IAEA-TECDOC-2053

IAEA TECDOC SERIES

IAEA-TECDOC-2053

# Safety and Performance Aspects in the Development and Qualification of High Burnup Nuclear Fuels for Water Cooled Reactors

*Proceedings of a Technical Meeting*

IAEA
International Atomic Energy Agency

# IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

## IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available at the IAEA Internet site

www.iaea.org/resources/safety-standards

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

## RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

# SAFETY AND PERFORMANCE ASPECTS IN THE DEVELOPMENT AND QUALIFICATION OF HIGH BURNUP NUCLEAR FUELS FOR WATER COOLED REACTORS

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ANTIGUA AND BARBUDA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHAMAS
BAHRAIN
BANGLADESH
BARBADOS
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA, PLURINATIONAL STATE OF
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BRUNEI DARUSSALAM
BULGARIA
BURKINA FASO
BURUNDI
CABO VERDE
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
COMOROS
CONGO
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DJIBOUTI
DOMINICA
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ESWATINI
ETHIOPIA
FIJI
FINLAND
FRANCE
GABON
GAMBIA
GEORGIA
GERMANY
GHANA
GREECE
GRENADA
GUATEMALA
GUINEA
GUYANA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LAO PEOPLE'S DEMOCRATIC REPUBLIC
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS, KINGDOM OF THE
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORTH MACEDONIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PAPUA NEW GUINEA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
RWANDA
SAINT KITTS AND NEVIS
SAINT LUCIA
SAINT VINCENT AND THE GRENADINES
SAMOA
SAN MARINO
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
TOGO
TONGA
TRINIDAD AND TOBAGO
TUNISIA
TÜRKİYE
TURKMENISTAN
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VANUATU
VENEZUELA, BOLIVARIAN REPUBLIC OF
VIET NAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

IAEA-TECDOC-2053

# SAFETY AND PERFORMANCE ASPECTS IN THE DEVELOPMENT AND QUALIFICATION OF HIGH BURNUP NUCLEAR FUELS FOR WATER COOLED REACTORS

## PROCEEDINGS OF A TECHNICAL MEETING

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2024

For further information on this publication, please contact:

Safety Assessment Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
Email: Official.Mail@iaea.org


Printed by the IAEA in Austria
May 2024
https://doi.org/10.61092/iaea.hknc-9tzt

**IAEA Library Cataloguing in Publication Data**

Names: International Atomic Energy Agency.
Title: Safety and performance aspects in the development and qualification of high burnup nuclear fuels for water cooled reactors / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2024. | Series: IAEA TECDOC series, ISSN 1011-4289 ; no. 2053 | Includes bibliographical references.
Identifiers: IAEAL 24-01681 | ISBN 978-92-0-118524-2 (paperback : alk. paper) | ISBN 978-92-0-118624-9 (pdf)
Subjects: LCSH: Fuel burnup (Nuclear engineering). | Nuclear fuels. | Water cooled reactors —Safety measures.

# FOREWORD

IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants, published in 2019, provides recommendations on meeting the safety requirements for the design of the reactor core for nuclear power plants. It is mainly intended for reactor cores that use uranium dioxide fuels and plutonium blended with uranium dioxide fuel (mixed oxide fuel) with zirconium alloy cladding, with conventional (low) rates of discharge burnup (e.g. below a rod average burnup of 62 GWd/tU).

The development and qualification of nuclear fuels with higher discharge burnup (e.g. beyond 62 GWd/tU) are being pursued by many Member States in the framework of the development of advanced technologies for nuclear fuel (including accident tolerant and advanced technology fuels). In particular, higher rates of nuclear fuel discharge burnup are researched in association with advanced fuel and cladding materials, which might require higher fissile enrichment rates (in some cases exceeding 5%) in order to counterbalance increased parasitic neutron absorption in the cladding.

The physical phenomena specifically associated with operation at high burnup (e.g. modification of the microscopic structure and rim formation; enhanced fission gas release; possible fragmentation, relocation and dispersal of the nuclear fuel in accident conditions) are associated with challenges for safety that need to be addressed in the safety assessment. In the case of loss of coolant accidents and of reactivity initiated accidents, for example, the fragmented nature of the nuclear fuel can facilitate fuel dispersal outside the cladding opening after burst, hampering coolability of the subchannel and determining a release of source term in the primary system and — in the case of loss of coolant accidents — in the containment.

The IAEA organized a technical meeting in November 2022 to provide a platform for the exchange of information on how its Member States are addressing safety and technological challenges encountered in the development, qualification and licensing of high burnup nuclear fuels for water cooled reactors. The present publication summarizes the material presented at the technical meeting. It is expected to provide useful insights for a future revision of SSG-52, as industry trends strongly signal to the need to broaden the scope of this Safety Guide to include high burnup nuclear fuels.

The IAEA thanks the experts who contributed to this publication. The IAEA officers responsible for this publication were S. Massara and T. Veneau of the Division of Nuclear Installation Safety and K. Sim of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

# 1. INTRODUCTION

## 1.1. BACKGROUND

The increasing maturity of nuclear technology and progress in knowledge of nuclear fuel behaviour, through operational experience of commercial nuclear power plant (NPP) operation, supported by experiments and numerical modelling and simulation, have driven efforts by the nuclear industry worldwide towards an increase in the discharge burnup of nuclear fuel in water cooled reactors (WCR).

Nuclear fuel vendors and NPP operating organizations continue extending the discharge burnup of nuclear fuel assemblies within the operating envelope accepted by regulatory bodies. The current operational limit for the discharge burnup varies slightly among licensees, but in many Member States it corresponds to a rod average burnup of 62 gigawatt-day/tons of uranium (GWd/tU) (average fuel sub-assembly of 55 GWd/tU), or other equivalent criteria. Nuclear fuel vendors and NPP operating organizations might obtain economic benefits, combined with the strategic approach for the nuclear fuel cycle, from such high burnup (HBU) operation via, for example, longer reactor cycle operation, as well as purchase, transport and handling of fewer fuel assemblies for the core reloading.

Currently, increased rates of the fuel discharge burnup (e.g. beyond 62 GWd/tU) are also being pursued by Member States in the framework of the development of advanced technology for nuclear fuel (which include so-called Accident Tolerant and Advanced Technology Fuels, ATFs). In particular, higher rates of the nuclear fuel discharge burnup are researched in association with advanced fuel and cladding materials, which might involve higher fissile enrichment rates (in some cases exceeding 5%) in order to counterbalance increased parasitic neutron absorption in the cladding. Some nuclear fuel vendors and NPP operating organizations are even exploring the possibility to increase the operating envelope up to 75 GWd/tU, which would imply the need to address considerable technical challenges, such as:

- Improvement of nuclear fuel analytical codes for the accurate prediction of fuel behaviour and performance at HBU rates;
- Update of source term calculations;
- Modifications for higher heat loads in the spent fuel pool;
- Performance of design and safety analyses at higher burnups addressing potentially new phenomena, including fuel fragmentation relocation and dispersal (FFRD);
- Degradation of safety performance in normal operation and in accident conditions, posing considerable challenges for the development and qualification of HBU nuclear fuels.

Several IAEA technical meetings have been held in the 2000’s to address the economic and safety aspects of HBU nuclear fuels. Since then, utilities have been mainly focusing efforts on safety improvements following the accident at the Fukushima-Daiichi NPP, as well as on the development and qualification of ATF.

The status of knowledge in the development, qualification and licensing of advanced nuclear fuels for WCR is illustrated in a TECDOC [1] on the same topic, which addresses nuclear fuels within an envelope to a rod average burnup of 62 GWd/tU (average fuel sub-assembly of 55 GWd/tU). It is a general observation that the nuclear fuel community worldwide intends to pursue a synergy that fulfils two objectives at the same time, enhanced economic performance and safety, by employing both ATF and the higher burnup operation beyond 62 GWd/tU in WCR.

Under these circumstances, in November 2022, a technical meeting was jointly organized by the Department of Nuclear Safety and Security and the Department of Nuclear Energy, with the objective of providing a platform for Member States to exchange information on the safety and performance of HBU nuclear fuels (e.g. beyond 62 GWd/tU) for WCR, considering their development, qualification and licensing.

Considering the relevance of the technical contributions provided during this technical meeting (which was attended by forty-three participants representing twenty-one Member States and two international organizations – the OECD Nuclear Energy Agency and the European Commission) the development of a TECDOC summarizing the material presented was identified as being a highly beneficial follow-up action to this technical meeting, reflecting the output from the IAEA efforts on the safety of HBU nuclear fuels for WCRs.

## 1.2. OBJECTIVE

The current TECDOC aims at illustrating the status in addressing safety and technological challenges encountered in the development, qualification and licensing of HBU nuclear fuels for WCR, summarizing the discussions held at this technical meeting.

## 1.3. SCOPE

The TECDOC encompasses design, qualification, licensing, and in-reactor operation of $UO_2$ Zr based alloys, including doped $UO_2$ and coated cladding, with discharge burnup exceeding 62 GWd/tU (fuel rod average) or 55 GWd/tU (fuel assembly average), or other equivalent criteria.

The TECDOC focuses on how safety and technological challenges associated with the development and qualification of HBU nuclear fuels for WCR are addressed by the different stakeholder involved. It covers the following topics:

- Nuclear utilities' experience in performing feasibility studies addressing burnup increase, including in revising loss of coolant accident (LOCA) and reactivity insertion accident (RIA) safety criteria, and in performing experimental campaigns of irradiation of lead test rods (LTR) and lead test assemblies (LTA) for improving the validation of numerical tools and methods used in safety analyses;
- Experimental testing (in-pile and out-of-pile) of fuels at HBU, aiming at progressing the phenomenological knowledge and related modelling and simulation capabilities, with a focus on microscopic restructuring, fission gas release (FGR) and fuel fragmentation and relocation, which determine the fuel behaviour in LOCA or RIA transients;
- Experience by fuel developers in development and qualification of fuel designs for HBU, with a focus on the adaptation and validation of tools and methods for safety analyses;
- Experience by regulatory bodies and technical support organizations in identifying technical issues that may be relevant to safety reviews, and the related assessments of the anticipated impact of HBU fuels on fuel dispersal during LOCA and RIA, and on the associated radiation doses and environmental impact.

## 1.4. STRUCTURE

The TECDOC consists of seven sections and one Annex.

Section 1 describes the background, objective, scope and structure of the publication.

Section 2 illustrates the drivers motivating the development of HBU nuclear fuels for WCR, as well as nuclear utilities' experience in performing feasibility studies addressing burnup increase, including in revising LOCA and RIA safety criteria, and in performing experimental campaigns of irradiation of LTR and/or LTA. It includes a table summarizing the current operational limits for the discharge burnup adopted in various Member States.

Section 3 describes the state of the art in the knowledge of physical phenomena occurring in nuclear fuel at HBU (including micro-structure and rim formation, FFRD, enhanced FGR), the associated multiphysics and multiscale modelling and simulation capabilities, as well as major challenges in simulating the fuel behaviour in LOCA or RIA transients.

Section 4 presents insights from experimental testing (in-pile and out of pile), supporting the development and validation of numerical models in thermomechanical codes, and identifies major gaps pertaining to the validation of numerical models used to simulate fuel behaviour in design basis accident (DBA) conditions (LOCA and RIA).

Section 5 describes the experience of fuel developers in development and qualification of fuel designs for HBU, including advanced fuel and cladding designs, irradiation of test fuel rods beyond 62 GWd/tU, pool-site inspections and post-irradiation examination (PIE), plans for obtaining licensing of fuel design codes and methods for HBU rates.

Section 6 illustrates the experience by regulatory bodies and technical support organizations in identifying technical issues that may be relevant to safety reviews, and related assessments of the anticipated impact of HBU fuels on fuel dispersal during LOCA and RIA, and on the associated radiation doses and environmental impact.

Section 7 presents the conclusions and highlights gaps which might be the subject of future activities.

The Annex contains extended abstracts for the presentations provided at the technical meeting.

## 2. NUCLEAR UTILITIES' PERSPECTIVE AND EXPECTATIONS FROM HBU NUCLEAR FUELS

Most current commercial light water reactors (LWR) around the world use low enriched uranium (LEU) fuel. In general, fuel costs comprise approximately 20% of NPPs' total generating costs. Fuel costs are driven, like for most commodities, by supply and demand with the costs for the uranium feedstock and conversion, enrichment, and fuel assembly fabrication processes. The discharge burnup and the uranium enrichment level are two amongst other constraints that directly impact the core design efficiency.

Practices relating to the design of equilibrium cycles show that the variation in cycle efficiency is primarily attributed to variations in enrichment and burnup, which are constrained by current regulatory limits on these parameters.

In the United States of America, utilities have leveraged recent activities associated with the development and deployment of ATF to revisit core design efficiencies that optimize batch reload quantities of fuel to reduce generation costs. The growing interest in greater fuel cycle efficiency will certainly lead over the next decade to an increasing demand for licensing of fuels enriched to more than 5 percent by weight (wt%) of U-235.

A study conducted by the Electric Power Research Institute (EPRI) [2] evaluates several issues related to these increasing limits ($UO_2$ enrichment and fuel burnup). Although revising these limits would result in economic benefits (increased flexibility in cycle length, reduced storage and disposal requirements for high level waste, and a positive benefit on the environmental impact of the entire fuel cycle), they may require regulatory changes and also long-term capital investments. Thus, the final decision to set new limits needs to consider the expected economic advantages, but also the commercial risks related to such an endeavour.

This section was prepared by Mr. Jinzhao Zhang (Tractebel-ENGIE, Belgium), while the session of the technical meeting was chaired by Mr. Al Csontos (Nuclear Energy Institute, United States of America).

### 2.1. SUMMARY OF PRESENTATIONS

#### 2.1.1. The drive to license and deploy ATF, LEU+ and HBU fuels in the United States of America

Mr. Al Csontos (Nuclear Energy Institute, United States of America) gave a presentation on the development and the implementation of new fuel technologies, allowing the improvement of their tolerance to accidents but also their reliability, while increasing their operational margins and enabling sustained economic performance, costs minimisation and efficiency improvement.

The presentation suggested safe and economical enabling of a 24-month cycle operation for the entire fleet of existing LWRs in the USA, with burnups up to ~75 GWd/tU, associated to enrichment extensions towards LEU from 5% to 10% (LEU+) beyond legacy limits in the mid-2020s.

The presentation also covered major milestones along this path, including on-going experimental programmes of irradiations of LTR and/or LTA in commercial reactors, as well as regulatory and research and development (R&D) activities underway [3].

### 2.1.2. Safety and performance of high burnup fuels: a utility's perspective

Mr. Jinzhao Zhang (Tractebel-ENGIE, Belgium) presented the feasibility study performed in 2007–2009 to assess the HBU safety and performance issues for Belgian nuclear power plants that are owned and operated by Electrabel [4-7]. The safety and performance feasibility in achieving limited burnup increase was confirmed for some Belgian NPPs, as follows:

- Up to a fuel assembly average burnup of 58 GWd/tU with existing fuel designs, with codes and methods for fuel design and safety analyses adapted for the purpose;
- Up to a fuel assembly average burnup of 62 GWd/tU or higher with advanced fuel designs, through improved codes and methods for fuel design and safety analyses.

The licensing effort was identified as being the most important economic constraint, particularly in the absence of NPP life extension. Therefore, it was concluded that burnup extension may not be profitable for all Belgian NPPs, and hence was not implemented.

The progress made since previous IAEA technical meetings on HBU fuels on topics such as fuel performance, safety and licensing aspects was presented. It was concluded that significant progress has been achieved since 2009, as follows:

- Better understanding of the fuel behaviour at HBU;
- Development and qualification of improved fuel designs for HBU;
- Revision of the LOCA and RIA safety criteria, based on tests of fuels at HBU;
- Improvement and validation of fuel codes for modelling and simulations of fuel behaviour at HBU;
- Development and licensing of advanced fuel design and safety analysis methodologies.

However, the lack of consistent HBU fuel licensing process was highlighted, which prevents utilities from requesting and implementing fuel burnup extension.

Hence, it was proposed to enhance international cooperation on safety and performance of HBU fuels and to ensure improved consistency of the various licensing frameworks to simplify or accelerate the licensing process.

### 2.1.3. Utility's experience with LTA irradiation in commercial reactors at Constellation

Mr. William Gassmann (Constellation, United States of America) highlighted the leading role of Constellation in supporting the development and/or deployment of ATF and/or HBU. Four ATF and/or HBU LTA campaigns are currently in progress at Constellation (at Byron-2, Limerick-2, Clinton and Calvert Cliffs-2 units), and an explicit HBU application is either in progress or under consideration for these LTA campaigns.

The presentation also introduced additional PIEs necessary for a complete characterization of phenomena involved at HBU and/or ATF (including gamma scan measurements for fission gas pressure and pellet integrity, detailed corrosion examination for cladding integrity, and destructive and non-destructive examination in hot cells).

## 2.2. SUMMARY OF DISCUSSIONS

Discussions in this session highlighted that current nuclear fuels have been developed, qualified, licensed, and operated in WCRs in operation up to the approved burnup limits in most Member States.

The maximum rate of discharge burnup approved by regulatory bodies in some Member States is indicated in the Table 1 below, based on the information presented at the technical meeting:

- In some Member States, significantly HBUs have been reached (e.g. up to a rod average burnup of between 70 and 80 GWd/tU in Sweden and Switzerland).
- The values are related to nuclear fuel (enriched uranium dioxide fuel and plutonium blended uranium dioxide fuel (MOX) with zirconium alloy cladding) currently loaded in pressurized water reactors (PWRs) as well as in boiling water reactors (BWRs) in operation.

There is an economic incentive to improve the efficiency of fuel use by increasing fuel enrichment and burnup. This fuel burnup increase is possible with the existing advanced fuels (e.g. PRIME [8], GAIA [9]) and in particular the recently developed ATFs (e.g. ENCORE [10], EATF [11]).

Lead test ATF rods or assemblies (LTRs and LTAs) are being irradiated in several NPPs in the USA and in Europe, and pool-side inspections or post-irradiation examinations are being performed to qualify these fuels.

However, implementation of HBU needs considerable efforts to resolve specific safety issues, such as FFRD. In addition, enhanced consistency in the licensing process adopted in Member States would be helpful to speed up the licensing of HBU worldwide.

TABLE 1. REGULATORY LIMITS FOR THE DISCHARGE BURNUP OF NUCLEAR FUEL CURRENTLY LOADED IN PWR AND BWR IN OPERATION

| Country | PWR max discharge burnup rate approved by regulatory body | | | BWR max discharge burnup rate approved by regulatory body | | |
|---|---|---|---|---|---|---|
| | Pellet average (GWd/tU) | Rod average (GWd/tU) | Average fuel sub-assembly (GWd/tU) | Pellet average (GWd/tU) | Rod average (GWd/tU) | Average fuel sub-assembly (GWd/tU) |
| Armenia | | 65 | | | | |
| Belgium | | | 55 | | | |
| Bulgaria | | 65 | 55 | | | |
| China | | 57 (62 for AP1000, 65 for WWER) | | | | |
| Czech Republic | | 64 | | | | |

TABLE 1. REGULATORY LIMITS FOR THE DISCHARGE BURNUP OF NUCLEAR FUEL CURRENTLY LOADED IN PWR AND BWR IN OPERATION (cont.)

| Country | PWR max discharge burnup rate approved by regulatory body | | | BWR max discharge burnup rate approved by regulatory body | | |
|---|---|---|---|---|---|---|
| | Pellet average (GWd/tU) | Rod average (GWd/tU) | Average fuel sub-assembly (GWd/tU) | Pellet average (GWd/tU) | Rod average (GWd/tU) | Average fuel sub-assembly (GWd/tU) |
| France | | | 55 for $UO_2$<br>52 for MOX | | | |
| India | | | 49 | | | |
| Japan | | | 55 | | | 55 |
| Republic of Korea | | 62 | | | | |
| Russian Federation | | 65 | | | | |
| Spain | | | 55 | 71.5 | | |
| Sweden | | 72 | | | | |
| Switzerland [a] | 76 (KKB)<br>82 (KKG) | 65 (KKB)<br>75 (KKG) | 60 (KKB)<br>70 (KKG) | 75 (KKL [b]) | n/a [b] | n/a [b] |
| United States of America | | 62 | | | | |

a In Switzerland, plant specific limits are applied for fuel burnup at Beznau NPP (KKB) and Gösgen NPP (KKG).

b For the KKL NPP, the criterion on the pellet average burnup is the most restrictive and envelopes the two others.

