is the area most affected by fuel

assembly bowing. Adjacent regions were also subdivided into subchannels to ensure accurate predictions for the first and second rows of fuel rods. The mesh resolution was progressively reduced with increasing distance from the region of interest.

The impact of inter-assembly gap variation is assessed by introducing penalties on the enthalpy rise factor. These penalties are derived by translating the margin between the Departure from Nucleate Boiling Ratio (DNBR) and its design limit (DL-DNBR, Ref 1.) into an equivalent power increase. This approach has been applied across a range of core thermal-hydraulic operating conditions, with the inter-assembly gap incrementally increased to evaluate its effect.

Based on the results of the DNBR analysis, penalties on the enthalpy rise factor have been quantified for the fuel rods of interest, specifically those located in the first and second rows at the periphery of the hot fuel assembly. These penalties account for the combined effect of thermal-hydraulic behaviour and the local power peaking in peripheral rods induced by the increase in the inter-assembly gap.

Keywords: DNBR; VVER-1000; Assembly Bow; Enthalpy Rise Factor; Thermal-hydraulic.

References:

1. S. Ray, A.J. Friedland, E.H. Novendstern. Westinghouse Advanced Statistical DNB Methodology – The Revised Thermal Design Procedure, Proceedings of the Third International Topical Meeting on Nuclear Power Plant Thermal Hydraulics and Operations, 1988, pp. 261–265

IRRADIATION-INDUCED GROWTH OF ADVANCED ZIRCONIUM ALLOYS

Stanislav Linhart[1]*, Josef Běláč[1], Radomír Řeháček[1], Alexander Y. Shevyakov[2], Anatoly Y. Gusev[2],
Pavel Pešek[3]

[1]ALVEL, a.s., Brno, Czech Republic
[2]SC VNIINM, Moscow, Russian Federation
[3]ÚJV Řež, Husinec-Řež, Czech Republic
E-mail of the Corresponding Author: Stanislav.linhart@alvel.eu

Abstract

A comprehensive irradiation program financed by JSC TVEL was launched in 2014 to investigate irradiation-induced growth and associated microstructural changes in advanced Russian zirconium alloys under standard operating conditions of a commercial VVER-1000 reactor. This work builds upon a general concept presented at the VVER conference in 2019 and has since developed into a structured research effort supported by experimental data. Irradiation and post-irradiation phases of this program are being organized and implemented by ALVEL.

The material matrix includes alloys that differ in chemical composition and heat treatment, resulting in a variety of initial microstructures. Six material cluster assemblies (MCAs), each containing longitudinal cladding tube segments (50 × 6 × 0.6 mm), were irradiated during five consecutive fuel cycles at Temelin NPP Unit 1, achieving six distinct levels of neutron fluence. Following each cycle,

one or two MCAs were extracted, cooled in the spent fuel pool for the necessary period of time, and separated using a dedicated cutting device (POMA) installed in the transport container pit.

Post-irradiation examinations (PIE) are being conducted at the hot-cell laboratories of UJV Řež and the Research Centre Řež. The analyses include:

- Neutron fluence determination via activation monitor activity;

- Dimensional measurements and irradiation growth evaluation;
- Transmission electron microscopy (TEM);
- Metallographic analysis and micro hardness testing.

Irradiation of all samples has now been completed. Growth measurements from the first four batches indicate that multicomponent Zr-Nb-Sn-Fe alloys exhibit higher irradiation-induced growth compared to binary Zr-Nb-(Fe, O) systems at equivalent fluence levels. These findings provide valuable insight into the microstructural mechanisms influencing dimensional stability in zirconium cladding materials used in VVER reactors.

Building on this experience, a follow-up program MCA-ATF within the OECD/NEA/FIDES II Framework has been initiated. Its objective is to evaluate the performance of claddings with different Accident - Tolerant Fuel (ATF) coatings under irradiation in the VVER-1000 reactor at Temelin NPP, followed by comprehensive post-irradiation examinations.

Keywords: Zr-alloys, irradiation, fuel performance, PIE, material research

FRAMATOME – IMPROVEMENT OF FUEL PRODUCT TH PERFORMANCE

Peter Pohl[1]*, M. Rehm[1], K. Marcinkiewicz[1], H. Gabriel[1], Fabien Filhol[2]

[1]Framatome GmbH, Erlangen, Germany
[2]Framtome SAS, Lyon, France
E-mail of the Corresponding Author: peter.pohl@framatome.com

Abstract

Framatome continuously improves the fuel product thermal-hydraulics (TH) performance thanks to its capacity to test new designs in its own loops and by high-fidelity simulation tools.

For thermal-hydraulic (TH) testing Framatome maintains its own multifunctional TH loop at Karlstein, Germany, the KArlstein Thermal-HYdraulic test facility, also known as KATHY [1,2]. It provides, among other types of measurements (pressure drop tests, void fraction measurements, stability tests and transient tests) the capability of CHF testing of PWR, BWR and also VVER fuel assembly designs. The facility started up in 1986 and has undergone an extended qualification for PWR CHF measurements through extensive benchmarks by comparisons, to the formerly used test loops at OMEGA (Grenoble, France) and the CU HTRF (Columbia University, USA) [1]. Since 2019 also CHF measurements are planned and performed for VVER-1000 and VVER-440 designs. These CHF tests are the basis for recently developed VVER non-mixing and mixing grid designs.

For designing new spacer designs, Framatome employs high-fidelity tools such as computational fluid dynamics (CFD) to support and verify developments and validate robust design variants. The design simulation is performed for early design assessment, extracting local subchannel flow information to enhance subchannel modelling, and intermediate validation of measured quantities to ensure alignment between testing and simulation results. These methodologies are supported by a comprehensive validation database covering various reactor types and scales.

Framatome's enhanced CFD methodologies are based on a large validation data base including pressure loss calculations of PWR/BWR/VVER products, several fuel assembly components, different scales (5x5 and 17x17, quadratic and hexagonal), and a large range of applicability regarding Reynolds numbers. It is enhanced by high-fidelity 2D and 3D data obtained from highly innovative measurement techniques such as Magnetic Resonance Imaging, as exemplarily shown for velocity and turbulence in [3] or particle deposition at filters in [4].