# 3. PHENOMENOLOGICAL KNOWLEDGE AND INSIGHTS FROM MODELLING AND SIMULATION

Increasing the discharge burnup leads to significant modifications of the fuel rods during irradiation. These changes may have a major impact on the fuel behaviour in normal and accident conditions (in particular RIA or LOCA transients).

The objective of this session at the technical meeting was to share the latest developments in multiphysics and multiscale modelling and simulation of nuclear fuels with a focus on HBU phenomena characterization (including micro-structure, rim, FFRD, enhanced FGR, source term, etc.).

This section was prepared by Mr. Vincent Georgenthum (Institut de Radioprotection et de Sûreté Nucléaire, France) and Mr. Nathan Capps (Oak Ridge National Laboratory , United States of America), and Mr. Vincent Georgenthum also chaired the session of the technical meeting.

## 3.1. SUMMARY OF PRESENTATIONS

A total of seven presentations were made in this session, on topics such as HBU fuel restructuring, FGR, FFRD, fuel rod model and code development and validation.

### 3.1.1. Current status and planned research activities for the TRANSURANUS code application to high burnup fuels

Mr. Paul Van Uffelen (European Commission, Karlsruhe) summarised the efforts made over the past decade towards the application of the TRANSURANUS [12] fuel rod performance platform to HBU fuels.

Specific developments were made to enhance the mechanistic modelling of the high burnup structure (HBS) formation in the SCIANTIX [13] and MFPR-F [14] codes that are coupled with TRANSURANUS. A benchmark of these codes is planned in the frame of the R2CA European project. Further developments aim at the simulation of DBA such as LOCA and RIA. The important role of international benchmarks organised by the IAEA and OECD NEA were thereby underlined, along with the role of creating an international reference database for code development and validation.

In parallel with the further refinement of the HBS modelling, ongoing code developments for modelling the hydrogen uptake and its consequences on the cladding behaviour during operation as well as subsequent storage were also outlined.

Finally, the current application of the best estimate plus uncertainty (BEPU) methods and the potential of machine learning were illustrated.

### 3.1.2. Consistent modelling of fuel fragmentation, grain boundary fracture and fission gas release in LWR fuel rods during loss of coolant accident and reactivity insertion accident

Mr. Grigori Khvostov (Paul Scherrer Institute, Switzerland) presented several amendments made in the FALCON code [15] coupled with the GRSW-A model [16], as well as in the FRELAX code [17], aiming at improvement of calculation for specific types of thermal transients, such as RIA and LOCA.

A model was developed and integrated into GRSW-A for the trapping of released fission gases by specific closed voids (‘pockets’), during base irradiation.

The capabilities to perform transient analysis with the GRSW-A model of the FALCON code were extended to simulate the processes of fuel fragmentation and concomitant FGR due to grain boundary fracture.

A fuel fragmentation dependent analytical criterion for the onset of grain boundary fracture and transient FGR has been proposed.

The FRELAX code was extended to allow the simulation of bulk flow and diffusion of the gases in the rod free volume during thermal transients.

After a calibration of the model parameters — performed using data of a selected RIA test in the Nuclear Safety Research Reactor (NSRR) in Japan, and a Halden LOCA test — the updated codes and models were applied to extended sets of RIA (NSRR and CABRI-SL) and LOCA (Halden) tests, assuming the same set of best estimate parameters.

#### 3.1.3. IRSN R&D activities on high burnup fuel behaviour in RIA and LOCA accident conditions

Mr. Vincent Georgenthum (IRSN, France) presented current IRSN R&D activities related to the study and modelling of the fuel behaviour in RIA and LOCA accident conditions.

Concerning the behaviour of HBU fuel during RIA transient, the following issues were identified as requiring further investigations after the experimental programmes performed in experimental reactors NSSR (Japan) and CABRI (France):

- Transient fission gas behaviour and its impact on clad loading during the entire transient;
- The rod behaviour with high clad temperature and internal pressure;
- Post-failure phenomena (fuel ejection, fuel–coolant interaction with finely fragmented solid fuel).

The CABRI International Project (CIP) investigates the behaviour of fuel rods in PWR conditions during RIA transients in particular for HBU fuels and advanced fuels and claddings. Numerical models, concerning the fission gas behaviour and fuel coolant interaction, are currently under development to be introduced in the SCANAIR [18] and CIGALON [19] codes.

With regards to the study of fuel behaviour under LOCA, the pending issues are fuel fragmentation, relocation and dispersal after cladding burst. In this frame, specific modelling is under development in the DRACCAR [20-21] code to simulate the 3D thermo-mechanical behaviour and reflooding of a fuel rod assembly during a LOCA transient taking into account fuel fragmentation, relocation and flow blockage. These models are based on academic studies and results of separate effect tests (such as the COAL experiments, part of the PERFROI project [22-23] run by IRSN in the period 2014–2020).

### 3.1.4. Fuel Fragmentation Dispersal and Consequence Evaluation of Higher Burnup Fuel

Mr. Ken Yueh (EPRI, United States of America) presented a study addressing the fuel fragmentation and dispersal for HBU fuel as well as the evaluation of consequence (in terms of possible blockage on the grid, lowering the rod coolability).

To this aim, it is necessary to determine the number of burst rods, the extent of fuel release from each rod, and — under high steam flow conditions — how the fuel disperses in the fuel assembly, the reactor coolant system (RCS) and the containment.

An EPRI in house thermohydraulic code, GOTHIC [24], was used to evaluate fuel fragment dispersal within the RCS and containment.

Model calculations show that fuel fragments carried by steam could impact a grid at significant velocities and recent drop tests of post-LOCA fuel fragments showed that sub-fragmentation is likely.

To evaluate possible blockage at the grid, a flow experiment with a 3x3 fuel bundle with two spacer grids will be conducted. The experiment will use crushed non-irradiated $UO_2$ fuel fragments.

### 3.1.5. Assessing the impact of prototypic high burnup operating conditions on fuel fragmentation, relocation and dispersal

Mr. Nathan Capps (ORNL, United States of America) presented an historical overview of the research, development and demonstration process that has accompanied the increase of fuel-to-rod average burnups from approximately 30 GWd/tU (for the very first NPP licensed in the USA) up to the current licensed limit of 62 GWd/tU peak rod average for $UO_2$ and Zr alloy-based fuels.

A BEPU pin-by-pin high burnup LOCA analysis technique developed at ORNL has been used to assess full core HBU FFRD and to identify approaches for minimizing or potentially mitigating FFRD through core design optimizations.

Results obtained on a four-loop Westinghouse commercial PWR were illustrated. A statistical analysis of the finite element method-based fuel performance code BISON [25] results was performed to predict the full-core susceptibility to FFRD during a large break LOCA (LBLOCA) event.

### 3.1.6. The SCIANTIX grain-scale code: recent developments for high burnup fuels

Mr. Davide Pizzocri (Politecnico di Milano, Italy) presented the last development of SCIANTIX code for HBU fuels. SCIANTIX operates at the scale of fuel grains.

Several developments were made to improve the semi-empirical description of HBS formation and porosity evolution at HBU:

- A fuel material representing the HBS, incorporating the properties of the restructured fuel (e.g. grain size, lattice parameter), was implemented in the code;

- The formation of the HBS based on the combination of an empirical threshold and the Kolmogorov-Johnson-Mehl-Avrami (KJMA) model;
- The HBS porosity prediction (on a local burnup basis);
- The porosity distribution evolution from a physics-based model (in terms of the pore number density, the average number of gas atoms per pore and of their variance);
- The size of the HBS pores as a function of the porosity and the number density of pores.

The SCIANTIX grain-scale code includes all features required for a semi-empirical description of the fuel structure formation and porosity evolution at high burnups.

These features allow direct couplings into fuel performance codes, in terms of numerical robustness, calculation time and verification and validation.

### 3.1.7. Review of high burnup fuel behaviour during normal operation and accidental transients

Mr. Ioan Arimescu (Nufology Plus LLC, United States of America) presented a critical review of HBU fuel behaviour.

The efforts undertaken by the nuclear industry to further extend the operational burnup range, beyond the currently approved limit, entails research and testing of fuel behaviour at extended HBU to confirm adequate fuel performance. This is due to the necessity of performing design analyses to demonstrate adequate fuel behaviour during normal operation as well as during anticipated abnormal occurrences and accidental situations.

The following new or enhanced-effects phenomena have been identified in HBU fuels, some of which may need further studies through international cooperation:

- Decreasing fuel thermal conductivity, also known as burnup degradation of fuel thermal conductivity;
- Enhanced FGR during normal operation and thermal transients;
- HBS formation;
- Fuel–Cladding bonding;
- Cladding corrosion and hydrogen uptake;
- Zirconium alloys irradiation stress-free growth;
- FFRD.

Fuel behaviour research is a complex process involving both out of pile as well as in-pile experiments and tests. When a new phenomenon or process is identified, separate-effects tests are devised to focus on the specific process in question, to understand the phenomenology and provide data for its quantification by models. However, integral tests are also necessary to verify that the interrelationships of the given process with all other phenomena occurring in the fuel are adequately captured.

The result of HBU fuel experimental and theoretical studies is the demonstrated capability to predict fuel performance for fuel design confirmation and licensing analyses to assure safe and reliable fuel operation to the desired maximum burnup.

## 3.2. SUMMARY OF DISCUSSIONS

The modelling of HBU fuel behaviour during normal and accident conditions needs improvement due to the significant evolution of the fuel rods during irradiation.

The first need is to refine the description of the fuel modifications for rods with average burnup for rods higher than 62 GWd/tU. In particular, the properties (e.g. grain size, pores size and shape and lattice parameter) of the restructured fuel, the gas distribution in intra- and intergranular bubbles and the associated radial profiles needs to be better characterized.

Modern mechanistic models and codes are under development to better evaluate the complex HBS state of the fuel (SCIANTIX, MFPR-F, GRSW-A/FALCON, etc.). These mechanistic codes have to be coupled with fuel performance code (such as TRANSURANUS, BISON).

Some questions on the fuel restructuring mechanism remain. Challenges remain concerning the HBU fuel behaviour modelling during transient conditions (RIA and LOCA) and work are under progress regarding:

- Transient FGR mechanism and criteria for burst release. Depending on the thermomechanical conditions and the transient kinetics, gas release can occur after grain boundary (GB) saturation with thermal diffusion or GB opening due to pore over pressurization.
- Radial and axial gas flow in the fuel and the gap. The fuel permeability and the fuel–cladding bonding are key parameters.
- Clad mechanical behaviour at high temperature: clad constitutive laws and failure criteria have to take into account for transient conditions (temperature, kinetics) cladding type and fuel burnup.
- Fuel cracking and fragmentation, associated fuel relocation and impact on clad temperature and coolability of ballooned cladding.
- Fuel fragmentation dispersal and the consequence after clad failure during a LOCA.
- Fuel coolant interaction after clad failure (for RIA), in particular the size and the kinetics of ejected fuel and the evaluation of the mechanical energy.

Fuel performance codes integrate progressively both new HBU models and BEPU methodology to assess single rod or full-core behaviour in accidental conditions.

# 4. EXPERIMENTAL PROGRAMMES SUPPORTING THE DEVELOPMENT OF HBU NUCLEAR FUELS

To support the understanding of phenomena particularly important to HBU, the collection (and availability) of experimental data on irradiated fuel are particularly important. Activities in this regard include both irradiation testing in material test reactors (MTRs), as well as hot cell studies on irradiated fuel. In addition, studies on un-irradiated material can also be of value to supplement hot cell studies, as well as also contributing to extending current testing capabilities.

The objectives of this session at the technical meeting were to review recent and ongoing experimental activities and programmes of particular relevance to HBU. Also important for the session was to provide information of available data that could be of valuable use for activities in fuel modelling development as well as evaluating current fuel safety criteria and safety margins.

This section was prepared by Mr. Terje Tverberg (Institute for Energy Technology, Norway). He also chaired the session of the technical meeting.

## 4.1. SUMMARY OF PRESENTATIONS

A total of seven presentations were provided in this session, covering the last techniques available at the Commissariat à l'Energie Atomique et aux Energies Alternatives (CEA), the Halden HBU fuel research, the Japan Atomic Energy Agency (JAEA) HBU fuel RIA testing, HBU fuel LOCA testing, post-irradiation examinations, etc.

### 4.1.1. Restructuring and fission gas bubbles in high burnup UO2 fuels

Mr. Jean Noirot (CEA, France) presented the latest techniques available in the CEA LECA-STAR facility to characterize the restructuring occurring in the fuel pellets and on the associated micrometric fission gas bubbles in HBU fuel.

Recent progress was made through the increasing use of improved micro-examination techniques, in particular electron backscattered diffraction (EBSD) and focused ion beam, scanning electron microscope (FIB-SEM) 3D examinations. These techniques provide local crystal orientation maps that can be used to quantify the restructuring progress and 3D geometrical information on the bubbles.

Results obtained on a set of PWR $UO_2$ samples at various burnup rates, including Cr doped $UO_2$ fuels with large grains, showed:

- On the pellet's periphery: the HBS restructuring starts first by the formation of sub-grains with orientations close to that of the original grains followed by the formation of randomly oriented sub-grains. 3D FIB-SEM examination in the HBS volumes shows the complexity of the HBS bubble surfaces, with the emerging small grain boundaries on these surfaces.
- In the central volumes: restructuring with sub-grains orientation spreading around that of their original grain was detected by EBSD for all the fuels with an average burnup higher than 60 GWd/tU. With the restructuring progress, the local porosity and the fraction of complex bubbles tend to increase without forming long distance networks.

The local temperatures and local burnup are major parameters in the restructuring process, but they are probably not the only ones.

The central restructuring, characterized in this work, certainly plays a role on the general increase in the FGR at high burnup. This increase without clear bubble interconnected networks rises the need for low scale studies on the grain boundaries and low angle grain boundaries relatively to fission gases under irradiation.

Fission gas density in bubbles, measured by combining electron probe micro analyser (EPMA), secondary ion mass spectrometer and FIB-SEM characterizations were provided.

Ongoing works on FIB-SEM cantilever micro-mechanical testing, providing micro-scale fracture properties of irradiated fuels, in particular in restructured zones, and on transmission electron microscope (TEM) works on thin foils cut in irradiated fuels using FIB/SEM were mentioned.

#### 4.1.2. Experience with irradiation testing in relation to high burnup investigations at the OECD Halden Reactor Project

Mr. Terje Tverberg (Institute for Energy Technology, Norway) presented experience with HBU fuels testing from various OECD Halden Reactor Project (HRP) research programmes.

The presentation offered an overview of different irradiation types and experimental setups utilized in the Halden reactor relevant for studies of HBU fuel, and also provided examples of data and results obtained from the various testing setups (including IFA-519.9 experiment data comparison to modelling predictions [26]). While in the later years the HBU fuels tests would preferably be done using re-instrumented commercially irradiated LWR fuel, over several years irradiations were also done starting from fresh fuel and collecting relevant data on fuel temperature, rod pressure, as well as pellet–cladding interaction. Of particular note, such HBU tests starting from fresh fuel were often done on fuel with an enrichment of 10% or higher, which could have some relevance to the current industry efforts to move to LEU+ enrichment.

Activities towards the latter stages of operation of the Halden Reactor were largely focused on in-pile LOCA testing of HBU $UO_2$ fuel in the IFA-650 test-series. The 13 tests conducted in this programme on re-instrumented LWR fuel ranging from medium to very HBU have provided valuable data on LOCA behaviour, in particular as regards FFRD. The IFA-650 LOCA test programme was also aiming at supplementing ongoing hot cell LOCA activities at other facilities, in particular the Studsvik Cladding Integrity Project (SCIP).

The presentation also provided a listing of available data sets to support modelling of HBU fuel in the Integral Fuel Performance Experimental Database (IFPE) [27-28] database at NEA as well as data used in the various modelling exercises organized by the IAEA in the coordinated research projects (CRPs) FUMEX [29-30] and FUMAC [31].

#### 4.1.3. JAEA studies on high burnup LWR fuel behaviour under Reactivity Initiated Accident conditions

Mr. Yoshinori Taniguchi (Japan Atomic Energy Agency, Japan) presented data collected from research programmes on RIA at HBU.

Several fuel safety research activities are ongoing at JAEA related to HBU aiming to provide an enhanced database supporting the development of future regulatory documents related to

HBU fuels, and to evaluate the adequacy of current safety criteria and limits. Studies related to LOCA, code development and RIA are actively pursued. The presentation gave a summary of extensive data collected at JAEA on RIA studies in the NSRR reactor. Over the years, a database of more than 1000 unirradiated test segments and around 100 irradiated segments was developed. Studies have been conducted on both $UO_2$ fuel and mixed oxide fuel (MOX) fuel.

Key findings from extensive HBU fuel test campaigns in the Advanced LWR Fuel Performance and Safety (ALPS) and ALPS-II programmes, were that extension of burnup significantly affects the pellet cladding mechanical interaction (PCMI) failure limit, heat transfer and transient FGR. Of particular importance to the PCMI failure limit in RIA conditions are hydrogen content and hydride distribution in the irradiated cladding. Planned new studies will in particular aim to fill in the existing knowledge gap on burnup effect on transient FGR in RIA conditions.

#### 4.1.4. Integral behaviour of high burnup fuel rod under LOCA: effect on steam oxidation, integral FFRD experiment, and regulatory implications

Mr. Youho Lee (Seoul National University, South Korea) presented results from studies on LOCA and FFRD at the Department of Nuclear Engineering at Seoul National University (SNU).

The design and capabilities of integral LOCA experiment equipment at SNU (i-LOCA facility) were presented. The test facility uses induction heating for pressurized fuel rods containing surrogate pellets (i.e. whole pellets or crushed powder form) that undergo the entire process of ballooning, burst, steam oxidation, and reflood quenching.

Both single rod and multiple (4) rod arrangements can be applied, with heating rates up to 10°C/s. Ballooning burst and oxidation behaviour are documented with a high speed digital camera as well as infrared camera recordings during the test.

Tests results from comparative tests of Zr alloy and Cr coated Zr alloy using surrogate $ZrO_2$ powder / powder mixture to simulate fragmented fuel pellets, showed similar burst temperature, balloon and burst opening for the two claddings.

In a test comparing the effect of single powder size (simulating >80 GWd/tU and mixed powder size), the mixed powder size test showed less ballooning and smaller burst opening as well as significantly less dispersal (100% dispersal vs. 31% dispersal).

The capabilities of, and results from, the TRANOX Zircaloy steam oxidation model for mechanistic modelling of Zircaloy oxidation in LOCA transients were also presented.

#### 4.1.5. HBU LOCA testing plan at the Idaho National Laboratory

Mr. Robert Armstrong (Idaho National Laboratory, United States of America) presented the status and plans for LOCA testing studies in the Transient Reactor Test (TREAT) and Severe Accident Test Station (SATS) facilities at INL (TREAT) and ORNL (SATS).

Research programmes at INL and ORNL aimed at supporting nuclear industry investigations to extend burnup and enrichment limits, tacking key issues for HBU fuel, in particular FFRD during LOCA. Developments include in-situ instrumentation for monitoring fuel motion, transient FGR, balloon extent, and improved surface temperature measurements. The SATS facility at ORNL is operational, with upgrades being implemented. Commissioning tests for

TREAT LOCA facilities are planned for 2023–2024, with the aim of starting tests with HBU fuel in 2024.

The presentation showed the proposed test matrix under discussion, which aims to align with Halden and/or Studsvik tests and build on and add to past investigations. Coordination with ongoing activities at Studsvik are ensured through the SCIP. Specific studies aimed at ascertaining the impact of decay energy heatup (DEH) vs. stored energy heating (SEH) on FFRD in LB-LOCA conditions.

#### 4.1.6. NEA activities supporting improved understanding of high burnup fuel behaviour

Ms. Michelle Bales (OECD NEA) presented current NEA activities related to understanding of HBU fuel behaviour.

Several NEA working groups and expert groups have ongoing activities relevant to HBU, including the Working Group on Fuel Safety (WGFS), the Expert Group on Reactor Fuel Performance (EGRFP), as well as in joint projects like the HRP [32], the Framework for Irradiation Experiments (FIDES-II) [33] and the SCIP [34].

Completed WGFS studies include a state of the art report on nuclear fuel behaviour under RIA [35], a Status Report on fuel safety implications of extended enrichment and high reactivity/high suppression core designs. Ongoing WGFS activities include a Status Report on good practices for analyses of design extension conditions as well as a Report on development of validation matrix and database for RIA related experiments.

EGFRP activities include benchmark exercises on PCMI and on burst FGR utilizing two IFPE cases (REGATE and HATAC-C2).

Regarding new and ongoing experimental programmes in the OECD NEA framework, the FIDES-II programme will include a joint experimental project on HBU experiments in RIA (HERAD) using the TREAT facility in the USA and the NSRR reactor in Japan. Also, the current SCIP-IV programme hosted at Studsvik in Sweden includes studies related to HBU, notably continued LOCA testing as well as on dry storage.

#### 4.1.7. Recent advanced post-irradiation examinations on HBU fuel at Idaho National Laboratory

Ms. Fabiola Cappia (INL, United States of America) presented recent developments in post-irradiation examinations capabilities at INL.

The presentation focused on recent developments at INL's hot cell facilities to extend the range of advanced techniques available for characterization of HBU fuel. The new developments in hot cell techniques at INL are largely aimed at supporting LOCA test programmes introduced in the presentation illustrated in subsection 4.1.5 with the integral tests at INL's TREAT facility and the separate effects tests in the SATS facility at ORNL.

In this respect, it is recognized that detailed information and characterization of irradiated fuel samples prior to such tests planned in the programs mentioned above will be of high importance. To that end, the existing techniques aim at producing detailed information on porosity distribution, distribution of caesium and fission gas bubbles, as well as detailed characterization of grain structure and grain subdivision using a combination of SEM and transmission electron microscope (TEM), EBSD and EPMA. Recent additions to existing capabilities also include techniques for detailed (local) characterization of elastic properties and fracture strength as well as local thermal conductivity measurements. Examples of results from characterization of a sample taken from a rod with burnup 75 GWd/tU that had been tested at in SATS were also presented.

### 4.2. SUMMARY OF DISCUSSIONS

Presentations in this technical session highlighted that many aspects regarding HBU fuel behaviour in transient conditions have been studied in large detail, and burnup effects (e.g. on cladding failure limit and fuel dispersal in RIA and LOCA conditions) have been identified and to a certain extent also quantified. Advanced post-irradiation examinations techniques are also available to provide insights into the details of formation of HBS, and also on the distribution of fission gases in the HBU fuel.

Nevertheless, the discussion in this session largely focused on the topic of transient FGR, and the role of gas release resulting from fuel (micro) cracking vs. diffusion in the time frame of a transient like LOCA. In this context, it was again underlined that good pre- and post-test characterization data are important to help understand the phenomena. The question was also raised which kind of data (specifically with respect to EPMA both pre- and post a transient test) — if any — is available in the open literature.

Reflecting discussions held in other sessions of the meeting, it was also mentioned here that open questions remain to be further addressed regarding FFRD. There is still a need for more research and data to fill in existing knowledge gaps hampering complete understanding of this phenomenon (e.g. detailed mapping of the pre-transient microstructure and improved understanding of how the distribution of fission gas bubbles and transient FGR might affect the extent of fuel fragmentation during a LOCA).

The importance of assessing and documenting associated uncertainties when reporting experimental results was also underlined during the discussion.

# 5. EXPERIENCE BY FUEL DEVELOPERS IN DEVELOPMENT AND QUALIFICATION OF FUEL DESIGNS FOR HIGH BURNUP

Reliable and high performance fuels are needed to support the licensing and target operation at HBU in NPPs.

This section of the technical meeting was devoted to review the current status of HBU fuel development, qualification, licensing and operation from technology developer's perspectives. The objective of this technical sessions was to share experience of fuel developers in:

- Development and qualification of fuel designs;
- Irradiation of 'lead test' or 'lead use' fuel rods and post-irradiation examinations;
- Validation and application of computer codes and methods for fuel design and safety analysis;
- Plans for licensing of fuel designs and codes, models and methods for implementation of HBU fuels beyond 62 GWd/tU (fuel rod average burnup).

This section summarizes the main outcomes of the presentations, the conclusions and discussions.

This section was prepared by Mr. Jinzhao Zhang (Tractebel-ENGIE, Belgium) and Ms. Nuria Doncel (ENUSA, Spain). Mr. Jinzhao Zhang also chaired the session of the technical meeting.

## 5.1. SUMMARY OF PRESENTATIONS

A total of five presentations were made in this session, covering fuel management strategy, fuel product development and qualification, testing, licensing and implementation plan.

### 5.1.1. Fuel management strategy and plan for increasing fuel burnup in China

Mr. Yongjun Jiao (Nuclear Power Institute of China, China) presented the current fuel management status together with the strategy and plan for increasing the fuel burnup in China.

The refuelling cycle length of NPPs in China is around 18 months and the fuel assembly (FA) burnup limit is around 52 GWd/tU (except for AP1000 and the Water Energetic Reactor (WWER), which feature a higher fuel rod and FA burnup limit).

To further improve fuel economy, some NPPs are considering adopting a 24-month refuelling strategy with increased uranium enrichment (5-6%) and an increased fuel assembly burnup limit (57 GWd/tU).

The plan for increasing FA burnup is associated with various phases, associated to innovative FA technologies:

- An updated version of the CF3 design [36] will allow increasing the burnup from the currently approved limit of 52 GWd/tU to 55 GWd/tU, and in-pile irradiation tests of lead FA will be carried out by 2025.
- AFA3GAA [37] will allow increasing FA burnup to 57 GWd/tU, based on international experience and hot cell inspection.
- ATF with Cr coated cladding will allow increasing the FA burnup to 62 GWd/tU, by 2030.

An irradiation test plan, associated with the deployment of innovative FA technologies, is being designed to provide the necessary evidence supporting applications to progressively increase the fuel burnup limit, hence meeting the expectations of the Chinese national nuclear safety authority (NNSA).

### 5.1.2. ENUSA experience on high burnup demonstration programmes

Ms. Nuria Doncel (ENUSA, Spain) presented the ENUSA experience in design, irradiation and inspection of HBU fuel demonstration programmes.

During the last two decades, ENUSA promoted several HBU fuel irradiation programmes, some with standard fuel features and others aiming at characterizing the behaviour of fuel rods in the framework of demonstration programmes in Spanish PWRs, associated to extensive post-irradiation examinations (including hot cell examinations), as follows:

- The Vandellós segmented fuel rods programme (1991-2002) and burnup extension programme (1991-2003), with rod average burnup up to 67 to 75 GWd/tU;
- The Vandellós HBU programme (1998-2009), with rod average burnup up to 74 GWd/tU;
- The Almaraz new cladding alloy programme (2006-2014), with rod average burnup up to 68 GWd/tU.

The data obtained from those HBU programmes (such as FGR or mechanical responses) was used to improve models and methods. However, at present there is no strategy for increasing the current licensed burnup limit (62 GWd/tU rod average burnup) in Spanish NPPs due to the planned phaseout, between 2027 and 2035.

ENUSA is working with Westinghouse and utilities on the irradiation of LTR and/or LTA with ATF features (e.g. Cr coated Optimized ZIRLO, etc.).

### 5.1.3. High burnup fuel development and implementation strategy at Westinghouse

Mr. Kevin Barber (Westinghouse, United States of America) presented Westinghouse’s HBU fuel development reported that:

- Westinghouse has developed and qualified the PRIME fuel with ADOPT® [38] pellets and AXIOM® [39] cladding for HBU applications.
- Westinghouse is developing and qualifying the ENCORE® fuel with ADOPT pellets and Cr coated Optimized ZIRLO [40] cladding (near term), as well as uranium nitride (UN) pellets and silicon carbide cladding (long term) for HBU applications.

Westinghouse has adopted a two-step process for the licensing of burnup extension programmes:

- An incremental fuel burnup limit increase to 68 GWd/tU (rod average burnup) by putting rods in peripheral assemblies in the reactor core (topical report submitted in December 2021);
- An increment on the fuel burnup limits up to 75 GWd/tU (rod average burnup) with an enrichment increase above 5%.

Westinghouse is supporting the EPRI Alternate Licensing Strategy (ALS) for FFRD by accounting for the extremely low likelihood of occurrence for a postulated LBLOCA, and justifying non-burst for small break LOCA, and in parallel developing licensing basis analysis capabilities for FFRD.

Data gaps to support FFRD analysis capabilities are being addressed via industry collaboration and Westinghouse testing.

### 5.1.4. Licensing Framatome's fuel designs for higher burnup and low enriched uranium beyond 5% (LEU+)

Mr. Brian Friend (Framatome, United States of America) presented Framatome's fuel and licensing plan for HBU and high enrichment (LEU+).

Framatome is developing Advanced Fuel Management (AFM), using innovative fuel material, associated with advanced codes and methods, capable of supporting burnups up to 75 GWd/tU (rod average burnup) utilizing LEU+ enrichments for the following reasons:

- Increasing enrichment and burnup enables the transition of PWR reactors from 18-month to 24-month refuelling with advanced fuel designs (e.g. GAIA);
- The use of LEU+ enables benefits through a higher power density;
- Advanced codes and methods along with LEU+ and BU higher than 62 GWd/tU allow for additional economical savings.

Framatome is adopting a two-step approach for HBU applications:

- Increased Enrichment Topical Report for PWRs (topical report submitted to the U.S. Nuclear Regulatory Commission, U.S. NRC, in 2021).
- Increased burnup licensing approval for $UO_2$, M5 Framatome, PROtect® Cr doped pellets and Cr coated cladding by 2027.

Framatome intends to use the following licensing strategy for increased burnup:

- Applicability of advanced codes and methods for rod average burnup higher than 62 GWd/tU;
- Validation of the ARCADIA code system (including COBRA-FLX) and mechanical codes;
- Validation of the applicability of safety analysis codes and methods at HBU (including the impact of FFRD).

Framatome will also use a similar licensing process to apply LEU+ enrichments and increased burnup to boiling water reactors (BWR) reactor methods.

### 5.1.5. Post irradiation characterization of high burnup oxide fuels

Ms. Prerna Mishra (Bhabha Atomic Research Centre, India) presented post-irradiation examinations carried out on nuclear fuel for a pressurized heavy water reactor (PHWR), irradiated up to burnup rates of 22 GWd/tU (with respect to core average discharge burnup and maximum design burnup of 7 GWd/tU and 15 GWd/tU, respectively) in India.

Post-irradiation examinations, which have been performed to characterize the extent of restructuring in the fuel and the residual ductility of the cladding at extended burnups, have shown satisfactory performance of the fuel from 15 GWd/tU to 22 GWd/tU, hence providing confidence in extending the average burnup of PHWR fuel with suitable fuel design modifications.

## 5.2. SUMMARY OF DISCUSSIONS

The presentations in this session highlighted efforts by fuel developers in the development and qualification of fuel products for operation at HBU rates:

- Fuel vendors have developed and qualified fuel products that have demonstrated a safe and reliable operation up to the licensed burnup limits (e.g. PRIME and GAIA);
- The existing operating experiences with most of the fuel products as LTR or LTA, or lead use fuel assemblies slightly beyond those licensed burnup limits, is also satisfactory (e.g. GAIA for fuel rod burnup up to 70 GWd/tU);
- ATFs open the possibility of a potential increase of burnup limits (rod average burnups up to 75 GWd/tU);
- Fuel vendors are adapting safety analysis codes and methods for their use in the HBU range, the validation of which will be made against already existing or new test data (which will require new experimental programmes), including the impact of FFRD;
- Gaps exist in the validation and licensing of codes and methods for extended burnup increase (higher than 68-70 GWd/tU), in particular for FFRD-related phenomena, which are being addressed via industry collaboration and internal testing;
- EPRI has proposed an ALS for addressing FFRD by accounting for the extremely low likelihood of occurrence for a postulated LBLOCA, and justifying non-burst for small break LOCAs, and in parallel developing licensing basis analysis capabilities for FFRD.

# 6. LICENSING EXPERIENCE AND REGULATORY PERSPECTIVE

Regulatory bodies worldwide are preparing to perform safety reviews to support licensing activities associated with HBU fuel. In certain countries, some licensing activities associated with HBU fuel are already underway or have been completed.

The objective of this session was to share relevant knowledge and experiences associated with licensing (or preparing to license) HBU fuel to encourage regulatory cooperation and consistency. The presentations in this session described both (1) technical issues that may be relevant to safety reviews and (2) regulatory considerations and process issues that may need to be addressed in safety reviews associated with HBU fuel.

This section was prepared by Mr. John Lehning (U.S. NRC, United States of America), while the session of the technical meeting was chaired by Mr. Salvador Goranov (Bulgarian Nuclear Regulatory Agency, Bulgaria).

## 6.1. SUMMARY OF PRESENTATIONS

A total of six presentations were made in this session, covering safety implications, safety design criteria, FFRD, licensing framework and practices for HBU fuel.

### 6.1.1. Safety implications of using high burnup fuel assemblies

Mr. Surik Bznuni (Nuclear and Radiation Safety Center, Armenia) provided an overview of anticipated impacts of HBU nuclear fuels on the fuel cycle back end, first introducing implications in terms of additional needs of validation of numerical tools (used for modelling of reactor physics, criticality safety, source term, decay heat and shielding effects) and related input data libraries in the range of uranium enrichment beyond 5% and discharge burnup beyond 62 GWd/tU.

The transport and storage of HBU spent nuclear fuel are likely to be impacted by increased values of thermal load, decay heat as well as neutron and gamma dose rates, which may need either longer cooling times prior to transport or a design modifications of transport casks and configurations for spent nuclear fuel storage, and which will inevitably involve adaptations of licensing processes.

### 6.1.2. Adequacy of safety design criteria at high burnup fuel type of unit 5&6 Kozloduy NPP

Mr. Salvador Goranov (Bulgarian Nuclear Regulatory Agency, Bulgaria) presented the results of an analysis demonstrating how relevant safety design criteria were met for a fuel transition involving HBU fuel (up to 65 GWd/tU rod average) for the Kozloduy NPP units 5 & 6 (WWER-1000/320).

The presentation also covered the necessary updates to the relevant chapters of the safety analysis report (including chapters 1 and 15), carried out in conjunction with the periodic safety review of these two units.

### 6.1.3. Approaches to safety regulation of high burnup nuclear fuel introduction in the Russian Federation

Mr. Sergey Makovskiy (Scientific and Engineering Centre for Nuclear and Radiation Safety, Russian Federation) presented an overview of the nuclear programme of the Russian Federation and discussed how relevant regulations ensure safe operation with HBU fuel.

The presentation focused upon fuel designed for WWER reactors, and discussed how, under the current enrichment limit of 4.95%, the fuel cycle has gradually been lengthened to 18 months.

The presentation identified several key considerations associated with increased fuel burnup, including the following:

- Changes in strength of fuel assembly components;
- Structural changes of the nuclear fuel;
- Dimensional changes in the fuel assembly geometry;
- Increased fuel swelling and consequent pressure increases on fuel cladding;
- Increased decay heat of spent fuel;
- Increased radioactivity and dose rates from spent fuel.

The presentation provided an overview of rules and regulations applicable to nuclear fuel and discussed requirements for adequate strength and acceptable thermomechanical behaviour of fuel assemblies, requirements for bench tests and in-reactor tests of new nuclear fuel to confirm satisfaction of safety criteria, and requirements for maximum fuel burnup.

The presentation further discussed the validation and certification of codes for performing safety analyses for nuclear fuel, noting that a key criterion for code certification is that sufficient experimental data need to be considered in the code validation process. The presentation noted that a number of computer codes used for calculations associated with HBU fuel may require additional validation (i.e. codes for nuclide kinetics and strength parameters).

Finally, the presentation emphasized the importance of performing independent confirmatory calculations to verify key safety parameters during the safety review.

### 6.1.4. Elements on the research information letter on state of knowledge on Fuel Fragmentation Relocations and Dispersal

Mr. James Corson (U.S. NRC, United States of America) presented information concerning a compilation of results and insights from research performed on the topic of fuel fragmentation, relocation, and dispersal.

In December 2021, staff from the U.S. NRC published Research Information Letter (RIL) 2021-13 [41]. RIL 2021-13 provides the staff's interpretation of recent research on FFRD. It focuses on five elements related to FFRD:

1. The conditions at which fuel becomes susceptible to fine fragmentation;
2. The cladding strain above which fuel can relocate axially within the fuel rod;
3. The mass of 'dispersible' fuel;
4. Transient FGR in LOCA conditions;
5. The packing fraction of fuel fragments within the balloon region of the fuel rod.

Notably, RIL 2021-13 provides the following conservative interpretations of the available data: fuel becomes susceptible to fine fragmentation starting at a pellet average burnup of 55 GWd/tU and to axial relocation once cladding hoop strain exceeds 3%. RIL 2021-13 also provides a conservative model for fuel dispersal based on the aforementioned burnup and strain thresholds, as well as significant uncertainty in the size of the burst opening.

RIL 2021-13 only includes information about the behaviour of uranium dioxide fuel in zirconium alloy cladding; it does not address the potential impacts of adding dopants (e.g. gadolinium, chromium) to the $UO_2$ fuel or of changes to the cladding design (e.g. chromium coated zirconium alloy cladding). The report acknowledges that FFRD thresholds are defined in terms of surrogate parameters like burnup or cladding strain. There are almost certainly other parameters influencing FFRD, but more research is needed to develop more mechanistic models. Finally, RIL 2021-13 only addresses the behaviour of a single fuel rod under LOCA conditions. However, the information in this report could be applied as part of a full core LOCA analysis to estimate the potential mass that could be dispersed under these conditions. Such estimates could then be used to evaluate the potential consequences of fuel dispersal. U.S. NRC is actively engaged in experimental and analytical research efforts to address some of these limitations, as described in RIL 2021-13 [41] and in Ref. [42].

#### 6.1.5. Assessment of the U.S. Nuclear Regulatory Commission's regulatory framework for the licensing of high burnup fuel

Mr. John Lehning (U.S. NRC, United States of America) presented on how the U.S. NRC is preparing to perform independent safety reviews of anticipated licensing applications to permit reactor licensees to increase end of life (EOL) fuel burnups beyond currently approved limits (e.g. 62 GWd/tU rod-average burnup). Proposals to increase fuel burnup may involve impacts across the nuclear fuel cycle that the NRC is assessing to assure adequate protection of the public.

In preparation for these anticipated safety reviews, the U.S. NRC has performed an assessment of the existing regulatory framework for the licensing of advanced fuel designs, including HBU fuel, fuel with increased enrichment, and ATF [43]. Key impacts of HBU fuel considered in the assessment include the following:

- Fuel dispersal during a LOCA;
- Fuel dispersal during accidents other than the LOCA;
- Cladding embrittlement;
- Radiation dose;
- Fuel storage and transportation;
- Environmental impacts.

In light of industry plans to propose increased fuel burnup limits, the U.S. NRC is considering modifications to regulations and guidance across the fuel cycle to promote the safe, consistent, and efficient licensing of HBU fuel. Licensing HBU fuel is a major effort, and significant work remains underway to resolve attendant technical and regulatory issues, particularly those associated with dispersed fuel fragments. Any regulatory or policy changes associated with HBU fuel would involve deliberation and decision from the Commission.

### 6.2. SUMMARY OF DISCUSSIONS

A number of key points arose from the presentations in this session on licensing experience and regulatory perspectives:

- While a physical understanding has been achieved for many fundamental processes associated with operating fuel at increased burnup, considerable uncertainty may exist in specific areas for certain fuel designs (e.g. transient FGR, behaviour of dispersed fuel fragments).
- Regulatory decisions concerning HBU fuel need to take into account such uncertainties in an appropriate manner. Different strategies were discussed during the technical meting for addressing uncertainties, including consideration of risk information and the application of conservative assumptions.
- Regulatory bodies would benefit from remaining abreast of ongoing research and analysis intended to address existing uncertainties and knowledge gaps, and from continuing to apply insights from these efforts in future regulatory decisions.
- Regulatory bodies will need to assess licensees' submissions raising potential impacts of increased burnup limits across all relevant portions of the fuel cycle. Namely, in addition to gaining confidence of safety under in-reactor operating conditions, safety impacts associated with activities such as transportation, short- and long-term spent fuel storage, and new fuel storage will also have to be addressed by licensees.
- Regulatory bodies that anticipate reviewing industry proposals to operate with HBU fuel may benefit from performing an early assessment of the need for major regulatory or policy changes. The implementation of regulatory or policy changes may involve a significant lead time to identify and assess options and interact with public stakeholders. An example of the potential range of regulatory activities and associated lead times to support the licensing of certain types of advanced fuel designs (including acceptance of increased burnup limits) is shown in Figure 1 in a "Roadmap to Readiness" graphic issued by the US NRC in June 2023.