Framatome is consistently working on incorporating modern and best-in-class statistical methods [5]. As a standard Framatome develops its own specific CHF correlations based on CHF measurements and its proven development processes. These development processes are enhanced by state-of-the art machine learning methods resulting in improved statistical models and better understanding of the underlying database. Framatome is working with the broad scientific community, for example within the OECD CHF ML and AI benchmark project, on turning the idea of a functional, machine learning based CHF Correlation into reality, resulting in safer and more precise CHF prediction.

Keywords: CHF, KATHY, CFD

References:

1.C. Herer, A. Beisiegel, P. Imbert, D. Farnsworth, F. Burtak, "Comparison of PWR Fuel Assembly CHF tests obtained at three different Test Facilities", 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Avignon, 2-6 October 2005

VALIDATION OF FRAMATOME SUBCHANNEL CODE COBRA-FLX FOR VVER FUEL

Michael Brunmayr[1]*, Mira Pashtrapanska[1], Harry Gabriel[1]

[1]*Framatome GmbH*
E-mail of the Corresponding Author: michael.brunmayr@framatome.com

Abstract

In 2019, Framatome launched the development of a Fuel Assembly (FA) for VVER-1000 with a strong focus on safety and reliability, while finding the best balance between performance and robustness. In support of FA design, licensing, and performance in operation, Framatome qualifies a comprehensive, state-of-the-art VVER toolchain, following a dedicated VVER Codes & Methods Strategy.

A cornerstone in this toolchain is COBRA-FLX, Framatome's standard subchannel code used for PWR thermal-hydraulic core and fuel assembly design, Critical Heat Flux (CHF) correlation developments and accident analyses. Globally, it is accepted by safety authorities for square lattice PWR fuel.

To substantiate the adequacy of COBRA-FLX for the hexagonal geometry of VVER-1000 fuel assemblies, a benchmark was performed between COBRA-FLX and the SUBCAL code with UJP Praha in 2022. SUBCAL is UJP Praha's subchannel code and it is licensed and applied in the Czech Republic. The presentation shows calculation results from the benchmark which demonstrate adequacy for VVER application.

As of summer 2025, Framatome's CHF test program for VVER-1000 is in its final phase to support a Lead Fuel Assembly (LFA) project with newly developed CHF correlations for both spacer and mixing grids. In this context COBRA-FLX is utilized to form the basis for the CHF correlation development. The derived correlations show excellent agreement with measured data.

Finally, an outlook of COBRA-FLX application for Framatome's recently started development of a FA design for VVER-

440 is given: In late 2026, Framatome is going to provide a preliminary CHF correlation to partners gathered in the EU-funded SAVE project. With this CHF correlation it will be possible to assess, with locally accepted subchannel and system codes, the thermal-hydraulic fuel performance in VVER-440 reactors operating in the EU and Ukraine. Also, in this FA development project, COBRA-FLX forms the backbone in the subareas of CHF test data evaluation while the code is continuously benchmarked to VVER-440 relevant open-source experimental data.

Acknowledgment:
The CHF test program for Framatome's VVER-1000 FA design is partly funded by the German Federal Ministry for the Environment, Climate Action, Nature Conservation and Nuclear Safety under research project "Open KHB-W", grant number FKZ 1501637.

The European Union is partly funding the Framatome VVER-440 FA development by SAVE, a new Innovative Action under the Euratom Research and Training Program.

Keywords: Subchannel, COBRA-FLX, SUBCAL, VVER, CHF

DIGITAL IMAGE PROCESSING FROM NUCLEAR FUEL INSPECTIONS IN CVR

Kopeć M.[1]*, Blažek J.[1], Knotek J.[1], Pašta O.[1]

[1]*Centrum výzkumu Řež s.r.o., Hlavní 130, Řež, 250 68 Husinec, Czech Republic*
E-mail of the Corresponding Author: marcin.kopec@cvrez.cz

Abstract

Nuclear fuel inspection is a critical component in ensuring the safety, reliability, and efficiency of nuclear reactors. The fuel used in reactors is subjected to extreme physical and chemical conditions, including high radiation, elevated temperatures, and significant pressure. Inspection—both prior to and following irradiation—is essential for detecting manufacturing and working defects, monitoring degradation, and verifying fuel integrity. These efforts help prevent failures that could compromise reactor safety or cause costly downtime.

Post-irradiation inspections, in particular, assess structural changes, corrosion, and overall fuel performance, providing valuable insights for optimizing future fuel designs. The standard approach to nuclear fuel inspection incorporates visual, dimensional, and non-destructive testing methods to support safe reactor operation. Pre-irradiation inspections verify manufacturing quality, while post-irradiation evaluations assess the fuel's in-reactor performance and integrity. Emerging automation technologies—such as robotic scanners and AI-driven imaging—are improving inspection accuracy and reducing radiation exposure for human operators. Operators who manually screen inspection videos in real time—often for extended periods—remain the most error-prone element of the process. Although current technology is gradually minimizing the potential for human mistakes, this challenge persists.

To address this, the Digital Image Processing (DIP) program at CVR has developed a novel approach that transforms the time-domain video data from nuclear fuel inspections into the spatial domain,

producing a One-Image Overview (OIO). This single image preserves the full visual context of the original video and enables advanced photometric analysis. Through precise image registration and object recognition within the video, the system enables the application of advanced tools such as artificial intelligence, aging process tracking (e.g., oxidation mapping), and automated anomaly detection. These capabilities are continuously refined within the DIP program and have already been deployed in operational inspections.

In practice, the system functions as a digital assistant, highlighting findings, anomalies, or deviations for the human operator. The operator's role remains essential—evaluating the severity of findings based on experience and expertise, and assuming responsibility for the final inspection outcome. The CVR system can be tailored to incorporate local data, including specific fuel designs, power plant types, and inspection equipment parameters. It also integrates operator insights based on historical fuel behavior. The system includes a predefined reporting structure, which can be adapted to meet utility or operator-specific standards. Now in its seventh year of development, the system has been validated on six different fuel designs across various nuclear power plant units and inspection platforms. It is fully equipped to support full-face inspections, including both single- and multi-face configurations, with parallel processing of multi-string video data.

Keywords: Nuclear fuel, fuel inspection image processing, fuel evaluation

INSPECTIONS AND MEASUREMENTS ON FUEL ASSEMBLIES

Ch. WENDLAND[1]*, J. MICHNA[1], S. GARNIER[2], A. GUEDICHI[2], V. DUNN[3], J. OLIVER[3]

[1]Framatome GmbH, Paul-Gossen-Str. 100, 91052 Erlangen – Germany
[2]Framatome SAS, 2 rue professeur Jean Bernard, 69007 Lyon – France
[3]Framatome Inc., 3315 Old Forest Road, 24501 LYNCHBURG VA – United States
E-mail of the Corresponding Author: Juergen.michna@framatome.com

Abstract

Ensuring the structural integrity and operational reliability of nuclear fuel assemblies is a fundamental aspect of nuclear power plant safety and performance. During routine maintenance periods, particularly during refuelling outages, fuel assemblies are subjected to a range of inspections and measurements designed to detect any signs of degradation, mechanical deformation, or other anomalies that could compromise reactor operation.

These evaluations typically employ advanced non-destructive techniques, including high-resolution underwater visual inspections using remotely operated cameras and manipulators. Such inspections enable the identification and documentation of surface defects, corrosion, fretting, wear, and deformation affecting fuel rods, spacer grids, and end fittings. In addition to visual assessments, supplementary methods such as eddy current or ultrasonic testing may be utilized to further investigate the condition of cladding, providing a comprehensive understanding of the assembly's physical state.

Dimensional and mechanical measurements are conducted using precision underwater metrology equipment. Parameters such as overall length of fuel assembly and fuel rods, bow, and grid-width are carefully recorded to monitor changes resulting from irradiation, thermal expansion, or mechanical stresses. These measurements are critical for detecting deviations from design specifications, assessing

the impact of operational conditions, and predicting future behaviour of the assemblies within the reactor core.

The systematic application of these inspection and measurement techniques supports design licensing, informed maintenance and replacement decisions, contributes to the optimization of fuel management strategies and ensures early detection of potential issues. Continuous advancements in inspection technologies and measurement methodologies further enhance the accuracy, efficiency, and reliability of fuel assembly evaluations, thereby reinforcing the safety and economic performance of nuclear power plants.

This presentation will provide a detailed overview of the current practices, technologies, and recent developments in the inspection and measurement of fuel assemblies, highlighting their significance for reactor safety, fuel performance and operational excellence.

Keywords: Fuel Assembly, Inspection, Measurement

CMS5 BENCHMARKING AGAINST A VVER-1000 PLANT DATA

Emiliya L. Georgieva[1]* and Tamer Bahadir[2]

[1]Studsvik Scandpower GmbH
Rathausallee 28, 22846 Norderstedt, Germany
[2]Studsvik Scandpower Inc.
1070 Riverwalk Dr., Suite 150, Idaho Falls, ID 82402, USA
E-mail of the Corresponding Author: emiliya.georgieva@studsvik.com

Abstract

Studsvik Scandpower is continuously developing a comprehensive Core Management System (CMS5) for production-level efficient design, analysis, and optimization of light water reactors, which covers the complete nuclear fuel life cycle. It includes CASMO5 advanced lattice physics code for the generation of homogenized data for downstream use by the SIMULATE5 advanced three- dimensional nodal simulator.

Initially developed for light water reactors with square lattice designs, CMS5 was extended in 2018 to support hexagonal geometries used in VVER-1000 and VVER-1200 reactors, followed by the addition of VVER-440 capability in 2022. The benchmarking of the code system against numerical problems as well as core design calculations and plant measurements is an essential part of the extension efforts.

This work summarizes the results of CMS5 benchmarking against a VVER-1000 plant data. Nine subsequent fuel cycles of operation are covered including one cycle with mixed core loading of TVEL and Westinghouse fuel designs. The following comparisons are performed: low power physics tests (such as critical boron, isothermal temperature coefficient, SCRAM worth) and full-power boron letdown (boric acid concentration vs cycle exposure). The results confirm the accuracy and the applicability of CMS5 for core design and operational support of VVER-1000 reactors.

Keywords: VVER, CMS5, CASMO5, SIMULATE5

WESTINGHOUSE FUEL THERMAL-HYDRAULIC TESTING CAPABILITIES

Fredrik Waldemarsson[1]*, Carrie Wood[2]

[1]*Westinghouse Electric Sweden*
[2]*Westinghouse Electric Company*
E-mail of the Corresponding Author: waldemf@westinghouse.com

Abstract

Light Water Reactors (LWRs) rely on complex core thermal-hydraulic behavior to ensure safe and efficient operation. As advanced fuel designs are developed to support more demanding reactor operation, advanced modeling and rigorous testing are essential to optimize performance, address emerging challenges and meet regulatory requirements. This paper focuses on the testing aspect and the capabilities currently within Westinghouse to support these developments.

The Westinghouse fuel thermal hydraulic laboratory located in Västerås Sweden has been operated since the 1960's and currently houses five test facilities used for assessing Boiling Water Reactor (BWR), Pressurized Water Reactor (PWR) and Vodo-Vodyanoi Enyergeticheskiy Reactor (VVER) fuel thermal hydraulic performance. The ODEN and FRIGG loops are Westinghouse facilities used for primarily determining fuel critical heat flux for PWR/VVER and BWR fuel designs respectively. Both loops operate at nominal and off-nominal reactor conditions for their respective intended fuel design application. The EMBLA and FRODE loops are used for assessing, among other things, PWR/VVER and BWR fuel assembly and component pressure drop. Both loops operate at or close to reactor representative Reynolds values. The BURE loop is used for assessing long term fretting in BWR fuel. The loop is run at BWR reactor conditions using full BWR assemblies containing natural uranium.

Westinghouse has another PWR fuel and component test laboratory located in Columbia, South Carolina, part of the Columbia Fuel Fabrication Facility (CFFF). This lab was established in the late 1980's from loops and test components originally at the Westinghouse Electric Forest Hills facility in Pennsylvania. The lab contains mechanical test rigs for fuel component and fuel assembly testing as well as three thermal-hydraulic test loops. The Fuel Assembly Compatibility Test System (FACTS) loop, used to gather pressure drop data as well as vibration data for single, full-size fuel assemblies. The Vibration Investigation and Pressure Drop Experimental Research (VIPER) loop, used for thermal-hydraulic testing in either a single or dual PWR or VVER fuel assembly configuration to demonstrate grid-to-rod fretting performance, hydraulic resistance as well as the vibration characteristics of the fuel assemblies and fuel rods based on flow at ~500,000 Reynolds number. The Vibration Investigation of Small-scale Test Assemblies (VISTA) loop, often used for small-scale hydraulic testing of fuel assemblies for proposed fuel design changes, prior to investing in FACTS or VIPER testing.