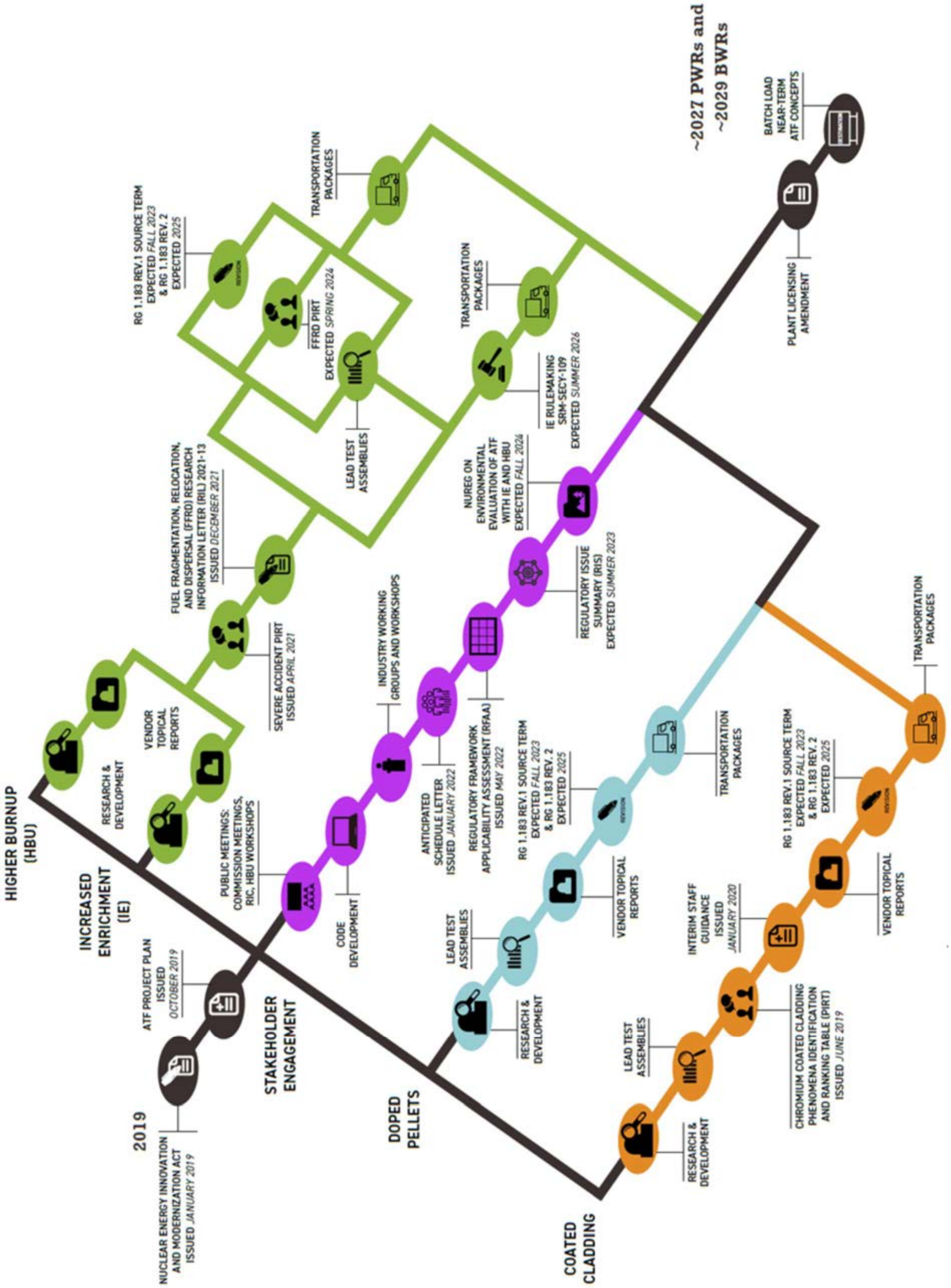



*FIG. 1. ATF licensing roadmap [44], courtesy of US-NRC, United States of America*

## 7. CONCLUSIONS AND PERSPECTIVES FROM THE TECHNICAL MEETING

### 7.1. CONCLUSIONS

The objective of this technical meeting was to exchange information on the safety and performance of HBU nuclear fuels (e.g. beyond 62 GWd/tU) for WCRs, considering their development, qualification and licensing.

It is clear that current nuclear fuels have been developed, qualified, licensed, and operated within the existing WCRs up to the approved burnup limits. Significantly higher burnup rates have been reached in some Member States (e.g. up to a rod average burnup between 70 and 80 GWd/tU in Sweden and Switzerland, see Table 1 in Section 2). Depending on the strategy for the fuel cycle, including the back end, there may be an economic incentive to further improve the efficiency of fuel use (e.g. longer cycles or power uprate) by increasing fuel enrichment and burnup. This fuel burnup increase is feasible in the near term with the existing fuels for limited burnup extensions (e.g. up to a rod average burnup of 68–70 GWd/tU), and in future for higher burnup limits (e.g. up to a rod average burnup of 75 GWd/tU) with ATFs.

The implementation of HBU fuels also needs considerable efforts to better understand and address uncertainties related to the following key phenomena:

- Transient Fuel Gas Releases (FGR);
- Fuel Fragmentation, Relocation and Dispersal (FFRD);
- Fuel coolant interaction after clad failure (RIA transient).

Improved modelling and fuel performance codes and BEPU safety analysis methodology are necessary to assess the above issues in all plant states including accidental conditions.

A large number of experimental data are available on HBU fuel behaviour, allowing the development and validation of the fuel rod codes. However, new tests with representative fuels are needed to support the development and validation of fuel performance codes for the above key HBU phenomena within the entire design basis. The consideration of the following items in the new tests may be warranted:

- Improved pre-and post-test characterization of the test rods;
- Quantification of uncertainties associated with reported experimental results.

To support the qualification and licensing of HBU fuels, lead test ATF rods or assemblies (LTRs and LTAs) are being irradiated in several NPPs worldwide, and pool-side inspections and post-irradiation examinations are being performed.

While roadmaps have been developed in certain Member States for licensing HBU, high enrichment and ATF (e.g. U.S. NRC), substantial benefits will be achieved through consistent licensing frameworks for HBU nuclear fuels.

### 7.2. PERSPECTIVES FOR FUTURE ACTIVITIES

The following activities were proposed by participants of the technical meeting to address the needs identified during the meeting.

### 7.2.1. Knowledge of phenomena and modelling and simulation

FFRD in LOCA and RIA transients remains a key issue for licensing of fuel to higher burnup. Any efforts that can help to further understand the phenomena and improve the simulation capabilities will be of great value to the international community, with particular focus on:

- Transient FGR, and the axial gas flow in the fuel and the gap;
- Cladding rupture strain and opening characteristics;
- Fuel relocation and dispersal in LOCA transients;
- Risk-informed assessment of LOCA and FFRD impacts;
- Development and application of BEPU approach to evaluate the impact of FFRD in safety analysis for LOCA and RIA.

Discussions at the meeting concluded the beneficial role that a steering group could play to monitor and coordinate the related activities worldwide.

### 7.2.2. Experimental programmes

There are plans for activities in this area that aim to work towards filling existing gaps, but there are other programmes with valuable research where data and results are perhaps not well known (or with limited access). The proprietary nature of vendor data may sometimes limit access to (and information from) potentially important material-specific experimental results.

The need for coordinating efforts between research programmes was recognized and acknowledged by the research institutions. The meeting participants suggested that the IAEA might contribute by facilitating sharing safety related information on the following topics (see also Section 7.2.4):

- Comparison of the central restructuring zone characterization for different types of fuel ($UO_2$, MOX, doped fuels at different burnup) obtained by different laboratories;
- Further assessment of the possible impact of considering base power history;
- Studies to further assess the impact of new ATF (cladding) types in possibly reducing the potential for fuel dispersal;
- Main phenomena contributing to ballooning and burst opening size, and the impact of these on FFRD.

### 7.2.3. Licensing

International meetings and cooperative activities on HBU fuel provide a vital means for the sharing of safety relevant research results, insights, and operating experience.

From a regulatory and licensing perspective, continued international engagement and cooperation on the topic of HBU fuel promotes several desirable outcomes:

- Incorporation of up to date insights from ongoing research in the development of regulatory positions and advanced analytical tools for HBU fuel;
- Incorporation of lessons learned and best practices from regulatory bodies around the world that are performing safety reviews for HBU fuel;
- Consistency in technical and regulatory decisions made by regulatory bodies concerning HBU fuel.

### 7.2.4. Potential future international cooperation

Within this framework to address the performance and safety issues encountered during the implementation of HBU fuel, international organizations, such as IAEA, could possibly provide valuable support to multilateral cooperation to tackle some of the challenges above through:

- Establishment of a steering group for HBU fuels to monitor and coordinate the related activities worldwide, by organizing workshops and technical meetings.
- Organization of technical meetings to promote sharing of technical and scientific knowledge, and to promote consistency in technical and regulatory decisions made by regulatory bodies, including on (but not limited to) the following topics:
  - FFRD;
  - Cladding rupture strain and opening characteristics;
  - Transient FGR;
  - Development and validation of fuel rod codes and BEPU method for assessing relevant phenomena;
  - Fuel qualification and licensing process for burnup limit increases in Member States.

- Establishment of joint research programmes, including coordinated research projects (CRPs), to deepen R&D on safety implication of HBU fuel phenomena, possibly involving activities such as:
  - Testing and simulation of FFRD-related phenomena;
  - Benchmarking activity to support adequate validation of relevant models and numerical codes across the full domain of application.

## REFERENCES


[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Knowledge for the Qualification and Licensing of Advanced Nuclear Fuels for Water Cooled Reactors, IAEA-TECDOC-2032, IAEA, Vienna (2023).

[2] ELECTRIC POWER RESEARCH INSTITUTE, Technical Expert Panel Assessment of Existing Fuel Fragmentation, Relocation, and Dispersal Data: Current Understanding and Needs for Future Research, EPRI Report No. 3002025542 (2022), https://www.epri.com/research/products/000000003002025542.

[3] NUCLEAR ENERGY INSTITUTE, The Economic Benefits and Challenges with Utilizing Increased Enrichment and Fuel Burnup for Light-Water Reactors, Nuclear Energy Institute White Paper, NEI, (2019).

[4] VERWERFT, M. and HESKETH, K., “High performance fuel–Potential impact on fuel safety margins and fuel reliability”, EU Post-FISA-2003 Workshop, Luxembourg, 10–13 November (2003).

[5] RUDLING, P. et al., High burnup fuel issues, *Nucl. Eng. Technol.*, 40 1 (2008).

[6] OECD NUCLEAR ENERGY AGENCY, Approach to the resolution of safety issues, Chapter IV Issue: High Burnup Fuel, OECD 2006, NEA/CSNI/R(2006)6 (2006).

[7] OECD NUCLEAR ENERGY AGENCY, Very high burn-ups in light water reactors, OECD 2006, ISBN 92-64-02303-8 (2006).

[8] WESTINGHOUSE ELECTRIC COMPANY LLC., PRIME™ Advanced Fuel Features Package, https://www.westinghousenuclear.com/Portals/0/flysheets/NFCM-0008%20PRIME.pdf.

[9] FRAMATOME, PWR Fuel Assembly GAIA, https://www.framatome.com/solutions-portfolio/portfolio/product?product=A0528.

[10] KAROUTAS, Z. et al., “Westinghouse EnCore accident tolerant fuel and high energy program”, Proc. Int Conf. TopFuel2022, Railegh, NC, 9–13 October (2022).

[11] FRAMATOME, Framatome’s EATF Program, https://nextevolutionfuel.com/framatome-eatf-program.

[12] LASSMANN, K., TRANSURANUS: a fuel rod analysis code ready for use, *Jour. Nuc. Mat.* 188 (1992) 295.

[13] PIZZOCRI, D., BARANI, T., LUZZI, L., SCIANTIX: A new open source multi-scale code for fission gas behaviour modelling designed for nuclear fuel performance codes, *Jour. Nuc. Mat.* 532 (2020), https://doi.org/10.1016/j.jnucmat.2020.152042.

[14] KREMER, F., DUBOURG, R., CAPPIA, F., RONDINELLA, V. V, SCHUBERT, A., VAN UFFELEN, P., WISS, T., “High Burn up Structure formation and growth and fission product release modelling: new simulations in the mechanistic code MFPR-F”, Proc. Int. Conf. TopFuel2018, Prague, Czech Republic, 30 September–4 October (2018).

[15] ANATECH, FALCON – Fuel Analysis and Licensing Code, Vol. 1: Summary Report on Development Activities, internal report, ANA-97-0230 (1997).

[16] KHVOSTOV, G., MIKITYUK, K., ZIMMERMANN, M.A., A model for fission gas release and gaseous swelling of the uranium dioxide fuel coupled with the FALCON code”, *Nucl. Eng. Des.* 241 8 (2011), 2983-3007. https://doi.org/10.1016/j.nucengdes.2011.06.020.

[17] KHVOSTOV, G., WIESENACK, W., ZIMMERMANN, M.A., LEDERGERBER, G., Some insights into the role of axial gas flow in fuel rod behaviour during the LOCA based on Halden tests and calculations with the FALCON-PSI code, *Nucl. Eng. Des.* 241 5 (2011), 1500-1507, https://doi.org/10.1016/j.nucengdes.2011.03.003.

[18] MOAL, A., GEORGENTHUM, V., MARCHAND, O., SCANAIR: A transient fuel performance code Part One: General modelling description, *Nucl. Eng. Des.* 280 (2014).

[19] RUYER, P., ZHOU, Y., ABBATE, A., ZOU, Z., AUSSILLOUS, P., RULLIÈRE, R., HABERSCHILL, P., "Evaluation of the consequences of fuel dispersion and interaction with coolant following a cladding failure induced by a RIA" Proc. Int Conf. TopFuel 2018, Prague, Czech Republic, 30 September–4 October (2018), https://www.euronuclear.org/archiv/topfuel2018/fullpapers/TopFuel2018-A0081-fullpaper.pdf.

[20] GLANTZ, T., TAURINES, T., DE LUZE, O., BELON, S., GUILLARD, G., JACQ, F., DRACCAR: A multi-physics code for computational analysis of multi-rod ballooning, coolability and fuel relocation during LOCA transients Part one: General modeling description, *Nucl. Eng. Des.* 339 (2018), 269-285.

[21] GLANTZ, T., TAURINES, T., BELON, S., DE LUZE, O., GUILLARD, G., JACQ, F., DRACCAR: A multi-physics code for computational analysis of multi-rod ballooning, coolability and fuel relocation during LOCA transients. Part Two: Overview of modeling capabilities for LOCA, *Nucl. Eng. Des.* 339 (2018), 202-214.

[22] INSTITUT DE RADIOPROTECTION ET SURETE NUCLEAIRE, PERFROI Project, 2019, https://en.irsn.fr/EN/Research/Research-organisation/Research-programmes/PERFROI-project/Pages/PERFROI-project.aspx.

[23] REPETTO, G., et al., "COAL experiments investigating the reflooding of a 7 X 7 rods bundle during a LOCA - Thermalhydraulics results", Proc. 19th Int. Topical Meeting on Nuclear Reactor ThermalHydraulics (NURETH 19), Brussels, Belgium, March 6-11 (2022).

[24] GEORGE, T.L. et al., GOTHIC containment analysis package, Version 3.4e Volume 1: Technical Manual, internal report, EPRI, TR-103053-V3 (1993).

[25] WILLIAMSON, R. L., BISON: A Flexible Code for Advanced Simulation of the Performance of Multiple Nuclear Fuel Forms, *Nucl. Tech.* 207 (2021), 954-980, DOI: 10.1080/00295450.2020.1836940.

[26] WIESENACK, W. et al., Review of Halden reactor project high burnup fuel data that can be used in safety analyses, *Nucl. Eng. Des.* 172 (1997) 83-92.

[27] MENUT, P., SARTORI, E., TURNBULL, J.T., "The Public Domain Database on Nuclear Fuel Performance Experiments (IFPE) for the Purpose of Code Develpment and Validation", Proceedings of Topical Meeting on LWR Fuel Performance (Poster Session), American Nuclear Society, Park City, Utah, 10–13 April (2000).

[28] OECD NUCLEAR ENERGY AGENCY, International Fuel Performance Experiments (IFPE) database (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_36358/international-fuel-performance-experiments-ifpe-database.

[29] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel modelling at extended burnup: report of the Co-ordinated Research Programme on Fuel Modelling at Extended Burnup – FUMEX, IAEA-TECDOC-998, IAEA, Vienna (1998).

[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling at extended burnup (FUMEX-II) – Report of a Coordinated Research Project, IAEA-TECDOC-1687, IAEA, Vienna (2012).

[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling in Accident Conditions (FUMAC) – Final Report of a Coordinated Research Project, IAEA-TECDOC-1889, IAEA, Vienna (2019).
[32] OECD NUCLEAR ENERGY AGENCY, Halden Reactor Project - Fuels and Material (2023), https://www.oecd-nea.org/jcms/pl_24970/halden-reactor-project-fuels-and-material.
[33] OECD NUCLEAR ENERGY AGENCY, Second Framework for Irradiation Experiments, FIDES-II (2023), https://www.oecd-nea.org/jcms/pl_70867/second-framework-for-irradiation-experiments-fides-ii.
[34] OECD NUCLEAR ENERGY AGENCY, Studsvik Cladding Integrity Project, SCIP (2023), https://www.oecd-nea.org/jcms/pl_25445/studsvik-cladding-integrity-project-scip.
[35] OECD NUCLEAR ENERGY AGENCY, State-of-the-art Report on Nuclear Fuel Behaviour Under Reactivity-initiated Accident Conditions (2022), https://www.oecd-nea.org/jcms/pl_74371/state-of-the-art-report-on-nuclear-fuel-behaviour-under-reactivity-initiated-accident-conditions-ria-soar.
[36] CHINA NATIONAL NUCLEAR CORPORATION ,capable of mass producing CF3 fuel assembly, https://en.cnnc.com.cn/2019-05/12/c_435851.htm, 2019.
[37] XIAO, M., ZHANG, H., MA, C., BAI, C., ZHOU, Z., WANG, L. and XIAO, X., Practice and prospect of advanced fuel management and fuel technology application in PWR in China, *J. Nucl. Sci. Technol.* 52 10 (2015) 1226–1231, http://dx.doi.org/10.1080/00223131.2015.1029557.
[38] HALLMAN L.H., Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel, internal report, Westinghouse Electric Company, WCAP-18482-NP-A Revision 0, September 2022.
[39] WESTINGHOUSE ELECTRIC COMPANY LLC., Westinghouse AXIOM Cladding in Pressurized Water Reactor Fuel, internal report, Westinghouse Electric Company, WCAP-18546-NP-A Revision 0, March 2023.
[40] WESTINGHOUSE ELECTRIC COMPANY LLC., Optimized ZIRLO™, internal report, Westinghouse Electric Company, WCAP-12610-NP-A & CENPD-404-NP-A, Addendum 1-A, July 2006.
[41] BALES, M., CHUNG, A., CORSON, J., KYRIAZIDIS, L., Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup, RIL 2021-13, U.S. Nuclear Regulatory Commission, Washington, DC (2021), ML21313A145.
[42] CORSON, J., CHUNG, A., KYRIAZIDIS, L. AND BALES, M., “NRC’s Research Information Letter on Fuel Fragmentation, Relocation and Dispersal”, Proc. Int Conf. TopFuel2022, Raleigh, NC, October 9–13 (2022), ML22151A117.
[43] NUCLEAR REGULATORY COMMISSION, Regulatory Framework Applicability Assessment and Licensing Pathway Diagram for the Licensing of ATF-Concept, Higher Burnup, and Increased Enrichment Fuels (2022), ML22014A112.
[44] NUCLEAR REGULATORY COMMISSION, Accident Tolerant Fuel (ATF), Increased Enrichment, and Higher Burnup Roadmap to Readiness (2023), ML23158A294.

## ANNEX. PROCEEDINGS OF THE TECHNICAL MEETING HELD IN VIENNA, AUSTRIA, FROM 14TH TO 18TH OF NOVEMBER 2022

**TECHNOLOGY DEVELOPERS' EXPERIENCE (SESSION IV)**



**LICENSING EXPERIENCE AND REGULATORY PERSPECTIVE (SESSION V)**

UTILITY PERSPECTIVE AND EXPECTATIONS FROM HBU NUCLEAR FUELS

(Session I)

**Chairperson**

**A. CSONTOS**

United States of America

# THE DRIVE TO LICENSE AND DEPLOY ATF, LEU+ AND HBU FUELS IN THE UNITED STATES OF AMERICA

A. CSONTOS
Nuclear Energy Institute
United States of America

## 1. INTRODUCTION

The objective is to safely and economically enable 24-month cycle operation for the entire fleet of existing LWR, with burnups up to ~75 GWd/tU, associated to enrichment extensions towards LEU from 5% to 10% (LEU+) beyond legacy limits in the mid-2020s.

The improved fuel performance aspects of ATF products will enable utilities to realize fuel cycle benefits. Fuel costs comprise approximately 20–25% of a NPP's total generating costs. The cost of this fuel is directly impacted by the efficiency of a core design developed to meet the plant's energy objectives. ATF products have fuel performance benefits, which can be realized in core design efficiency and fuel reliability and are expanded upon below.

## 2. INCREASED CYCLE LENGTH AND BURNUP, AND REDUCED BATCH SIZES

Cycle length and discharge burnup provides the largest opportunity to significantly impact NPP operating costs and strategies. Furthermore, cycle length affects the NPP's total energy production, fuel costs, and outage costs. Most early NPP designs were assumed to operate with 12-month intervals between refuelling outages. However, many US NPPs have extended their nominal cycle lengths to 18- or 24-months, while many European and Asian NPPs continue to operate with 12- to 15-month cycles. Extending operating cycles would reduce the number of times an NPP would need to shut down for refuelling over the course of its operating life, reducing worker exposure and outage costs.

The increased uranium density in the fuel pellets provided by some ATF concepts enables more efficient fuel usage, decreasing the number of fresh fuel assemblies needed to achieve a specific energy output. This can reduce fuel costs and the number of assemblies that enter the back end of the fuel cycle (i.e. storage and disposal). This will lower worker dose and utility costs.

ATF products can enable increased burnup and enrichment because these concepts provide enhanced performance during postulated accidents that are expected to be a limiting factor in HBU applications. A sufficiently increased allowable discharge burnup would allow NPPs to utilize some fuel assemblies for a longer time in the core, resulting in a more efficient use of the fuel. As a result, fewer fresh fuel assemblies would be required for each cycle. This would reduce the component costs of fuel including fabrication, conversion, and mining. These smaller batch sizes also could enable over a billion dollars in savings for the back end of the fuel cycle over the life of the reactor fleet.

## 3. CONCLUSION

With the current burnup and enrichment limits, a few PWR plants (about 20%) can economically operate on a 24-month fuel cycle basis in the U.S. (while this cycle length is the virtual basis for U.S. BWR reactors operation). Increased burnup provided by ATF concepts coupled with increased enrichment would remove a key limitation in core design resulting in additional fuel cycle flexibility that could permit many PWRs to operate economically on a 24-month cycles.

The net savings for the fleet, assuming all PWR and BWR plants not currently operating on a 24-month cycle switch to a 24-month cycle, is $3.1 B or an average annual saving of $1.5M per reactor per year assuming the reactors have a 60 year operating life. Additionally, the number of dry casks needed to store spent fuel would be reduced by ~500 casks. If operation continues for an 80 year operating life, the net savings increases to $12.5B or an annual savings of ~$2.3M per reactor per year, and the number of dry casks needed to store spent fuel is reduced by ~1800. Additional details are described in the NEI White Paper [1].

## REFERENCES

[1] NUCLEAR ENERGY INSTITUTE, The Economic Benefits and Challenges with Utilizing Increased Enrichment and Fuel Burnup for Light-Water Reactors, Nuclear Energy Institute White Paper, NEI (2019).

# SAFETY AND PERFORMANCE OF HIGH BURNUP FUELS: A UTILITY'S PERSPECTIVE

J. ZHANG
Tractebel (ENGIE)
Belgium

A. MEERT
ENGIE-Electrabel
Belgium

## 1. INTRODUCTION

A nuclear utility's motivation for using HBU fuels is driven by the reduction of reload batch sizes and longer fuel cycles (e.g. from 12 months to 18 or 24 months), that may lead to potential improvement of fuel cycle economics for NPPs due to a positive balance between front end and back end costs [1–4]. However, there are constraints and considerations for implementing HBU fuels that need to be assessed globally before deciding on their feasibility:

- Performance aspects: Fuel design available and qualified? Core design feasible? Fuel and core design codes validated? Design methods demonstrated? Margins sufficient?
- Safety aspects: Fuel design and/or safety criteria applicable? Fuel safety analysis codes validated? Safety analysis methodologies approved? Margins sufficient?
- Economic aspects: licensing efforts? Impact on reprocessing or back-end cost? Stability to neutron irradiation?

Tractebel (ENGIE) has performed a feasibility study in 2007–2009 to assess the HBU safety and performance issues, which has been presented at previous IAEA Technical Meetings on HBU in 2006 [5] and 2009 [6]. The feasibility in safety and performance with limited burnup increase was confirmed for some Belgian NPPs, as follows:

- Up to a fuel assembly average burnup of 58 GWd/tU with existing fuel designs, adapted codes and methods for fuel design and safety analyses;
- Up to a fuel assembly average burnup of 62 GWd/tU or higher with advanced fuel designs, improved codes and methods for fuel design and safety analyses.

The licensing effort was the most important economic constraint and would be justified only by a plant life extension. Therefore, in the absence of plans for life extension, a burnup extension was judged to be not profitable for all Belgian NPPs, and hence was not implemented.

This paper presents the progress made since the last IAEA Technical Meeting on HBU fuels on HBU fuel performance, safety and licensing aspects. The perspectives for an international cooperation on safety and performance of HBU fuels are proposed to simplify or accelerate the licensing process.

## 2. FUEL PERFORMANCE ASPECTS

Currently, a number of fuel products have been developed and qualified for HBU:

- Existing fuel designs (RFA-2 Opt., AGORA or AFA-3Gi): qualified for a limited burnup extension to a maximum fuel assembly burnup up to 58-60 GWd/tU;

- Advanced fuel designs (NGF, PRIME™, GAIA or HTP): qualified (NGF, GAIA, HTP) or being qualified (PRIME) for a maximum fuel assembly burnup above 60 GWd/tU;
- ATF or evolutionary ATF (ENCORE™, EATF™, ARMOR™) with higher enrichment fuels (>5%): still under development and qualification for a maximum fuel assembly burnup above 60 GWd/tU.

New fuel products need to be fully qualified and licensed for HBU applications. This requires:

- Validation of fuel rod thermal mechanical design or performance codes for all key physical phenomena at HBU during normal operation and anticipated operational occurrences:

    - Rim formation (associated to HBS);
    - Thermal conductivity degradation;
    - Enhanced FGR;
    - Enhanced water-side corrosion;
    - Hydrogen pickup (HPU) and secondary hydriding, etc.

  New or updated codes (PAD5, STAV7, TREQ, COPERNIC, GALILEO, etc.) need to be fully verified and validated against an extended experimental database, and licensed for the target maximum fuel rod average burnup up to a maximum BU rate of 70–75 GWd/tU.

- Application of advanced fuel rod design methodologies (if needed):

    - To define more realistic and physical design/safety criteria (e.g. technological limits based on cladding or fuel temperatures, stress or strain);
    - To include adequate provisions or conservatisms, while keeping realisms, in the initial and boundary conditions (e.g. using more realistic design power histories to replace the conservative bounding power histories);
    - To treat uncertainties in a statistical approach (e.g. using BEPU methodologies).

  New methods need to be demonstrated and licensed.

- Verification of the key fuel design criteria at higher burnups:

    - Rod internal pressure (non-lift-off limit to be justified);
    - Cladding oxidation;
    - Hydrogen pickup;
    - Rod and assembly growth;
    - Stress and strain in anticipated operational occurrences (pellet–cladding mechanical interaction or PCMI, pellet–cladding interaction and stress corrosion cracking).

  Design and operating margins need to be quantified by the fuel vendors for the target burnup limits.

- Use of new fuel and cladding materials or new fuel designs in case of presence of insufficient margins with the existing fuel products:

    - Advanced claddings (AXIOM, Q12, Coated Optimized Zirlo or M5, etc.);
    - Doped pellets (ADOPT, etc.);

    - HBU grids and/or guide tubes and/or nozzles, etc.

  Irradiation feedback of LTA is needed for the qualification of new materials and fuel designs.

## 3. FUEL SAFETY ASPECTS

Higher burnup may have significant effects on fuel safety in accident conditions, in particular:

- FFRD;
- Embrittlement due to hydriding during LOCA;
- Hydrogen assisted PCMI brittle failure during RIA;

In the presence of higher burnups, design bases and safety analyses need to be updated. This requires:

- Demonstration of the adequacy of fuel safety criteria for LOCA and RIA [7], including:

    - Large amount of experimental data (legacy results obtained at Argonne National Laboratory, Halden reactor, Studsvik, NSRR, CABRI, MIR, etc.) available to identify key phenomena and to quantify the uncertainties;
    - Identification of LOCA and RIA fuel damage and/or failure mechanisms at HBU;
    - State of the art reports (SOARs) for LOCA and RIA fuel behaviour updated at OECD NEA [8-9];
    - Identification of open issues (e.g. in status report on FFRD [10]): Halden LOCA tests IFA-650 analysed and/or confirmed, cladding ballooning and burst and fuel relocation models improved or developed, and model adequacy assessed;
    - Ongoing international cooperation (IAEA, OECD NEA, NEI, EPRI, PWROG…)
    - Burnup dependent safety criteria, as proposed by the U.S. NRC and the regulatory bodies of other countries:
        - U.S. NRC 10CFR50.46c LOCA criteria proposed in 2016 [11], rulemaking still awaited;
        - U.S. NRC RG1.236 RIA criteria published in 2020 [12] for forward-fitting;
        - French LOCA and RIA criteria defined in 2017 [13-14].

- Demonstration of the adequacy of codes and methodologies for LOCA and RIA:

    - Adaptation of safety analysis models and codes to predict fuel behaviour and performance at HBU.
    - Best estimate neutronic and thermohydraulic and thermomechanical codes and their coupling need to be validated to realistically simulate key physical phenomena at HBU.
    - Advanced methodologies (e.g. BEPU or integrated probabilistic and deterministic safety analysis) are being developed and demonstrated to more realistically consider uncertainties and adequate conservatisms at higher burnups.

  The new models, codes and methodologies need to be licensed for HBU applications.

- New safety analyses to quantify the margins regarding the new safety criteria for the target burnups.

  Accidents, for which the safety criteria are directly impacted by the increased burnup, need to be re-analysed:

  - LOCA;
  - RIA;
  - RCS pump locked rotor.

  Two options can be taken depending on the impact on the margins and a possible modification of nuclear key safety parameters (NKSPs):

    - Fuel safety evaluation based on the fuel interface file for the nuclear steam supply system from the current licensing base safety analysis (in case of sufficient margins and no change of NKSPs);
    - Complete re-analyses (in case of insufficient margins and/or change of NKSPs).

  Other accidents that are not directly impacted by the increased burnups will be also assessed:

    - Justifications for non-reanalysis in case of sufficient margins in the current licensing basis safety analyses and in the NKSPs (e.g. FΔH);
    - Re-analyses in case of insufficient margins in the current licensing basis safety analyses and in the NKSPs (e.g. FΔH).

- The new safety analysis needs to consider at least the following aspects (additional details are provided in Refs. [15–28]):

  - Consideration of burnup effects and thermal conductivity degradation in the LOCA safety analyses;
  - Use of up-to-date fuel rod design codes for reference fuel initial conditions (e.g. fuel temperatures, cladding corrosion, hydrogen contents, rod internal pressures);
  - Use of improved fuel rod transient codes to consider burst and FFRD;
  - Consideration of uncertainties and adequate margins in operating conditions and models.

- Other safety assessments are also necessary:

  - Radiological consequence calculations (source terms);
  - Environment impact assessment;
  - System verifications;
  - Operation procedures adaptations;
  - Spent fuel storage assessment and modifications (e.g. criticality safety analysis of spent fuel pools for higher heat loads);
  - Transportation casks.

As a result of these safety assessments, some modifications to the plant systems and equipment may be necessary.

## 4. FUEL LICENSING ASPECTS

The current fuel average discharge burnup limits were set in the 1980's and 1990's in different Member States, along with the technology advancement, periodic safety review or major plant and/or core modifications, safety analysis and licensing, e.g.:

- USA: the limit of fuel rod average burnup (BU < 62 GWd/tU) was specified in most of the licensing topical reports for products, codes and methods;
- Belgium: The limit of fuel assembly average burnup (BU < 55 GWd/tU for $UO_2$ and 50 GWd/tU for MOX (equivalent to fuel rod average BU < 62 GWd/tU as in the USA) was specified in the royal decrees for each NPP, together with the rated power, maximum enrichment, and/or irradiation time.

A major modification to those limits or requests for licensing of burnup extension requires a new safety analysis and licensing process:

- The request for a burnup extension needs to be licensed on a case-by-case basis (e.g. in Belgium), depending on the magnitude of the burnup limit increase considered;
- The licensees will need to ask the vendors to develop the data and tools (codes and methods) needed to support and justify changes to the current limits;
- Sufficient experience feedbacks are needed to support the licensing process.

A specific licensing framework will need to be defined with the regulatory body (e.g. in Belgium), typically following the licensing process for major modifications, including:

- Safety analysis programme;
- Accident analyses including reference core and fuel design;
- Radiological consequence calculations;
- Environment impact assessment;
- System verifications for impact of boron concentration and residual heat removal;
- Licensing (questions/responses, safety evaluation report);
- Final safety analysis report update;
- Authorization.

Depending on the level of novelty of the codes and methods and the target burnup limits, the licensing process may be long and detailed. Therefore, early identification of new codes and methodologies to be licensed by the regulatory body is needed to set up a realistic licensing plan.

## 5. DISCUSSION AND CONCLUSION

Significant progresses have been made since 2009:

- Better understanding of the fuel behaviour at HBU;
- Development and qualification of improved fuel designs for HBU;
- Revision of the LOCA and RIA safety criteria, based on tests of fuels at HBU;
- Improvement and validation of fuel codes for modelling and simulations of fuel behaviour at HBU;
- Development and licensing of advanced fuel design and safety analysis methodologies.

However, there is a lack of a harmonized HBU fuel licensing process, which prevents the utilities from requesting and implementing the burnup extension.

International cooperation on the safety and performance aspects and harmonization of the licensing process will be helpful for the utilities to implement HBU fuels.

## REFERENCES


[1] VERWERFT, M. and HESKETH, K., “High performance fuel – Potential impact on fuel safety margins and fuel reliability”, EU Post-FISA-2003 Workshop (2003).

[2] RUDLING, P. et al., High burnup fuel issues, Nucl. Eng. Des. 40 1 (2008).

[3] OECD NUCLEAR ENERGY AGENCY, Approach to the resolution of safety issues, Chapter IV Issue: High Burnup Fuel, OECD 2006,NEA/CSNI/R(2006)6 (2006).

[4] OECD NUCLEAR ENERGY AGENCY, Very high burn-ups in light water reactors, OECD 2006, NEA No. 6224 (2006).

[5] ZHANG, J. et al., “Perspectives on a limited burnup extension for Belgian PWRs”, paper presented at IAEA Technical Meeting on High Burnup Fuel Experience and Economics, Sofia, Bulgaria, 26–28 September 2006.

[6] ZHANG, J. et al., “The Belgian approach to assessing and licensing high burnup fuel safety issues”, paper presented at IAEA Technical Meeting on High Burnup Fuel Safety Issues, Vienna, Austria, 1–4 December 2009.

[7] OECD NUCLEAR ENERGY AGENCY, Nuclear Fuel Safety Criteria Technical Review - Second Edition, Nuclear Safety, OECD, 2012, NEA No. 7072 (2012).

[8] OECD NUCLEAR ENERGY AGENCY, State of the Art Report on Nuclear Fuel Behavior under LOCA Conditions, OECD, NEA report N°7483 (in press).

[9] OECD NUCLEAR ENERGY AGENCY, State of the Art Report on Nuclear Fuel Behavior under RIA Conditions, OECD 2022, NEA No. 7575, (2022).

[10] OECD NUCLEAR ENERGY AGENCY, Report on Fuel Fragmentation Relocation, Dispersal, OECD 2016, NEA/CSNI/R(2016)16 (2016).

[11] NUCLEAR REGULATORY COMMISSION, Draft Final Rule – Performance Based ECCS Requirements and Related Fuel Cladding Acceptance Criteria, Staff Memo from V. M. McCree to the Commissioners, SECY-16-0033, 16 March 2016 (ADAMS Accession No. ML15238A933).

[12] NUCLEAR REGULATORY COMMISSION, Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents, RG-1.236, June 2020 (ADAMS Accession No. ML20055F490).

[13] GRAFF, S. et al., “New LOCA and RIA fuel safety criteria in France,” paper presented at 2017 Water reactor fuel performance meeting, Jeju Island, Korea, September 10–14, 2017.

[14] BERNAUDAT, C. et al., “Updated RIA criteria in France,” Proc. Int Conf. TopFuel2018, Prague, September 30–October 4 (2018).

[15] ZHANG, J. and DALLEUR, J.P., "Core and fuel aspects in LOCA safety analysis and reload safety evaluation for Belgian NPPs," Proc. Int Conf. TopFuel2006, Salamanca, Spain, 22-26 October (2006).

[16] ZHANG, J., HUYGEN, G., MEERT, A., SHOCKLING, M.A., BARBER, K.J., SECONDI, F., DONCEL GUTIERREZ, N., PASTOR GARCIA, I., “Quantification of Margins to USNRC 10 CFR 50.46c LOCA Criteria for Belgium Nuclear Power Plants,” Proc. Int Conf. TopFuel2021, Santander (Spain), 24–28 October (2021).

[17] RAYNAUD, P., PORTER, I., “Predictions of Fuel Dispersal During a LOCA, Proceedings of WRFPM 2014, Sendai, Japan, Sep. 14–17 (2014).

[18] ZHANG, H., BLAKELY, C., YU, J., STEWART, R., ASGARI, M., Fuel Rod Burst Potential Evaluation under LOCA Conditions for an Existing Plant with Extended

Burnup Exceeding the Current Limit by 20%, internal report, Idaho National Laboratory, INL/EXT-19-55888, September 2019.
[19] SHOCKLING, M.A., EVERHARD, A.M., LAM, H. Q., BACHRACH, U. “Assessment of loss-of-coolant accident fuel dispersal for high burnup core designs,” Proc. Int Conf. TopFuel2021, Santander (Spain), 24–28 October (2021).
[20] MARCIULESCU, C., SUNG, Y. and BEARD, C. L., “Three-dimensional analysis of 3-loop PWR RCCA ejection accident for high burnup,” Proc. Int. Conf. on Nuclear Engineering (ICONE14), Miami, Florida, USA, July 17–20 (2006).
[21] KÜHNEL, K. and JENICHE, D., “PWR rod ejection analysis with the 3-dimensional transient core simulator PANBOX,” Proc. Int Meet. on LWR fuel performance, Orlando, FL, USA, September 19–22 (2004).
[22] GENSLER, A. et al., LWR core safety analysis with AREVA’s 3-dimensional methods, ATW Internationale Zeitschrift für Kernenergie, 58 2 (2013) 82–88.
[23] GENSLER, A. et al., PWR core safety analysis with 3-dimensional methods, *Ann. Nucl. Energy* 84 (2015) 131–139, 2015.
[24] GONNET, M., “Coupled thermal analysis applied to the study of the rod ejection accident,” Proc. Int. Conf. on advances in reactor physics (PHYSOR 2012), Knoxville, TN, USA, April 15–20 (2012).
[25] BIGOT, J., SEGARD, K., BROCK, R., PARKER, J., CURCA-TIVIG, F., KUCH, S., “Successful deployment of Framatome advanced PWR codes and methods worldwide,” Proc. Int Conf. TopFuel2018, Prague, September 30–October 4 (2018).
[26] SUDJANA, J., UMIDOVA, Z., ZHANG, J., HAEDENS, M., SCHNEIDESCH, C., “Evaluation of PWR rod ejection accident margins using PANTHER/COBRA and FRAPTRAN,” Proc. 2013 International LWR Fuel Performance Meeting and TopFuel2013, Charlotte, North Carolina, USA, September 15–19 (2013).
[27] ZHANG, J., DETHIOUX, A., DRIEU, T. SCHNEIDESCH, C., “Development and Application of Statistical Uncertainty Analysis Method to Multi-Physics Modelling to Support Reactor Core Design and Operation”, Proc. Int. Conf. BEPU-2018, Lucca, Italy, May 13–18 (2018).
[28] ZHANG, J. and SCHNEIDESCH, C., Application of the BEPU safety analysis method to quantify margins in nuclear power plants, *Nucl. Eng. Des.* 406 (2023).