Keywords: BWR, PWR, VVER, CHF, Thermal-Hydraulics

ANALYSIS OF MAIN STEAM LINE BREAK AT HOT ZERO POWER USING ATHLET, COCOSYS and DYN3D

Nadejda Rijova[1]*, Alexander Yordanov[1]

[1]*Enpro Consult, Sofia*
E-mail of the Corresponding Author: nrijova@enproco.com

Abstract

Main Steam Line Break at Hot Zero Power (MSLB at HZP) is a design basis accident for VVER-1000 reactors. This accident scenario involves a significant cooling of the affected loop and the corresponding section of the reactor core. As a result, return to power occurs in the cooled-down part of the core due to negative moderator temperature feedback. The steam released into the containment increases the pressure within it, leading to the initiation of safety signals that influence the behavior of both the primary and secondary circuit equipment. Consequently, the event encompasses complex thermal-hydraulic and 3D neutron kinetics phenomena that must be addressed in the analysis.

Three distinct approaches were employed in this investigation to analyze MSLB at HZP:

- **ATHLET-COCOSYS Coupled Calculation with Quasi Point Kinetics Model**: This method enables the calculation of power in each core section using reactivity coefficients and average moderator and fuel temperatures by sections.
- **Hybrid ATHLET-COCOSYS/DYN3D Approach**: This approach begins with an ATHLET-COCOSYS calculation similar to the first method. Subsequently, the results for reactor pressure, reactor flow rate, and temperatures at the reactor inlet for each of the four sections are utilized as boundary conditions in a DYN3D calculation. This allows for determining the power of each fuel assembly, considering its position in the core, as well as its enrichment and burnup.

- **Fully Coupled ATHLET-COCOSYS-DYN3D Calculation**: This approach integrates all three codes to provide a comprehensive simulation of the event.

The results obtained from these three approaches are compared, and the advantages and disadvantages of each method are discussed.

Acknowledgment: Dr.Yurii Bilodid, Helmholtz-Zentrum Dresden-Rossendorf, Reactor Safety Department

Keywords: MSLB at HZP, VVER-1000, point kinetics, 3D kinetics, ATHLET-COCOSYS-DYN3D coupling

SESSION 5 FUEL SAFETY AND QA

FUEL PERFORMANCE ANALYSIS CAPABILITY DEVELOPMENT IN SURO

Katalin Kulacsy[1]*

[1]*National Radiation Protection Institute (SÚRO)*
E-mail of the Corresponding Author: katalin.kulacsy@suro.cz

Abstract

The Czech Republic is undergoing a very intense period of nuclear development. Within a few years four new fuel types have been or will be licensed, two large new commercial units have been contracted and siting and preparations have started for several small modular reactors.

The Czech National Radiation Protection Institute (SURO) is the TSO (technical support organisation) of the regulatory body (State Office for Nuclear Safety). Its tasks consist of the assessment of licensing documentations and of performing confirmatory analyses, the latter being done by the Research and Analysis of Nuclear Safety Department. The department has a developing expertise in the fields of neutronics, thermo-hydraulics, severe accidents, structural mechanics and fuel performance.

Over the last two years fuel performance analyses in SURO have been put on a new footing. A new methodology is under development, covering a wider and wider range of conditions, large-scale batch analyses are being implemented, and BEPU methods are being introduced. In the paper this development is presented, including details of the methodology and some results.

Acknowledgment:
The work was finance by the State Office for Nuclear Safety.

Keywords: fuel performance analysis, methodology

DEVELOPMENT OF VVER-1000 FUEL ASSEMBLY SHIPPING SOLUTION BY FRAMATOME: PROJECT OBJECTIVES AND STATUS

Dounia Lhachemi[1]*, Maximilian Stark, Johannes Von Morstein[2]

[1] rue Professeur Jean Bernard, 69007 Lyon – FRANCE
[2]Am Seitenkanal 1, 49811 Lingen – GERMANY
E-mail of the Corresponding Author: Dounia.lhachemi@framatome.com

Abstract

Framatome, a key player in the nuclear sector, is advancing the development of new fuel assemblies for VVER-1000 and VVER-440 reactors. The company aims to provide European utilities with diverse and fully European-controlled VVER fuel solutions. A crucial aspect of this strategy is the development of safe and practical shipment solutions.

For the Framatome VVER-1000 project, the selected container for transporting fuel assemblies is a modified version of the existing PWR fuel transport container ANF-18, which has demonstrated safety and reliability over the past 25 years. This container was licensed in the early 2000s and can transport two fuel assemblies. Notably, it offers excellent protection for products through effective absorption of vibrations and external forces, using rubber support system.

Since the container has only been licensed for square fuel elements in various countries, a licensing extension and new drop tests series according are necessary to validate its outstanding properties for VVER-1000 fuel assemblies. This adaptation requires a new licensing test campaign to comply with both international standards, such as those set by the IAEA, and relevant national regulations.

IAEA physical test campaign includes a comprehensive evaluation using specially manufactured test samples, adhering to the latest technical standards in collaboration with BAM (Bundesanstalt für Materialforschung und -prüfung). The testing program encompasses:

- Finite Element (FE) calculations for container behavior and drop angle orientation
- Drop tests (1.2 m, 9 m, and spike drop tests)
- Heating tests (exposure to high temperatures for 30 minutes at 800°C)
- Helium leakage tests (assessing the tightness of the fuel rods within a test chamber)
- Puncture tests (drilling each rod and checking for helium escape)

Framatome's licensing concept for this upgraded container, compatible with VVER-1000 fuel assemblies, has been approved by safety authorities. The IAEA tests, involving a test container that includes a fuel assembly and counterweight, are currently ongoing, having started in July 2025.

In parallel, and immediately following the IAEA test series, Framatome will conduct new Criticality Safety Analysis and an upgraded Safety Analysis Report (SAR). This report will detail the design, safety features, and analyses demonstrating compliance with regulatory requirements.

Keywords: FA bow, FA distortion, Operating experience, FA performance

SESSION 6 SPENT FUEL PERFORMANCE AND MANAGEMENT

FRAMATOME'S PERFORMANCE IN SPIZWURZ CREEP BENCHMARK

Radan Sedláček [1]*, Dietmar Deuble [1]

[1]*Framatome GmbH, Paul-Gossen-Str. 100, 91052 Erlangen, Germany*
E-mail of the Corresponding Author: radan.sedlacek@framatome.com

Abstract

Framatome GmbH participated in the SPIZWURZ creep benchmark with the code CSAS that is normally used for dry storage Fuel Rod Design (FRD) calculations. The computer code CSAS conservatively calculates the dry storage design relevant quantities, namely the maximum cladding stress and maximum cladding creep strain, under given axial temperature profile as a function of time. The input to CSAS is the end-of-life (EOL) state of the fuel rod as calculated by the FRD code CARO-E: amount of gas, rod dimensions, rod free volumes, oxide thickness, and fast fluence. The program calculates the cladding tangential creep strain during storage conservatively by choosing an unfavorable combination of conditions (temperature, fluence, oxide thickness).

For the benchmark calculations, a more flexible R&D version of CSAS was used. It calculates the creep strain in several axial zones and is capable of dealing with the specific benchmark conditions, i.e. the mutually independent pressure and temperature input.

The cladding creep model used in CSAS (both FRD and R&D versions) is implemented in CSAS with proprietary parameter values that were calibrated and validated on Framatome own cladding

materials: Zry-4 and Duplex, Zry-2 RX, **M5Framatome**[1]. The calibration and validation database consists of about one hundred creep tests per material (irradiated and non-irradiated).

The results of the SPIZWURZ benchmark confirm the calibration and validation of the Framatome cladding creep model for dry storage. The effects of stress and temperature on creep strain are well reproduced.

Keywords: fuel rod design, dry storage, cladding creep

SPENT NUCLEAR FUEL MANAGEMENT OF UKRAINIAN NUCLEAR OPERATOR

Roman Glushenkov[1]*

[1]JSC "NNEGC "Energoatom"
01032, Ukraine, Kyiv, str. Nazarivska, 3
E-mail of the Corresponding Author: r.glushenkov@direkcy.atom.gov.ua

Abstract

The way a country manages its spent nuclear fuel is primarily determined by the type of nuclear fuel cycle it employs, which are broadly categorized into two main approaches: open (or once-through) and closed fuel cycles. There are several options for the final stage of the fuel cycle (spent fuel management):

- Spent fuel reprocessing: this process involves chemically separating reusable uranium and plutonium from spent fuel so they can be returned to the fuel cycle. The highly radioactive fission products are isolated as high-level waste, which then requires long-term storage.

---

[1] M5Framatome is a trademark or registered trademark of Framatome or its affiliates, in the USA and other countries.

- Deferred solution: This approach involves storing spent fuel in temporary facilities, either at reactor sites or at centralized interim storage facilities, without immediate reprocessing or permanent disposal.
- Direct disposal: this strategy involves preparing spent fuel (after optional interim storage) for direct emplacement in deep geological repositories without prior reprocessing.

Ukraine's current spent nuclear fuel management scheme is focused on deferred solution (interim dry storage), aiming for long-term storage within the country and reducing its historical reliance on russia for reprocessing and storage. Key elements of Ukraine's strategy include:

Initial storage in on-site spent fuel pools (wet storage), which allows sufficient time for the fuel to cool and its residual heat and radioactivity to decay to levels safe for subsequent handling and transport. Once sufficiently cooled, the spent fuel is then transferred to a dry spent fuel storage facility.

The Centralized Spent Nuclear Fuel Storage Facility, located in the Chernobyl Exclusion Zone, which receives spent nuclear fuel from Rivne, Khmelnytskyi, and Pivdennoukrainsk NPPs.

The On-site Dry Spent Fuel Storage Facility at Zaporizhzhia NPP, which stores spent nuclear fuel generated by Zaporizhzhia NPP itself.

Energoatom's spent nuclear fuel management represents a significant move towards self-sufficiency and strategic independence within a challenging geopolitical environment, while laying the groundwork for eventual permanent disposal solutions.

Keywords: spent nuclear fuel management, interim dry storage, deferred solution

STUDYING THE INTEGRITY OF SNF ROD CLADDING AT VARIOUS STAGES OF DRY STORAGE TECHNOLOGY IMPLEMENTATION

Valeriy Zuyok[1]*, Mykhaylo Tretyakov[1] , Roman Rud[1] , Volodymyr Zigunov[1] , Yana Kushtym[1] , Vadym Hrudnytskyy [1]

[1] *National Science Center "Kharkiv Institute of Physics and Technology" of the NAS of Ukraine, Ukraine, Kharkiv city, 61108*
E-mail of the Corresponding author: valeriyz@kipt.kharkov.ua

Abstract

The Dry Storage of Spent Nuclear Fuel (DSSNF) system at the Zaporizhzhya NPP (Ukraine) is designed to store leak-tight spent fuel assemblies, in which the fuel rod claddings are the first protective barrier against the release of radioactive materials into the environment. The project provides that during the entire design life of interim dry storage, which is 50 years, no more than 1% of fuel rod cladding is expected to be leaky. Ensuring the integrity of fuel rod claddings in dry storage is based on compliance with a number of requirements (criteria) specified by the fuel manufacturer. The criteria are established with regard to operation experience, as well as the results of experimental tests and research on fuel rod cladding material under conditions that do not always correspond to the effects that occur during the implementation of dry storage technology.

The study considers temperature regimes specific for the vacuum drying stage (up to 436 °C) and filling the basket with helium, as well as reaching a steady state mode of long-term storage under normal conditions with an initial temperature of 350 °C, storage under abnormal conditions (+11 °C) and under accident conditions (+19 °C), which are typical for the ZNPP DSSNF.

Calculations revealed that the maximum stresses typical for rods with maximum GFP pressure (5.0 MPa) at the vacuum drying stage (436 °C) are 67.7 MPa and 84.7 MPa for claddings made of E110 and ZIRLO alloys, respectively. These values do not exceed the strength limit of the irradiated claddings, taking into account the level

of their degradation and temperature effects. The maximum circumferential deformation of the cladding at the end of a 50-year storage period, under normal, abnormal, and accident conditions, does not exceed the design criteria established for each fuel design (2% for TVS-M, TVSA, and 1% for RWFA). The maximum reduction in the fuel pellet-cladding gap is typical for the vacuum drying stage for cladding with lower GFP pressure and does not result in mechanical interaction between them or the occurrence of additional stresses in the cladding. The maximum stress intensity factor in the vicinity of the maximum postulated crack in the cladding remains below the conservative limit factor for the onset of delayed hydride cracking (5.0 $MPa \cdot m^{1/2}$).