# UTILITY EXPERIENCE WITH LTA IRRADIATION IN COMMERCIAL REACTORS AT CONSTELLATION

W. GASSMANN
Constellation Nuclear
United States of America

Constellation Nuclear is a utility leader in the programme to develop and deploy new advanced LWR fuels. Constellation has loaded lead test fuel assemblies from all three US nuclear fuel vendors in order to demonstrate numerous improved performance capabilities. Advanced LWR fuels are believed to confer many safety and economic benefits; and chief among them is HBU capability due to the more robust (safer) fuel designs, improved fuel performance, and improved analytical techniques.
Preliminary evaluations conducted by Constellation indicate that HBU operation (typically a rod average burnup exceeding 62 GWd/tU) can enable a direct fuel cycle economic benefit ($ USD Millions per reactor per year). Other potential benefits include flexible grid energy and capacity, green hydrogen, e-methanol, and e-ammonia, support for data centres, and direct air capture of carbon dioxide.

Commercial reactor LTA deployment and feedback is necessary to assist the fuel vendors, regulatory bodies, and industry in full licensing of advanced ATF and HBU fuel types. To this end, four ATF and/or HBU LTA campaigns are currently in progress at Constellation, and an explicit HBU application is in progress or under consideration for all of these LTA campaigns. At Byron-2, Westinghouse ATF LTAs (containing doped $UO_2$ and $U_3Si_2$ ATF fuel and Cr coated zirconium alloy ATF cladding) are under irradiation. At Limerick-2, GNF HBU LTAs (containing standard $UO_2$ and zirconium alloy fuel) are under irradiation. At Clinton, GNF ATF LTAs (containing Cr coated zirconium alloy and advanced steel ATF cladding) are under irradiation, and at Calvert Cliffs-2, a Framatome ATF LTA (containing doped $UO_2$ ATF fuel and Cr coated zirconium alloy ATF cladding) is under irradiation.

Standard methods of post-irradiation examinations (PIEs) are planned for all LTA deployments. Additional PIE is required for a complete characterization of all HBU and ATF phenomena, including gamma scan measurements for fission gas pressure and pellet integrity, detailed corrosion examination for cladding integrity, and destructive and non-destructive examination in hot cells.

These LTA applications have typically required a careful consideration of a number of issues unique to ATF and HBU operation, which need to be addressed in the design and licensing process to ensure reactor safety prior to deployment. These include FFRD in case of LOCA, fuel rod design and assembly mechanical design, analysis methods adequacy, steady state/core physics, thermomechanical and thermal-hydraulic considerations, transient and accident performance, core monitoring system compatibility, fuel handling, storage, and shipping, fuel handling accident, and overall design and licensing basis compliance.

Following the successful completion of LTA programmes, numerous additional issues unique to HBU and ATF operation need to be addressed in the design and licensing process to enable U.S. NRC approval and full reload application of any new fuel type. These issues are typically also considered early in the process of new fuel type development, and include radiological source term/dose, spent fuel pool decay heat and criticality, reactor vessel internals compatibility, RCS chemistry, seismic performance, and reactivity management.

PHENOMENOLOGICAL KNOWLEDGE AND MODELLING AND SIMULATION

(Session II)

**Chairperson**

**V. GEORGENTHUM**

France

# CURRENT STATUS AND PLANNED RESEARCH ACTIVITIES FOR THE TRANSURANUS CODE APPLICATION TO HIGH BURNUP FUELS

P. VAN UFFELEN, A. SCHUBERT, T. WISS, Z. SOTI
European Commission, Joint Research Centre
Karlsruhe, Germany

G. ROBERTSON
Uppsala University, Department of Physics and Astronomy, Applied Nuclear Physics
Sweden

C. GYORI
NucleoCon
Slovakia

F. KREMER
Institut de Radioprotection et Sûreté Nucléaire, Severe Accident Department
France

R. CALABRESE
ENEA
Italy

A. SCOLARO
Ecole polytechnique fédérale de Lausanne
Switzerland

D. PIZZOCRI, L. LUZZI
Politecnico di Milano, Department of Energy, Nuclear Engineering Division
Italy

The developments for modelling HBS [1, 2] in the TRANSURANUS code [3] started together with the experimental investigations in the 1990s. Various aspects of the impact of the HBS [4] on the fuel behaviour have been taken into consideration. First, the relative degradation of the thermal conductivity with an increasing amount of fission damage and products as well as the increased porosity as measured by means of the laser diffusivity technique was incorporated in the phonon term for the expression of the thermal conductivity and the correction factor for porosity [5]. This was also supported by the NFIR project of EPRI [6]. Secondly, the variations of the mechanical properties based on micro-indentation [7-8] in combination with detailed acoustic microscopy analysis were incorporated in the expression for the elastic modulus of the fuel [9–12]. Thirdly, the measurements on the fission product depletion observed by means of EPMA [13–15], together with the evaluations of the porosity and the results of rod puncturing contributed to the first empirical model for fission gas behaviour in the HBS implemented in the TRANSURANUS code. It was based on a detailed analysis of the radial distribution of the Pu build up [16], as well as on the phenomenological evolution of the variation of the Xe concentration in the matrix. At first it was linearly increasing with burnup according to the fission yield. Once a first threshold burnup was exceeded, it was considered that the inert gas atoms would start to diffuse to and accumulate in the micron sized so-called HBS pores. This would lead to a gradual decrease of the Xe concentration in the matrix down to a minimum value of around 0.25 wt%, concomitant with a linear increase of the local porosity. When a second threshold was exceeded, it was assumed that the HBS formation was complete so that any additional fission gas would be released to the free volume in the rod. In a similar way, the HBS porosity build up was saturated at a value of 15%. This first model has been implemented and tested within the D-COM [17] and FUMEX [18] CRPs of the IAEA.

In parallel to this modelling work, a dedicated international HBU rim research project (HBRP) managed by the Central Research Institute of Electric Power Industry in Japan (CRIEPI) [19–24] was carried out in the OECD HRP. In order to better characterize the formation of HBS in terms of local burnup and temperature, specific discs were irradiated. The post irradiation measurement enabled the determination of the corresponding limits. Dedicated modelling work was then carried out to translate such limits in the TRANSURANUS code by means of an effective burnup limit [25]. This accounts for the defect annealing above a threshold temperature around 1000 °C.

Newer post-irradiation examinations (PIE) based on secondary ion mass spectroscopy [26, 27] and separate effect tests underlined that the fission gas formed in the pellet rim at HBU was not directly released during normal operation in an NPP as originally assumed. Nevertheless, given the large amounts of fission products contained in the outer parts of the conventional $UO_2$ fuel at large burnup, it was realized that the HBS could have a strong impact during DBAs. Separate effect measurements during a well-controlled temperature ramp of fuel samples in a Knudsen cell [28–30] were used to develop the first model for a release from HBU fuel under LOCA type conditions. Such experiments were complemented with the IFA-650 series [31] in the Halden reactor or the out of pile experiments at Studsvik that provide integral measurements of gas release during such events. Many of the experiments dedicated to HBU fuel behaviour have been included in the IFPE database of the IAEA and NEA [32, 33]. In line with this, the results of the FUMEX series and ensuing FUMAC CRPs have also been added [34, 35]. More recently, the specific RIA tests have also been included, and there is an ongoing discussion about the specific establishment of a database for RIA related experiments and models as a follow-up to the series of RIA benchmarks organised by the NEA [36–38], in which also the TRANSURANUS code was involved.

With the increase of computational power and the advent of improved experimental techniques over time, more detailed models have been developed. This was not only implemented directly in the TRANSURANUS code (for example as part of the first mechanistic model for fission gas behaviour [39,40]), but later also in standalone codes developed for the detailed analysis of fission product behaviour that can be coupled with the TRANSURANUS code to replace the standard models. First there was the SCIANTIX code [41]. The approach for the mechanistic fission gas behaviour model starts from an engineering point of view and progressively increases the level of detail. In a second step, a coupling was made with the MFPR-F code [42], which unlike SCIANTIX, started from the most elementary level of point defects and enabled other fission products and their chemical interactions with oxygen and actinides to be considered. The coupling with both codes has now been demonstrated to simulate irradiated fuel rods during a LOCA in the frame of the European project on 'Reduction of Radiological Accident Consequences' (R2CA).

As far as the cladding behaviour of HBU rods is concerned, model extensions have been introduced to account for the absorbed hydrogen as a result of corrosion of Zircaloy, both during normal operation and in accident conditions [43, 44]. In the frame of the R2CA project, this work was extended to M5 cladding properties [45, 46]. The next step includes the consideration of hydrogen redistribution, precipitation of hydrides and their re-dissolution. For this purpose, a coupling of the HYDCLAD model [47] developed by CIEMAT, is planned in a similar way as it was coupled with the FRAPCON code [48]. The model will take into consideration the most recent developments presented by Passelaigue et al [49, 50] for the BISON code, which consider the nucleation and growth of hydrides as a function of the local hydrogen concentration and temperature. This will also enable a better analysis of the irradiated fuel rod during storage.

Owing to the more detailed description of HBU fuel behaviour in the TRANSURANUS code, in combination with higher computational power, high fidelity simulations for the reactor safety analysis have been initiated within the McSAFE project. These include pin-by-pin simulations of PWRs or WWERs by means of the coupled Serpent-Subchanflow-TRANSURANUS code system [51, 52]. This work is being extended to simulate fuel rods with accident tolerant materials such as FeCrAl cladding and $U_3Si_2$ fuel in various water cooled SMRs such as the Nuscale reactor within the McSAFER project [53].

The increased computational power also prompted research to better account for the uncertainties in the BEPU analysis that is becoming the standard approach. In order to derive an accurate fuel performance code with better quantified uncertainties, the new code needs also to be calibrated and validated against measurements based on modern methods for uncertainty quantification. In a first step, to address model inadequacies, Robertson et al [54] therefore adapted the mean model parameters and their covariance to propagate uncertainties so that they conform with the spread of the residuals (instead of directly calibrating these model parameters). In a second step they developed a Bayesian inverse uncertainty quantification using Markov Chain Monte Carlo. For this purpose, Robertson et al. [55] developed an ensemble of Gaussian-process surrogate models that replace the predictions of several experiments using a separate Gaussian process for each experiment and demonstrated the applicability in the calibration of fuel performance modelling by fitting the surrogate models to FGR predictions. In a next step, it is planned to couple the developed Gaussian-process ensemble with a derivative-based calibration method to incorporate treatment of unknown sources of uncertainty into a Bayesian framework within the Accelerated Program for Implementation of secure WWER fuel Supply. In a similar way, the potential of machine learning is also contemplated within the OperaHPC project, in which industrial fuel performance codes will be improved owing to machine learning and model order reduction techniques and reference three-dimensional simulation results, for example computed with the OFFBEAT code.

## REFERENCES


[1] RONDINELLA, V.V., WISS T., The high burn-up structure in nuclear fuel, *Mater. Today* 13 (2010) 24.

[2] WISS T., RONDINELLA, V.V., KONINGS, R.J.M., STAICU, D., PAPAIOANNOU, D., BREMIER, S., PÖML, P., BENES, O., COLLE, J.Y., VAN UFFELEN, P., SCHUBERT, A., CAPPIA, F., MARCHETTI, M., PIZZOCRI, D., JATUFF, F., GOLL, W., SONODA, T., SASAHARA, A., KITAJIMA, S., KINOSHITA, M., Properties of the high burnup structure in nuclear light water reactor fuel, *Radiochim. Acta* 105 (2017) 893.

[3] LASSMANN, K., TRANSURANUS: a fuel rod analysis code ready for use, *J. Nucl. Mater.* 188 (1992) 295.

[4] VAN UFFELEN, P., SUZUKI, M., Fuel Performance Modeling and Simulations: Oxide Fuel Performance and Simulation, in: R.J.M. Konings (Ed.), Comprehensive Nuclear Materials, Vol. 3, Elsevier Ltd, Amsterdam (2012) 535.

[5] WALKER, C.T., STAICU, D., SHEINDLIN, M., PAPAIOANNOU, D., GOLL, W., SONTHEIMER, F., On the thermal conductivity of $UO_2$ nuclear fuel at a high burn-up of around 100 MWd/kgHM, *J. Nucl. Mater.* 350 (2006) 19.

[6] BARON, D., Fuel thermal conductivity: A review of the modelling available for $UO_2$, (U-Gd)O2 and MOX fuel, Proceedings of Thermal Performance of High Burn-up LWR Fuel, Cadarache, France, 3-6 March 1998, Nuclear Science, OECD 1998, 99–118 (1998).

[7] SPINO, J., VENNIX, COQUERELLE, M., Detailed characterisation of the rim microstructure in PWR fuels in the burn-up range 40–67 GWd/tM, *J. Nucl. Mater.* 231 (1996) 179.

[8] SPINO, J., COBOS-SABATE, J., ROUSSEAU, F., Room-temperature microindentation behaviour of LWR-fuels, part 1: fuel microhardness, *J. Nucl. Mater.* 322 (2003) 204.

[9] MARCHETTI, M., LAUX, D., CAPPIA, F., LAURIE, M., VAN UFFELEN, P., RONDINELLA, V.V., DESPAUX, G., "High Frequency Acoustic Microscopy for the Determination of Porosity and Young's Modulus in High Burnup Uranium Dioxide Nuclear Fuel", paper presented at Advancements in Nuclear Instrumentation Measurement Methods and their Applications (ANIMMA), Lisbon, Portugal, 2015.

[10] CAPPIA, F., PIZZOCRI, D., MARCHETTI, M., SCHUBERT, A., VAN UFFELEN, P., LUZZI, L., PAPAIOANNOU, D., MACIAN-JUAN, R., RONDINELLA, V.V., Microhardness and Young's modulus of high burn-up $UO_2$ fuel, *J. Nucl. Mater.* 479 (2016) 447.

[11] CAPPIA, F., PIZZOCRI, D., SCHUBERT, A., VAN UFFELEN, P., PAPERINI, G., PELLOTTERO, D., MACIAN-JUAN, R., RONDINELLA, V.V., Critical assessment of the pore size distribution in the rim region of high burnup $UO_2$ fuels, *J. Nucl. Mater.* 480 (2016) 138.

[12] MARCHETTI, M., LAUX, D., CAPPIA, F., LAURIE, M., VAN UFFELEN, P., RONDINELLA, V.V., WISS T., DESPAUX, G., High Frequency Acoustic Microscopy for the Determination of Porosity and Young's Modulus in High Burnup Uranium Dioxide Nuclear Fuel, *IEEE Trans. Nucl. Sci.* 63 (2016) 1520.

[13] WALKER, C.T., MOGENSEN, M., On the rate determining step in fission gas release from high burn-up water reactor fuel during power transients, *J. Nucl. Mater.* 149 (1987) 121.

[14] LASSMANN, K., O'CAROLL, C., VAN DER LAAR, J., WALKER, C.T., Radial Distribution in High Burn-Up $UO_2$ Fuels, internal report, Institute for Transuranium Elements, Report No. K0293171, 1993.

[15] LASSMANN, K., WALKER, C.T., VAN DER LAAR, J., LINDSTRÖM, F., Modelling the high burnup $UO_2$ structure in LWR fuel, *J. Nucl. Mater.* 226 (1995) 1.

[16] LASSMANN, K., O'CAROLL, C., VAN DER LAAR, J., WALKER, C.T., The radial

distribution of plutonium in high burnup $UO_2$ fuels, *J. Nucl. Mater.* 208 (1994) 223.
[17] WALKER, C.T., LASSMANN, K., RONCHI, C., COQUERELLE, MOGENSEN, M., The D-COM blind problem on fission gas release: The predictions of the TRANSURANUS and future codes, *Nucl. Eng. Des.* 117 (1989) 211.
[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel modelling at extended burnup: report of the Co-ordinated Research Programme on Fuel Modelling at Extended Burnup – FUMEX, IAEA-TECDOC-998, IAEA, Vienna (1998).
[19] SONODA, T., KINOSHITA, M., RAY, I.L.F., WISS T., THIELE, H., PELLOTTIERO, D., RONDINELLA, V.V., MATZKE, H., Transmission electron microscopy observation on irradiation-induced microstructural evolution in high burn-up $UO_2$ disk fuel, *Nucl. Instrum. Methods Phys. Res.*, 191 (2002) 622.
[20] KINOSHITA, M., KITAJIMA, S., KAMEYAMA, T., MATSUMURA, T., KOLSTAD, E., MATZKE, H., "High Burnup Rim Project, progress of irradiation and preliminary analysis", paper presented at Int. ANS Topical Meeting on LWR Fuel Performance, Lagrange Park, Portland, Oregon, USA, 1997.
[21] SONODA, T., MATZKE, H., KINOSHITA, M., "High Burnup Rim Project (IV) Threshold Burnup of Rim Structure Formation", paper presented at EHPG, Leon, Norway, 1999.
[22] KINOSHITA, M., SONODA, T., KITAJIMA, S., SASAHARA, A., KOLSTAD, E., MATZKE, H., RONDINELLA, V.V., STALIOS, A.D., WALKER, C.T., RAY, I.L.F., SHEINDLIN, M., HALTON, D., RONCHI, C., "High Burn-up Rim Project, (II) Irradiation and examination to investigate rim-structured fuel", paper presented at Int. ANS Topical Meeting on LWR Fuel Performance, Park City, Utah, USA, 2000.
[23] KINOSHITA, M., KOLSTAD, E., MATZKE, H., RONDINELLA, V.V., "High Burnup Rim Project (V) - Completed Scope and Needs for Further Study", paper presented at Enlarged Halden Progamme Group Meeting, Vol. HPR-357, OECD Halden Reactor Project, Lillehammer, Norway, 2001.
[24] KINOSHITA, M., SONODA, T., KITAJIMA, S., SASAHARA, A., KAMEYAMA, T., MATSUMURA, T., KOLSTAD, E., RONDINELLA, V.V., RONCHI, C., HIERNAUT, J.P., WISS T., KINNART, F., EJTON, J., PAPAIOANNOU, D., MATZKE, H., "High-Burnup Rim Project: (III) Properties of Rim-Structured Fuel", paper presented at Int. ANS Topical Meeting on LWR Fuel Performance, Orlando, Florida, USA, 2004.
[25] HOLT, L., SCHUBERT, A., VAN UFFELEN, P., WALKER, C.T., FRIDMAN, E., and SONODA, T., Sensitivity study on Xe depletion in the high burn-up structure of $UO_2$. *J. Nucl. Mater.* 452 (2014), 166–172, https://doi.org/10.1016/j.jnucmat.2014.05.009.
[26] BREMIER, S., HASNAOUI, R., PORTIER, S., BILDSTEIN, O., WALKER, C.T., Installation of a Shielded SIMS for the Analysis of Irradiated Nuclear Fuels, *Mikrochim. Acta* 155 (2006) 113.
[27] PORTIER, S., BREMIER, S., WALKER, C.T., Secondary ion mass spectrometry of irradiated nuclear fuel and cladding: An overview, *Int. J. Mass Spectrom.* 263 (2007) 113.
[28] HIERNAUT, J.P., WISS T., "Fission Gas Release and Microstructural Features in $UO_2$ irradiated from 25 to 95 GWd/t", Proc. of Int. Work. on the High Burn-up Structure in Nuclear Fuels, ITU Karlsruhe, 28–30 June.
[29] SONODA, T., KAMEYAMA, T., SASAHARA, A., KITAJIMA, S., NAUCHI, Y., KINOSHITA, M., RONDINELLA, V.V., WISS T., HIERNAUT, J.P., PAPAIOANNOU, D., SHEINDLIN, M., STAICU, D., "Clarification of rim structure effects on properties and behaviour of LWR UO2 fuels and gadolinia doped fuels", Proc. Int. LWR Fuel Performance Meeting, San Francisco, California, USA, 30 September–3 October (2007).
[30] HIERNAUT, J.P., WISS T., H. Thiele, WALKER, C.T., GOLL, W., KONINGS, R.J.M., Fission product release and microstructure changes during laboratory annealing of a very