At all stages of dry storage technology implementation, the maximum diameter of the fuel rod cladding remains lower than that of unirradiated fuel, and its length increase due to abnormal creep and thermal expansion will not result in axial compression of the fuel rod between the support plates of each design of stored SNF.

The results of calculations and analytical studies enabled to draw a general conclusion: during the implementation of dry storage technology, the above characteristics of fuel rods with E110 and ZIRLO alloys claddings remain virtually unchanged compared to their condition prior to placing SNF at dry storage, and therefore do not exceed the design and additional criteria for maintaining the integrity of TVS-M, TVSA, and RWFA fuel rod claddings integrity at all stages of dry storage technology at the ZNPP DSSNF.

Keywords: dry storage, fuel rod, spent nuclear fuel, E110, ZIRLO

DEVELOPMENT OF NEUTRON-ABSORBING MATERIALS AND RCCA WITH INCREASED PERFORMANCE FOR VVER-1000 REACTOR

Volodymyr Krasnorutskyy, Viktor Grytsyna, Ihor Chernov, Volodymyr Zigunov, Valeriy Zuyok, Anton Kushtym

The Science and Technical Establishment "Nuclear Fuel Cycle" of National Science Center Kharkov Institute of Physics and Technology, 1, Akademichna str., Kharkiv, 61108, Ukraine

E-mail of the Corresponding Author: viktorgrytsyna56@gmail.com

Abstract

NSC KIPT has an extensive experience in the development of neutron-absorbing materials for use in absorber rods (AR) for reactors of various types and purposes. The developments currently under way are largely due to the situation that Ukraine had to abandon supplying reactor system components from Russia, including nuclear fuel and RCCA for reactors operating in Ukraine. Within the shortest possible time, NFC STE NSC KIPT specialists developed an alternative to absorber rods previously supplied from Russia, using boron carbide and dysprosium titanate for VVER-1000 reactors.

In the range of neutron-absorbing materials elaborated for future use, boron carbide and dysprosium titanate are used most frequently.

Due to their nuclear-physical characteristics, boron carbide and dysprosium titanate are widely used as neutron-absorbing materials in thermal reactors. In VVER-1000 absorber rods these materials are used in powder form. Modern ideas about increasing service life of VVER-1000 absorber rods and RCCA are based on their use in powder form, which allows increasing the number of neutron absorbers in products, and reducing their influence on cladding degradation during operation.

In order to improve the performance characteristics and diversify the supplies of RCCA for European power units operating VVER-1000 reactors, an RCCA design was developed based on selection and substantiation of advanced neutron-absorbing and structural materials, development of absorber rod and RCCA manufacturing and assembly technology, conduct of comprehensive out-of-pile testing and calculations to substantiate RCCA operability.

The paper presents some results of the development of a standard powder-type absorber rod for VVER-1000 reactor, namely, absorber rod manufacturing scheme, characteristics of dysprosium titanate, absorber rod quality control methods and characteristics of absorber rod pilot batches.

The results of the development of advanced absorber rods with a neutron-absorbing core based on boron carbide, dysprosium titanate and dysprosium hafnate are presented. Absorber rod drawings, main requirements for neutron absorbing material pellets and absorber rods, results of out-of pile tests and investigations, including corrosion resistance and compatibility of neutron absorbing materials and absorber rod cladding are presented.

Keywords: absorber rod, dysprosium titanate, dysprosium hafnate.

POSTER SESSION

EXTENDED VALIDATION OF TRANSURANUS CODE FOR THE PWR CLADDING TYPES CONDUCTED AT UJV REZ

Vítězslav Matocha[1]*, Martin Dostál[1], Jan Klouzal[1]

[1]ÚJV Řež, a. s.
E-mail of the Corresponding Author: vitezslav.matocha@ujv.cz

Abstract

ÚJV Řež provides an analytical support to utilities in many fields including nuclear fuel performance evaluation and safety analyses. TRANSURANUS code is used by ÚJV for the analysis of the fuel performance for Temelín (VVER-1000) and Dukovany (VVER-440) NPPs. Its predictions are systematically compared with the recent experimental data in order to validate the applicability of the code to the fuel designs and operating conditions of Czech NPPs. The diversification of fuel supplies lead to recent introduction of new nuclear fuel types with Opt. ZILRO cladding in both NPPs which meant the need to expand the validation base.

The poster presents the evaluation of TRANSURANUS performance by comparison with relevant cases, i.e. PWR claddings like Opt. ZIRLO and M5 (and also Zry-4 and ZIRLO). The cases include IFPE datasets, data from international projects (like FIDES-II, HRP, SCIP, CIP) and also other not publicly available data and cover normal operation as well as AOOs and accident conditions (LOCA, RIA). The standard material properties already presented in the code or newly implemented material properties are used in the calculations of different cases to demonstrate reasonable agreement of measured and calculated values of important parameters. The examples of such agreements are shown in the form of graphs with predicted vs. measured values for selected parameters (cladding outer diameter, oxide thickness, fuel rod elongation etc.). The overall agreement is

acceptable, but there is still room for improvement, e.g. in the fuel pellet transient FGR and swelling.

Keywords: fuel, modelling, validation, transuranus

References:

1. K. Lassmann. TRANSURANUS: a fuel rod analysis code ready for use, Journal of
Nuclear Materials, Vol. 188, pp. 295-302, 1992.
2. C. Györi et al. Development, Validation and Application of the TRANSURANUS Code for Westinghouse Fuel Designs. NENE 2017, September 11-14, Bled, Slovenia.
3. V. Matocha, J. Klouzal, M. Dostál. Complex analysis of Halden Reactor Project experimental results in support of fuel rod safety and reliability assessments. Proceedings of 13th International Conference on WWER Fuel Performance, Modelling and Experimental Support, 15-21 September 2019, Sol Nessebar Resort, Nessebar, Bulgaria.
4. J. Karlsson et al. Modelling Out-of-Pile LOCA Tests on High Burnup Fuel Rods. Results of the fourth SCIP Modelling Workshop. Proceedings of TopFuel 2018, Prague, 30 Sept-4 Oct, 2018.
5. S. Seo et al. International fuel performance study of fresh fuel experiments for PCMI effects during RIA experiments, Nuclear Engineering and Design 430 (2024).