high burn-up fuel specimen, *J. Nucl. Mater.* 377 (2008) 313.
[31] OECD NUCLEAR ENERGY AGENCY, "Safety Significance of the Halden IFA-650 LOCA Test Results", OECD 2010, NEA/CSNI/R(2010)5, (2010).
[32] MENUT, P., SARTORI, E., TURNBULL, J.T., "The Public Domain Database on Nuclear Fuel Performance Experiments (IFPE) for the Purpose of Code Develpment and Validation", Proc. of Topical Meeting on LWR Fuel Performance (Poster Session), American Nuclear Society, Park City, Utah, 10–13 April (2000).
[33] OECD NUCLEAR ENERGY AGENCY, International Fuel Performance Experiments (IFPE) database (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_36358/international-fuel-performance-experiments-ifpe-database.
[34] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling in Accident Conditions (FUMAC) – Final Report of a Coordinated Research Project, IAEA-TECDOC-1889, IAEA, Vienna (2019).
[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling at extended burnup (FUMEX-II) – Report of a Coordinated Research Project, IAEA-TECDOC-1687, IAEA, Vienna (2012).
[36] OECD NUCLEAR ENERGY AGENCY, Reactivity Initiated Accident Benchmark Phase III Report, OECD 2022, NEA/CSNI/R(2020)10 (2022).
[37] OECD NUCLEAR ENERGY AGENCY, Reactivity Initiated Accident (RIA) Fuel Codes Benchmark Phase-II: Uncertainty and Sensitivity Analyses, OECD 2017, NEA/CSNI/R(2017)
[38] OECD NUCLEAR ENERGY AGENCY, RIA Fuel Codes Benchmark. Volume 1, OECD 2013, NEA/CSNI/R(2013)7 (2013).
[39] PASTORE, G., Modelling of Fission Gas Swelling and Release in Oxide Nuclear Fuel and Application to the TRANSURANUS Code, Department of Energy, Politecnico di Milano (2012).
[40] PASTORE, G., LUZZI, L., DI MARCELLO, V., VAN UFFELEN, P., Physics-based modelling of fission gas swelling and release in $UO_2$ applied to integral fuel rod analysis, *Nucl. Eng. Des.* 256 (2013) 75.
[41] PIZZOCRI, D., BARANI, T., LUZZI, L., SCIANTIX: A new open-source multi-scale code for fission gas behaviour modelling designed for nuclear fuel performance codes. *J. Nucl. Mater.* 532 (2020) 152042.
[42] KREMER, F., DUBOURG, R., CAPPIA, F., RONDINELLA, V.V., SCHUBERT, A., VAN UFFELEN, P., WISS T., "High Burn Up Structure formation and growth and fission product release modelling: new simulations in the mechanistic code MFPR-F", Proc. Int Conf. TopFuel2018, Prague, September 30–October 4 (2018).
[43] BLAIR, P., "Application of the Westinghouse version of TRANSURANUS in LOCA fuel rod licensing analyses", paper presented at International Workshop "Towards nuclear fuel modelling in the various reactor types across Europe", Karlsruhe, Germany, 2013.
[44] GYÖRI, C., JONSON, M., ROBERTSON, G., BLAIR, P., SCHUBERT, A., VAN UFFELEN, P., "Extension and validation of the TRANSURANUS code in the course of the ESSANUF project", paper presented at 12th International conference on WWER fuel performance, modelling and experimental support (INRNE), Nessebar, Bulgaria, 2017.
[45] CALABRESE, R., "Crystallographic Phase Transition of Zirconium Alloys: New Models For The TRANSURANUS Code", Proc. 31st Int. Conf. Nuclear Energy for New Europe 2022, Portoroz, Slovenia, 12–15 September (2022).
[46] CALABRESE, R., SCHUBERT, A., VAN UFFELEN, P., "Review of M5™ Cladding Models Relevant for LOCA Simulation with the TRANSURANUS Code", Proc. 30th Int. Conf. Nuclear Energy for New Europe 2021, Bled, Slovenia, 6–9 September (2021).
[47] FERIA, F., HERRANZ, L.E., Effect of the oxidation front penetration on in-clad hydrogen migration, *J. Nucl. Mater.* 500 (2018) 349.

[48] FERIA, F., AGUADO, C., HERRANZ, L.E., Extension of FRAPCON-xt to hydride radial reorientation in dry storage, *Ann. Nucl. Energy* 145 (2020) 107559.

[49] PASSELAIGUE, F., SIMON, P.C.A., A. MOTTA, T., Predicting the hydride rim by improving the solubility limits in the Hydride Nucleation-Growth-Dissolution (HNGD) model, *J. Nucl. Mater.* 558 (2022) 153363.

[50] PASSELAIGUE, F, LACROIX, E., PASTORE, G., A. MOTTA, T., Implementation and Validation of the Hydride Nucleation-Growth-Dissolution (HNGD) model in BISON, *J. Nucl. Mater.* 544 (2021) 152683.

[51] GARCIA, M., BILODID, Y., BASUALDO PERELLO, J., TUOMINEN, R., GOMMLICH, A., LEPPÄNEN, J., VALTAVIRTA, V., IMKE, U., FERRARO, D., VAN UFFELEN, P., SEIDL, M., V. Sanchez-Espinoza, Validation of Serpent-SUBCHANFLOW-TRANSURANUS pin-by-pin burnup calculations using experimental data from a Pre-Konvoi PWR reactor, *Nucl. Eng. Des.* 379 (2021) 111173.

[52] GARCIA, M., TUOMINEN, R., GOMMLICH, A., FERRARO, D, VALTAVIRTA, V., IMKE, U., VAN UFFELEN, P., MERCATALI, L., SANCHEZ-ESPINOZA, V., LEPPÄNEN, J., KLIEM, S., A Serpent2-SUBCHANFLOW-TRANSURANUS coupling for pin-by-pin depletion calculations in Light Water Reactors, *Ann. Nucl. Energy* 139 (2020) 107213.

[53] SANCHEZ-ESPINOZA, V., GABRIEL, S., SUIKKANEN, H., TELKKÄ, J., VALTAVIRTA, V., BENCIK, M., KLIEM, S., QUERAL, C., FARDA, A., ABÉGUILÉ, F., SMITH, P., VAN UFFELEN, P., AMMIRABILE, L., SEIDL, M., SCHNEIDESCH, C., GRISHCHENKO, D., LESTANI, H., The H2020 McSAFER Project: Main Goals, Technical Work Program, and Status, *Energies* 14 (2021) 6348.

[54] ROBERTSON, G., SJÖSTRAND, H., ANDERSSON, P., HANSSON, J., BLAIR, P., Treating model inadequacy in fuel performance model calibration by parameter uncertainty inflation, *Ann. Nucl. Energy* 179 (2022) 109363.

[55] ROBERTSON, G., SJÖSTRAND, H., ANDERSSON, P., BLAIR, P., "Surrogate modeling with derivative prediction for implementation in inverse uncertainty quantification methods for fuel performance modelling", Proc. Int Conf. TopFuel2022, Raleigh, NC, October 9–13 (2022).

# CONSISTENT MODELLING OF FUEL FRAGMENTATION, GRAIN-BOUNDARY FRACTURE AND FGR IN LWR FUEL RODS DURING LOCA AND RIA

G. KHVOSTOV
Paul Scherrer Institute
Switzerland

Presented are several amendments made in the FALCON code coupled with the GRSW-A model [1], as well as in the FRELAX code [2], aiming at improvement of calculation for specific types of thermal transients, such as RIA and LOCA.

Because analysis of pre-transient, base irradiation of the fuel is deemed to be an integral part of transient behaviour calculation [3], attention was paid to improvement of this analysis, as well. Specifically, a model was developed [4] and integrated into GRSW-A [5] for trapping of the released fission gases by the specific closed voids ('pockets'), as formed within a pellet–cladding bonding layer during base irradiation. The effect is linked with an azimuthal non-uniformity of fission rate at the fuel pellet edge, and shown to be particularly important for calculation of FGR in BWR HBU fuels.

In the next step, transient analysis with the GRSW-A model of the FALCON code was extended to simulate the processes of fuel fragmentation and concomitant FGR due to grain-boundary fracture [6]. At the stage of base irradiation analysis, the GRSW-A model shows a correlation between the fuel pellet zones fragmentation (and eventually pulverization) and the specific surface area of the emergent vented pores. The transient FGR is deemed to be largely caused by grain-boundary fracture (micro-cracking) due to over-pressure of the gaseous pores. Fuel fragmentation-dependent analytical criteria for the on-set of grain-boundary fracture and, consequently, transient FGR were proposed.

Finally, the FRELAX code was extended for simulation of bulk flow and diffusion of the gases in the rod free volume during thermal transients, e.g. LOCAs [7]. To this end, the FRELAX code considers the fuel rod active part as a system with lumped parameters, where the parameters of simulated processes are ascribed to one of the two separated model elements. In addition, the analysis is fed by integrated models for the dynamic viscosity and diffusivity of arbitrary gas mixtures in an extended range of temperature. The FALCON and FRELAX updated codes coupling (F2F Coupled Code System) allows data exchange during their simultaneous computation.

The model parameters were first, set up based on model calibration using data of a selected NSRR RIA test, and a Halden LOCA test. Then, the updated codes and models were applied to extended sets of RIA (NSRR and CABRI-SL) and LOCA- (Halden) tests, assuming the same set of best-estimate parameters, as defined based on the calibration. The results of calculation compare well with the experimental data. Furthermore, results of calculation with the updated models were compared with the RG 1.236 empirical correlation for FGR during an RIA, as was recently proposed by the U.S. NRC for the use in proper fuel safety calculations [8]. The correlation was shown to be substantially conservative in comparison to the FALCON-based mechanistic modelling, which suggests applicability of such a simpler approach to fuel RIA safety licensing in Swiss LWRs. Concerning the near- and mid-term future activity, apart from additional verification (e.g. for Gd doped fuel, fuel doped with Cr and/or Al additives), consideration is currently taken of updating the Swiss LWR full-core LOCA analysis with a view to finding out whether transient FGR is able to affect cladding ballooning and burst under the conditions of a postulated LOCA transient.

## REFERENCES


[1] KHVOSTOV, G., MIKITYUK, K., AND M. ZIMMERMANN, A., A model for fission gas release and gaseous swelling of the uranium dioxide fuel coupled with the FALCON code, *Nucl. Eng. Des.* 241 8 (2011) 2983–3007, https://doi.org/10.1016/j.nucengdes.2011.06.020.

[2] KHVOSTOV, G., WIESENACK, W., M. ZIMMERMANN, A., AND G. LEDERGERBER, Some insights into the role of axial gas flow in fuel rod behaviour during the LOCA based on Halden tests and calculations with the FALCON-PSI code, *Nucl. Eng. Des.* 241 5 (2011) 1500–1507, https://doi.org/10.1016/j.nucengdes.2011.03.003.

[3] F. RIBEIRO AND KHVOSTOV, G., Multi-scale approach to advanced fuel modelling for enhanced safety, *Prog. Nucl. Energy* 84 (2015) 24–35, https://doi.org/10.1016/j.pnucene.2015.03.022.

[4] BRANKOV, V. et al., Analysis of effects of pellet-cladding bonding on trapping of the released fission gases in high burnup KKL BWR fuels, *Nucl. Eng. Des.* 305 (2016) 559–568, https://doi.org/10.1016/j.nucengdes.2016.06.021.

[5] KHVOSTOV, G., "Models for numerical simulation of burst FGR in fuel rods under the conditions of RIA", *Nucl. Eng. Des.* 328 (2018) 36–57, https://doi.org/10.1016/j.nucengdes.2017.12.028.

[6] KHVOSTOV, G., Analytical criteria for fuel fragmentation and burst FGR during a LOCA, *Nucl. Eng. Technol.* 52 10 (2020) 2402–2409, https://doi.org/10.1016/j.net.2020.03.009.

[7] KHVOSTOV, G., Modelling effects of transient FGR in LWR fuel rods during a LOCA, *J. Nucl. Mater.* 559 (2022), 153446. https://doi.org/10.1016/j.jnucmat.2021.153446.

[8] KHVOSTOV, G. AND A. GORZEL, FALCON code-based analysis of PWR fuel rod behaviour during RIA transients versus new U.S.NRC and current Swiss failure limits, *Nucl. Eng. Technol.* 53 11 (2021), 3741–3758, https://doi.org/10.1016/j.net.2021.06.001.

**IRSN R&D ACTIVITIES ON HIGH BURNUP FUEL BEHAVIOUR IN RIA AND LOCA CONDITIONS**

V. GEORGENTHUM, T. TAURINES, A. MOAL, P. RUYER, T. GLANTZ
Institut de Radioprotection et Sûreté Nucléaire
France

## 1. FUEL ROD EVOLUTION DURING BASE IRRADIATION

During base irradiation in nuclear reactor, fuel rods undergo significant evolutions.

Under a high temperature water environment, oxidation of the cladding material leads to a zirconia layer formation outside the clad. As a result of this oxidation, hydrogen is generated and diffuses into the base metal before precipitating as zirconium hydrides beyond the solubility limit. This results in observed hydride distribution and orientation. Under thermal gradients during plant operation, soluble hydrogen migration is observed towards the clad outer diameter leading to hydrides accumulation to form a hydride rim. Under certain conditions (for example if a cold spot appears on the clad), hydrogen can move towards a cold spot resulting in a massive zirconium hydride (hydride blister), embrittling the metal.

The fuel is characterized by a high fission gas retention resulting from a low gas release under normal operating conditions. Created mainly inside the fuel grain, a significant part of these gases moves towards the grain boundaries (GB) with thermal diffusion to the initial fabrication pores or the inter-granular bubbles created during irradiation. The GB gas fraction increases with the formation of high burnup regions in the outer part of $UO_2$ fuel or in the $(U,Pu)O_2$ clusters of the MOX fuel. In addition, a typical HBS is observed, characterized by the subdivision of the original grains and the development of a high porosity up to ~20%, that contain a major portion of the created gases. In $UO_2$ fuels, this microstructure change appears in the outer cold zones of the clad for average pellet burnups above ~40 GWd/tU. These changes are well established on a peripheral layer (50-100 µm wide) for an average pellet burnup varying between 60 and 65 GWd/tU. Regarding MOX fuels, the structure changes at high burnup are observed for a lower mean pellet burnup (~30 GWd/tU). In these cases, the measured local burnup for the $(U,Pu)O_2$ agglomerates varies between 100 to 160 GWd/tU, as a function of the initial enrichment.

Although a large dispersion is observed in the experimental results, the GB gas concentration under similar burnup levels, appears to be much higher in MOX fuel than in $UO_2$ fuel. This relative difference is mainly observed for low and medium burnups, and decreases for high burnups. As a matter of fact, at higher burnups, the influence of clusters decreases for MOX fuels, while the HBS formation increases for $UO_2$ fuels.

HBU fuels are also characterized by a quasi-closed gap, with a more-or-less pronounced bonding between fuel and internal zirconia layer.

These evolutions and their impact on the fuel behaviour during a RIA or a LOCA are addressed by IRSN with “academic studies”, separate effect tests and integrals test in reactor.

## 2. HIGH BURNUP FUEL BEHAVIOUR DURING RIA TRANSIENTS

The tests performed in the NSSR and CABRI REP-Na programme [1], performed in the 1990’s, showed the deleterious influence of a high clad corrosion level with hydride concentration (rim or blisters) on clad failure and the contribution of GB gases on FGR and potential gas loading, especially in MOX fuel, during the early phase of a fast power transient with limited clad heat-up. The remaining questions related to HBU fuel during RIA transient mainly concern the fission gas behaviour during the entire transient and its impact on clad loading, but also the rod behaviour under high clad temperature and internal pressure, and the related post-failure phenomena (e.g. fuel ejection, fuel–coolant interaction with finely fragmented solid fuel).

### 2.1. CABRI International Project (CIP)

The CIP started in March 2000 to study the behaviour of nuclear fuel during RIA under PWR conditions [2]. The project is funded by organizations from twelve countries and is chaired by IRSN and consists of tests in the CABRI reactor operated by CEA. CIP investigates the behaviour of both uranium dioxide ($UO_2$) and mixed-oxide (MOX) PWR fuels during RIA transients. The project aims at extending the database for high burnup fuels and advanced fuels and claddings. Changes were implemented to the reactor coolant loop after performing the first two tests to achieve conditions representative of PWRs. The objectives included the investigation of departure from nucleate boiling (DNB) and post-DNB phenomena, but also fuel ejection in water and fuel-water interaction.

### 2.2. Fission gas behaviour - Estimation of a GB failure limit

There is substantial experimental evidence from RIA tests that during fast transients the GB gas can be easily available for a rapid release; it is related to a quasi-instantaneous mechanical opening of GBs allowing a significant gas release in a very short time, including at low temperature.

To obtain mechanical and fracture parameters of grain boundaries, IRSN builds a methodology based on calculations at two scales (atomistic/mesoscopic) [3]. The approach consists of feeding a mesoscopic model with data from simulations at the atomic scale. The atomistic simulations are of the molecular dynamics type and use a variable charges semi-empirical potential to describe the interactions between atoms. These calculations, carried out using the LAMMPS software, are performed on three GB nanoscale structures representative of $UO_2$ at different temperatures. The atomistic simulations allow to obtain elastic properties and local fracture parameters (maximum stress and fracture energy) characteristic of the grain boundaries studied.

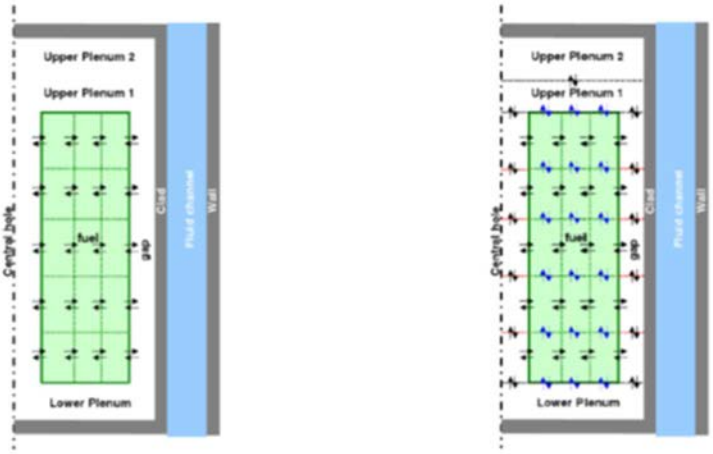
The results from atomistic calculations are then used as input data in cohesive zone models to perform simulations at the scale of the uranium dioxide grains which is that of the micron. At this scale, a cohesive-volumetric approach using the concepts of a Frictional Cohesive Zone Model in a multibody systems framework based on the Non-Smooth Contact Dynamics is employed. The impact of the presence of intergranular bubbles of different sizes is studied. A plastic model is used in the volume to consider possible dislocation movement in the system. The associated calculation code, called XPER, allows to analyse cracking induced by grain boundaries.

### 2.3. Modelling of gas flow in the fuel rod with SCANAIR code

In the SCANAIR code [4], gas flow in the fuel rod can be modelled using two different models (see the two schemes of gas flow modelling in Figure 1 below). The first model assumes that the equilibrium of pressure is instantaneous in the free volumes. Consequently, the pressure remains uniform during the transient.

The second model named “2D multi-species gas flow model” assumes that the equilibrium of pressure is not instantaneous and allows possible local over-pressurisation. The modelling of such phenomenon is of prime importance for the numerical simulation of post-DNB phenomena which can generate large clad deformation due to the conjunction of high clad temperature, increase of material ductility and local overpressure in the pellet-clad gap. Local pressures are computed in each modelled free volumes (open pores, cracks, dishes and chamfers in the pellet stack, pellet-clad gap, central hole, upper and lower plena). The flow of each gaseous species (fission gases, helium, air and argon) can be considered. The resolution of the mass balance equation takes into account a possible reverse flow from the free volumes towards the open pores. Although axial flow through the pores network is generally neglected in RIA conditions, it can be considered by the model. In the current model version, the flow through the free volumes is assimilated to a flow in a porous media following the Darcy's law. The velocity depends on the pressure gradient, the permeability coefficient and the cross section. The permeability coefficient is assumed to be the same for all the species. This implies that the gaseous species flow with the same velocity.

The validation of such a model is difficult in RIA representative conditions because of a lack of knowledge considering the flow path evolution induced by the change of geometry during the accidental transient. The ongoing Fission Gas Dynamics experimental programme dedicated to the study of the dynamic behaviour of fission gases present in highly irradiated fuels will certainly provide valuable information about the kinetic of FGR and flow. This programme performed in the NSRR reactor is jointly defined by JAEA and IRSN.