ADVANCED DIGITAL IMAGE PROCESSING FOR MATERIAL EVALUATION IN NUCLEAR APPLICATIONS

Pašta O[1]*., Knotek J.[1], Blažek J.[1], Kopeć M.[1]

[1]Centrum výzkumu Řež s.r.o., Hlavní 130, Řež, 250 68 Husinec, Czech Republic
E-mail of the Corresponding Author: ondrej.pasta@cvrez.cz

Abstract

The integration of digital image processing (DIP), neural networks, and artificial intelligence (AI) is reshaping the field of materials science by enabling detailed, reliable, and automated analysis of microscopic structures. In nuclear applications, these technologies have become essential tools—ranging from fuel assembly inspections to high-resolution microscopy—facilitating the extraction of meaningful data across multiple imaging modalities (SEM, TEM) and scales.

At Centrum výzkumu Řež (CVR), the implementation of DIP and AI has significantly improved workflow efficiency and data consistency. A particularly important use case is the identification and classification of microstructural precipitates, which play a key role in the mechanical behavior of materials. These include secondary phase particles (SPPs), typically originating from manufacturing and thermal treatment, and radiation-induced precipitates (RIPs), which form as a result of neutron irradiation and subsequent atomic diffusion. The morphology and spatial distribution of both types influence mechanical properties such as hardness, brittleness, and internal stress states. By optimizing alloy composition and processing conditions, SPP formation can be controlled, while RIP characterization is critical in post-irradiation examination (PIE).

DIP techniques are indispensable for segmenting and quantifying these features in SEM and TEM data, particularly when evaluating material changes before and after irradiation. Automated

analysis minimizes manual bias, reduces operator workload, and improves reproducibility of results.

Another key application area is the structural assessment of biological shielding concrete in nuclear reactors. This material is exposed to intense thermal and radiation stress, leading to degradation such as cracking at the interface between aggregates and the cement matrix. DIP allows for accurate detection and monitoring of these cracks and enables volumetric damage analysis. These results can be used to cross-validate traditional non-destructive testing methods like ultrasonic testing, thus supporting integrity assessments through non-invasive approaches.

In fuel inspections, Digital Image Processing (DIP) is used to analyze video data sequences and generate high-resolution images without any loss of context. Beyond inspections, DIP is also a powerful tool for processing images obtained from nuclear material research and development activities. When applied to data from hot cell operations, DIP can play a crucial role in Post-Irradiation Examination (PIE), such as in the analysis of cladding material tests or other structural components of nuclear reactors. In this context, DIP enhances the consistency, efficiency, and optimization of nuclear material data analysis workflows.

Overall, DIP technologies developed at CVR contribute directly to long-term operation (LTO) strategies by improving the reliability and speed of data evaluation in nuclear material monitoring. While these methods are rooted in nuclear R&D, their versatility extends far beyond: sectors requiring precise microstructural or surface analysis—such as construction or metallurgy—can also benefit from these advanced capabilities.

Keywords: material science, data analysis, PIE, precipitates, biological shielding

FROM OPEN TO CLOSED FUEL CYCLE – ADVANTAGES AND DRAWBACKS OF THE PROCESS

Mladen Mitev [1,2]*, Kalin Filipov[1]

[1]*Technical University - Sofia*
[2]*Institute for Nuclear Research and Nuclear Energy of the Bulgarian Academy of Sciences*

E-mail of the Corresponding Author: mrmitev@tu-sofia.bg

Abstract

Nuclear energy plays a key role in today's energy sector, providing a significant part of global electricity demand. It is characterised by low carbon intensity of electricity production, reliability and stability of energy supplies, and a positive impact on energy independence. Despite its advantages, nuclear energy also has drawbacks, one of the major challenges being the growing quantity of spent nuclear fuel (SNF).

Several concepts for final disposal exist, one of which is deep geological disposal. However, this approach faces difficulties related to site selection, design, and licensing, since the long-lived radioactivity of SNF requires geological repositories that must not need maintenance, must ensure isolation from the environment, and must remain secure for thousands of years.

The world's first deep geological repository for high-level radioactive waste is under construction in Olkiluoto, Finland, and is expected to start operation in 2025. Similar projects are advanced in Sweden and France. The high radiotoxicity levels and residual heat release require strict regulatory control and management of all radioactive waste for tens of thousands of years, which represents one of the most significant problems for nuclear energy development.

Gradually, the need for reprocessing SNF (closing the nuclear fuel cycle) is increasing. This will improve resource efficiency, while reducing the masses and volumes of spent nuclear fuel and radioactive waste, and shorten the storage periods. Anyway, also number of challenges, related to SNF manipulation and proliferation issues remain to be solved. These are discussed in this work to foster the sustainable closing of the nuclear fuel cycle.

Acknowledgment: This research was funded by the Ministry of Education and Science of the Republic of Bulgaria, through the National Program D01-99: Qualification improvement in the field of nuclear technologies and nuclear engineering.

Keywords: spent nuclear fuel, fuel cycle, fuel reprocessing, non-proliferation

STUDY OF FUEL BEHAVIOUR DURING LBLOCA WITH PARTIAL DAMAGING OF REACTOR CORE

Petya Vryashkova*[1], Petya Petrova[1], Pavlin Groudev[1]

[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, 72 Tsarigradsko chaussee Blvd, 1784 Sofia, Bulgaria
E-mail of the Corresponding Author: pivryashkova@inrne.bas.bg

Abstract

This study investigates the behaviour of nuclear fuel and oxidation phenomena during a Large Break Loss of Coolant Accident (LB LOCA) in combination with a Station Blackout (SBO) in a VVER-1000 reactor. The initiating event is a cold leg rupture of the Reactor Coolant System (RCS) with an inner diameter of 850 mm, occurring simultaneously with the total loss of both AC and DC power sources.

The scenario was simulated using the ASTEC computer code, version 2.2.0.1, specifically applied to the VVER-1000 design. The analysis considered a case including Severe Accident Management Guideline (SAMG) actions at Kozloduy Nuclear Power Plant (KNPP). Operator actions are modelled as core quenching by activating one high pressure pump and one low pressure pump at a core exit temperature of 980 °C, following the assumed injection failure at 650 °C.

The simulation results provide a description of the physical processes inside the core. Progressive uncovering of the fuel assemblies leads to rapid cladding temperature escalation, extensive zirconium-steam oxidation, and significant hydrogen generation. These oxidation-driven processes accelerate fuel cladding embrittlement and failure, causing early release of fission products and initiating partial core degradation.

The results highlight the crucial influence of oxidation on fuel integrity and accident progression. They also emphasize the

importance of timely SAMG actions in reducing hydrogen generation, partial core damage, and mitigating the severe consequences of LB LOCA + SBO conditions in VVER-1000 reactors.

Acknowledgment: This research was funded by the Ministry of Education and Science of the Republic of Bulgaria, through the National Program D01-99: Qualification improvement in the field of nuclear technologies and nuclear engineering.

Keywords: fuel behaviour, oxidation, accident progression, damage

MONTE CARLO-BASED STUDIES FOR RADIATION SAFETY IN THE DESIGN OF BGNCC

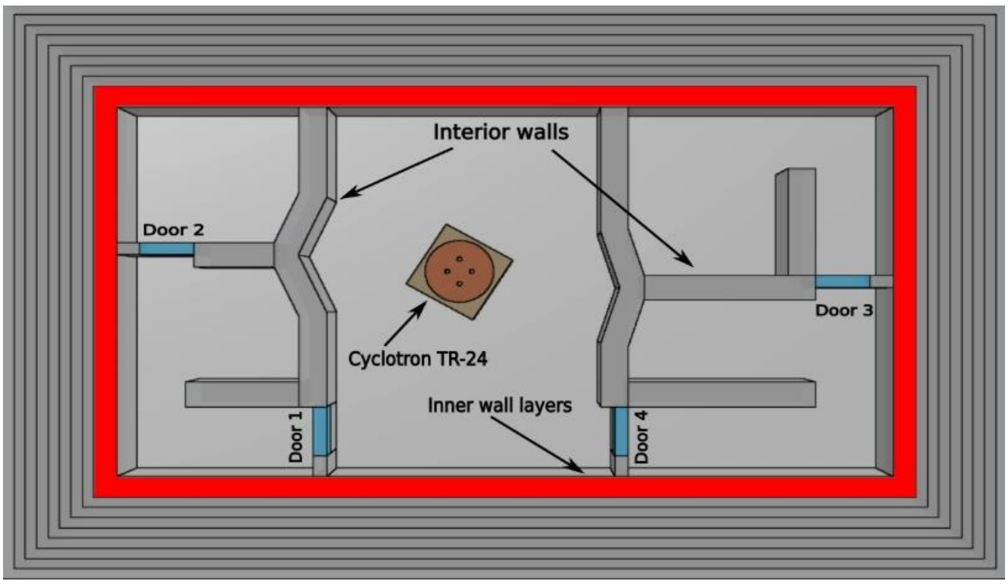
G.D. Dimitrova[1*], A. Demerdjiev[1], D. Tonev[1], N. Goutev[1], E. Geleva[1], V. Variyska[1], N.N. Petrov[1], B. Bonchev[1],

[1]Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, Tzarigradsko Chaussee 72, Sofia, Bulgaria

Abstract

This work focuses on the radiation safety of the planned Bulgarian National Cyclotron Centre. The facility will be dedicated to the production of medical radioisotopes and radiopharmaceuticals, with additional research and development activities in these fields. An intermediate-energy cyclotron TR-24 will be installed. For cyclotrons of this type, standard radiation protection typically involves a bunker with concrete walls 200–250 cm thick.

In this study, we present results obtained through Monte Carlo simulations performed with the FLUKA code [1–2]. We evaluated the distribution of radiation fields induced by beam losses in the cyclotron. To enhance the radiation safety of the facility, we investigated the replacement of parts of the bunker walls with low-activation concrete [3]. Our results indicate reduced dose rates inside the cyclotron bunker.



Horizontal view of 3D FLUKA Monte Carlo model of the cyclotron bunker.

Acknowledgment: This research was funded by the Ministry of Education and Science of the Republic of Bulgaria, through the National Program D01-99: Qualification improvement in the field of nuclear technologies and nuclear engineering.

Keywords: TR-24 cyclotron, Monte Carlo simulations, FLUKA, radiation protection

References:

1. A. Ferrari, P.R. Sala, A. Fassó, J. Ranft. FLUKA: A Multi-Particle Transport Code., CERN-2005-010, INFN-TC-05-11, SLAC-R-773, 2005.

2. T.T. Böhlen, F. Cerutti, M.P.W. Chin, A. Fassò, A. Ferrari, P.G. Ortega, A. Mairani, et al. The FLUKA Code: Developments and challenges for high energy and medical applications. *Nuclear Data Sheets* **2014**, 120, pp. 211-214.

3.Stichelbaut F., Pí erard J., Kupers L. Compositions of low activation concrete and use thereof. *Eropean Patent,* 3266754A1, 2018.

POST-FUKUSHIMA STRESS TESTS AT KOZLODUY NPP: ASSESSING RESILIENCE TO EXTREME CLIMATIC EVENTS

D. Tonev[1], E. Geleva[1*], N. Goutev[1], A. Demerdjiev[1], G. Dimitrova[1], V. Variyska[1], N.N. Petrov[1], B. Bonchev[1],

[1]*Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, 72 Tsarigradsko chaussee Blvd, 1784 Sofia, Bulgaria*

**E-mail of the Corresponding Author: elenag@inrne.bas.bg*

Abstract

Following the Fukushima accident, the European Commission mandated stress tests for all European nuclear power plants. The Institute for Nuclear Research and Nuclear Energy (INRNE) at the Bulgarian Academy of Sciences successfully led a safety reassessment of the Kozloduy Nuclear Power Plant (NPP) and its facilities against extreme climatic impacts.

The assessment covered all nuclear facilities at the site, analyzing the performance of key structures, systems, and components (SSCs). We evaluated the plant's design bases against severe conditions, including extreme temperatures, precipitation, winds, and low river levels. The results demonstrated that Kozloduy NPP's systems have sufficient resilience to withstand these extreme meteorological events and anticipated climate changes.

This project was unique as Kozloduy NPP was the only plant to use its national scientific expertise for the studies, a fact recognized by the EU. The INRNE report was accepted by the Bulgarian and European regulatory bodies, confirming the high qualifications of our scientists in addressing critical challenges for Bulgaria's economic development.

Acknowledgment: This research was funded by the Ministry of Education and Science of the Republic of Bulgaria, through the National Program D01-99: Qualification improvement in the field of nuclear technologies and nuclear engineering.

Keywords: Nuclear Safety, Stress Tests; Kozloduy NPP.

MAIN SPONSOR:

framatome

ORGANIZED BY:

50 INRNE 1972

SUPPORTED BY:

Westinghouse

KOZLODUY NPP

Studsvik