(a) (b)

*FIG. 1: Schemes of gas flow modelling with SCANAIR considering an instantaneous equilibrium of pressure in the free volumes (a) and with the "2D multi-species gas flow model" (b) ([4], courtesy of IRSN).*

The validation of the "2D multi-species gas flow model" is also underway thanks to numerical simulations of various out-of-pile gas transport experiments [5–7] carried out on irradiated fuel rods.

The computer architecture of the model and certain physical modelling options are still being developed at IRSN in order to make as general as possible its use for the simulation of various accidental transients by other codes of the FUEL+ software platform dedicated to the numerical simulation of fuel rod behaviour.

### 2.4. Fuel coolant interaction during RIA transient

In some hypothetical conditions, reactivity-initiated accidents could lead to rod failure and, subsequently to the clad opening, and then to the ejection toward the coolant flow of a mixture of fission gases and of fuel particles. Such an ejection implies hot fuel particles coming into contact with relatively cold liquid coolant, initiating a heat and mass transfer process. The associated vaporization rate under such large thermal disequilibrium condition is very fast: the timescale for the expansion rate of the produced steam is governed by a momentum balance at the liquid–steam interface. Therefore, if a sufficiently large vaporization rate has to be considered, pressure relief is not instantaneous, leading to the generation of a sharp pressure increase followed by a rapid expansion. Those phenomena generate mechanical energy as well as transient mechanical forces related to sharp pressure gradients through the fluid. The impact of the corresponding pressure wave on surrounding structures is then a matter of study for safety since the integrity of those structures has to be ensured. This type of fuel–coolant interaction and associated pressure peak and mechanical energy formation have been observed in several past in-pile experiments considering typical RIA pulses effect on fuel rodlets. Nevertheless, the measurements obtained from the instrumentation of those experiments have deficiencies in term of representativity with respect to the fluid pressure and temperature conditions, the latter affecting the phenomena. The violence of the interaction and of the resulting mechanical effects also depend on the fuel properties. The most violent ever observed RIA related interactions are related to the waterlogging of fuel rods. The specificity of HBU fuel with respect to this fuel coolant interaction has not been specifically studied. Since burnup affects the fuel fragmentation, following fuel failure, smaller fuel fragments could be ejected through the clad opening, and therefore a larger mass. Moreover, their relatively smaller size increases the surface area to volume ratio of the fuel in contact with the coolant and therefore enhances the heat and mass transfer process. Both larger mass and size will therefore lead to a more violent fuel–coolant interaction with respect to fresh fuel, as outlined in Ref. [8]. Without any experimental data in those specific conditions, estimation of this violence could be only inferred from a model that has to take into account the basic fuel ejection dynamics (that scale the typical amount of fuel in contact with the coolant) and of thermal interaction (that scales the pressure surge intensity with respect to a given temperature difference between solid and coolant). IRSN has launched an R&D programme on those topics. First, dedicated experimental programme have focused on the flow of dense granular flow, modelling the fragmented fuel within the clad, from a confined cylinder through an aperture (the clad opening), [9]. The coupling of such flow with the ejection of pressurized gases (modelling the fission gases initially within the rod) has been taken into account [10]. The study of the thermal interaction has been performed in the study of the consequences of a very rapid heating power pulse within a confined pressurized and subcooled liquid tank at various pressure and temperature conditions, reaching similar conditions of water in PWR with $CO_2$ as a simulant fluid [11]. It has then been clearly shown how the pressure surge was associated with the very intense vaporization process at the time of the power pulse, and the scaling of the pressure surge with absolute pressure revealed a less intense interaction for larger pressures [12]. All these experimental data are then used to feed the modelling of the IRSN CIGALON code, developed for the interpretation of in-pile RIA experiments leading to fuel-coolant interaction, as those of the CABRI International Project [2].

## 3. HIGH BURNUP FUEL KEY PHENOMENA TO ADDRESS DURING A LOCA TRANSIENT AT THE ROD FUEL SCALE

With regards to the study of fuel behaviour under LOCA, the extensive state of the art review performed by IRSN led to identification of questions related to fuel fragmentation, relocation and dispersal after cladding burst. In this frame, specific modelling is under development in the

DRACCAR code to simulate the 3D thermomechanical behaviour and reflooding of a fuel rod assembly during a LOCA transient taking into account fuel fragmentation, relocation and flow blockage.

### 3.1. Fuel Fragmentation, relocation and dispersal

FFRD was studied by various in-pile and out-of-pile experimental programmes and showed extensive relocation, fragmentation and dispersal for very high burnup fuels [13]. The DRACCAR code is developed at IRSN to model complex configurations such as the deformation and possible contact between neighbouring rods leading to a blockage of thermal-hydraulic channels of the fuel assembly affected by the ballooned zone. Models for axial relocation, fragments size impact on thermal conductivity and fuel dispersal were developed and used to simulate various LOCA experiments on high burn up fuel, such as Halden LOCA tests.

Sensitivity studies with the DRACCAR code showed that the most influential parameter on peak cladding temperature is the filling ratio in the ballooned region which has a strong impact on the power profile. Therefore, a new model was developed in the DRACCAR code to allow a variable filling ratio with cladding strain. Indeed, recent relocation data from literature and obtained in the OECD SCIP-III programme suggest that there is a strong link between the filling ratio, cladding strain and fuel fragments size distribution. The data from Ref. [14] obtained by simulation using discrete elements methods were analysed in terms of filling ratio versus cladding strain and are plotted in Fig. 2. In this numerical work, fragmented pellets with large polyhedral fragments (1–3 mm) and a varying proportion of small spheres (400 nm), were axially relocated by gravity. For small particles proportions (lower than ~30%), the filling ratio tends to decrease with the local circumferential strain. Some very valuable data from the OECD SCIP-III programme were also analysed to propose two correlations for maximum filling ratio versus circumferential strain.

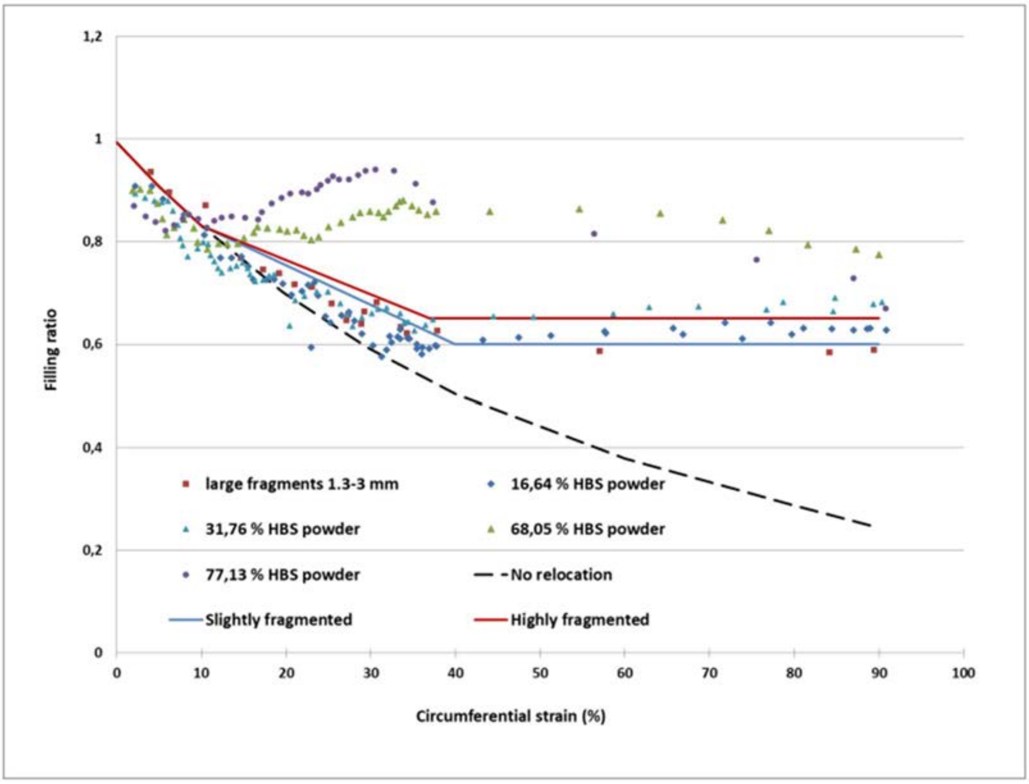



*FIG. 2: Filling ratio versus cladding strain ([15], courtesy of IRSN).*

Experimental programmes on high burn up fuels are expensive and material intensive, therefore a new manufacturing process was developed to build pre-fragmented fuel pellets surrogates with various mechanical properties [16]. Laser stereolithography is used to print fuel surrogates in the shape of pre-fragmented alumina pellets. The internal 3D architecture of these alumina pellets can be adjusted on demand. The pellets are typically made up of 50 fragments spaced apart by an average space of 200 µm and interconnected by solid bridges of 400 µm in diameter as illustrated in Fig. 3. The fracture strength of these pellets was tailored to simulate nuclear fuel by investigating debinding and sintering thermal treatments. That fracture strength was tested by lateral compression with a piece of cladding or by Brazilian tests. These pellets will be used to study axial relocation inside a cladding with dedicated experiments and discrete elements methods modelling.

*FIG. 3: Pre-fragmented alumina pellet printed by stereo-lithography.*

### 3.2. Core coolability of ballooned region with power relocation

Validation of the DRACCAR code on the coolability of assemblies including ballooned rods with power relocation is performed using the recent COAL experiments carried out in the framework of the PERFROI project [17]. The COAL bundle consists of rods (electrically heated), some with a pre-deformed zone (Fig. 4a) with local overheating representing fuel relocation. A bundle is made of 49 (7 x 7) rods, including 16 deformed rods in the centre, 30 non-deformed rods and 3 guide tubes (Fig. 4b). The test section also includes 6 spacer grids (4 mixing vanes grids plus 2 holding grids at the bottom and at the top of the bundle) each 500 mm apart. The test section is surrounded by a heated square housing (Fig. 4c) which is included in a pressure vessel (Fig. 4d) with two lower and upper plenums for the electrical and instrumentation connections and the fluid supply. The test device is inserted in the STERN-Laboratory loop (Hamilton, Canada) to carry out the reflooding scenario.

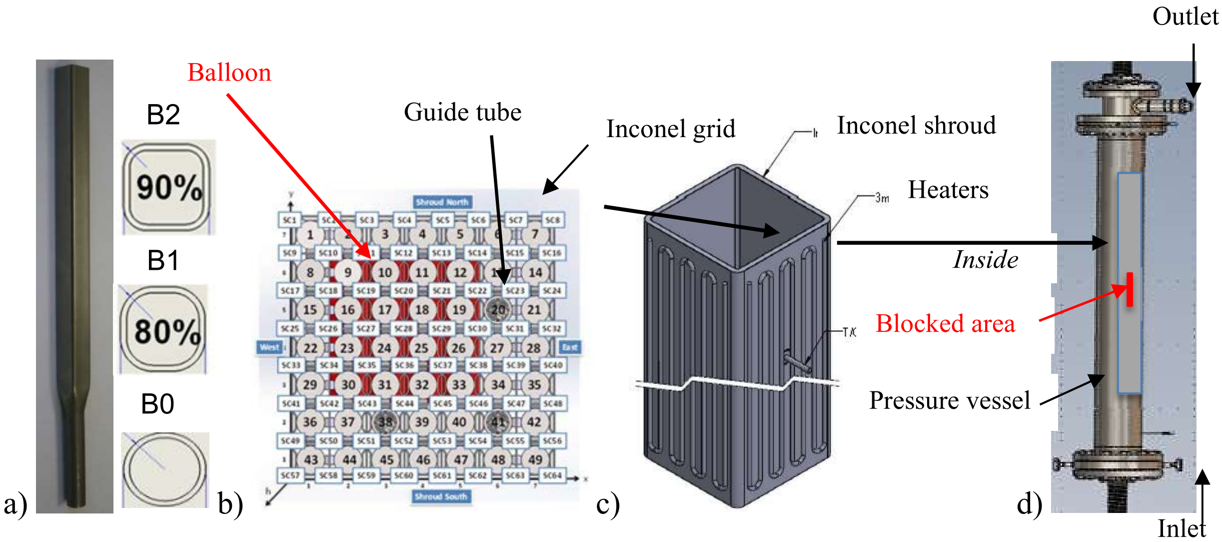



*FIG. 4: Overview of the COAL device ([18], courtesy of IRSN).*

The COAL bundle is fitted with many thermocouples (up to 400) located in the rods, guides tubes, fluid channels and housing external faces to cover all the spatial volume of the bundle. Additional measurements give access to the inlet and outlet flowrate (steam and liquid phase) and to evolution of mass inventory during the transient (through differential pressure transducers).

The scenario is divided in two phases as for classical "Reflooding experiments" performed in the 1980's:

- Heat-up phase in a dry steam atmosphere (the depressurization is not simulated);
- Reflooding by water injection with different thermal hydraulic conditions.

During a COAL reflooding campaign, almost 25 tests are carried out to cover a wide range of main thermal hydraulics parameters expected during intermediate (high pressure) and large break (low pressure) LOCA:

- Coolant pressure: 0.2–3 MPa;
- Inlet water flow velocity: 1.5 –8 cm/s (15 to 80 kg/s/m$^2$);
- Sub-cooling water temperature: -60°C to 0°C (saturation conditions);
- Power per rod: 1.5–3.3 kW.

Three experimental campaigns have already been achieved. In the first campaign, the bundle does not include ballooned rod: it is the reference bundle B0 with intact rods. In the two other campaigns, the geometry of the ballooned zone changes (B1 with a blockage ratio of 80% and a balloon length of 100 mm, and B2 with a blockage ratio of 90%, and a balloon length of 300 mm) in order to study the impact of the blocked area on the reflooding behaviour. An additional campaign is planned in which the impact of the balloons coplanarity will be studied (the bundle design is in progress).

## 4. SUMMARY

Key phenomena to study for modelling HBU fuel behaviour during RIA and LOCA include:

- FGR in transient conditions;
- Axial and radial gas flow in the rod;
- Fuel coolant interaction after clad failure (RIA);

- Fuel fragmentation, relocation and dispersal (LOCA);
- Core coolability of ballooned region with power relocation (LOCA).

The FUEL+ software platform was created in 2020 with the aim to model the behaviour of nuclear fuel during normal, incidental and accidental conditions, as well as during storage in pools and transport. FUEL+ includes several long-standing and newer software:

- FIRST: Fuel Irradiation in Reactor, Storage and Transport (coupled with SCIANTIX for the calculation of gas distribution in the fuel);
- SCANAIR: System of codes for normal operating conditions and RIA analysis;
- CIGALON: Fuel-coolant interaction during RIA;
- DRACCAR: Deformation and reflooding of a fuel rod assembly during LOCA;
- SHOWBIZ: Study of Hydrogen and Oxygen Weakening Behaviour In Zirconium.

## REFERENCES


[1] PAPIN, J., et al., Summary and interpretation of the CABRI REP-Na Program, *Nucl. Techn.* 157 (2007).

[2] BIARD, B., et al., Reactivity Initiated Accident transient testing on irradiated fuel rods in PWR conditions: The CABRI International Program, *Ann. Nucl. Energy* 141 (2020) 107253.

[3] MANOROSOA, Z. A., et al., “Multiscale - atomistic/mesoscopic approach for the determination of fracture criterion in uranium dioxyde - $UO_2$”, paper presented at the Seventh International Conference on Computational Modeling of Fracture and Failure of Materials and Structures, Prague, Czech Republic, 2023.

[4] MOAL, A., GEORGENTHUM, V., MARCHAND, O., SCANAIR: A transient fuel performance code Part One: General modelling description, *Nucl. Eng. Des.* 280 (2014).

[5] RONDINELLA, V.V,. et al., “Measurement of gas permeability along the axis of a spent fuel rod”, JRC94524, Proc. Int. Conf. TopFuel2015, Zurich, 2015.

[6] MAGNUSSON, P., Integral LOCA test to study fuel fragmentation of the high burnup fuel rod 3V5-Q13, Studsvik Report, STUDSVIK/N-17/068, STUDSVIK-SCIP III-209, February 2017.

[7] DAGBJARTSSON, S.J., et al., Axial Gas Flow in Irradiated PWR Fuel Rods, internal; report, INEL, TREE-NUREG-1158, September 1977.

[8] WACHS, D.M., “Performance of waterlogged LWR fuel during RIA”, Proc. Int. Conf. Top Fuel2016, Boise, ID, USA, September 11-15 (2016).

[9] ZOU, Z., et al., Discharge of a silo through a lateral orifice: Role of the bottom inclination versus friction, *Phys. Rev. E* 102 (2020) 052902.

[10] ZOU, Z. et al., Nonsteady discharge of granular media from a silo driven by a pressurized gas, *Phys. Rev. Fluids* 7 (2022) 064306.

[11] MULLER, J., et al., First characterization of two-phase phenomena occurring during a rapid energy discharge in saturated carbon dioxide, *Exp. Therm. Fluid Sci.* 129 (2021) 110471.

[12] MULLER, J., et al., Pressure transient and vaporization process following the rapid heating of a liquid, *Exp. Therm. Fluid Sci.* 137 (2022) 110674.

[13] OECD NUCLEAR ENERGY AGENCY, Report on Fuel Fragmentation, Relocation and Dispersal, OECD 2016, No. NEA/CSNI/R(2016)16 (2016).

[14] MA, Z., et al., A three-dimensional axial fuel relocation framework with discrete element method to support burnup extension, *J. Nucl. Mater.* 541 (2020).

[15] TAURINES, T., et al., “New burst criteria, axial fuel relocation and fission gas release models in the DRACCAR code”, Proc. Int Conf. TopFuel2021, Santander (Spain), 24–28 October (2021).

[16] BELQAT, M., et al., “Development of a manufacturing process for a simulating fuel for the study of fine fragmentation in LOCA”, paper presented at 2022 Nuclear Materials Conference (NuMAT 2022), Ghent, Belgium, Oct 2022.

[17] REPETTO, G., et al., “COAL experiments investigating the reflooding of a 7 X 7 rods bundle during a LOCA - Thermalhydraulics results”, Proc. 19$^{th}$ Int. Topical Meeting on Nuclear Reactor ThermalHydraulics (NURETH 19), Brussels, Belgium, March 6-11 (2022).

[18] LEJOSNE, A., et al, “State of the art of IRSN’s studies regarding fuel behavior during LOCA, Proc. Int. Conf. on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs”, Vienna, Austria, 18–21 October (2022).

# FUEL FRAGMENTATION, DISPERSAL AND CONSEQUENCE EVALUATION

K. YUEH
Electrical Power Research Institute
United States of America

Nuclear fuel pellets are known to be susceptible to fragmentation when subjected to temperature transients where the temperature profile within the fuel pellet departs from the previous thermal equilibrium. Fresh or low burnup fuel pellets typically fragment into few large pieces due to thermal stress on initial powering up or under transient conditions. Recent tests of HBU fuel showed fine fragmentation, beyond thermal fracturing, is possible without exposure to extreme temperatures, e.g. below 1273°C [1]. The fine fuel fragmentation is typically not an issue for current fuel designs since most plants operate HBU fuel at power sufficiently low to preclude fuel cladding burst. However, the industry's desire for higher burnup and greater fuel utilization requires operation of HBU fuel at high power and thus fuel rod bursts and release of fuel fragment is possible. In order to assess potential consequences of fuel fragments release from burst of HBU fuel rods, it is necessary to determine the number of burst rods, the extent of fuel release from each rod and, under high steam flow conditions, how the fuel disperses in the fuel assembly, the RCS and containment.

An 18-month HBU core design with peak rod average burnup of ~75 GWd/tU was used to evaluate HBU fuel rods burst potential. The evaluation showed very high rod internal pressure at EOL, significantly exceeding the system pressure. This significantly changes how the cladding balloons. Lower burnup core designs that operate intermediate HBU fuel at lower power release less fission gas during operation. During the initial temperature ramp from stored energy in a LOCA, the rod pressure is typically low enough that the cladding does not creep out prior to RCS depressurization. The pressure inside a HBU rod is sufficiently high to overcome the pre-depressurization system pressure and plastically deforms the cladding. This sets up the condition that potentially determines how the cladding balloons later in the transient since the portion that experiences plastic deformation would creep preferentially. Burst assessment using criteria/data documented in references [2–3] suggests HBU rods operating above the core average power may burst and thus, depending on the core design, significant mass could be released.

Experimental data indicates the process of fuel fragmentation may be retarded by cladding restraint and fuel in the region may become immobile when the outward cladding deformation is small [4]. Model calculations show the lower power underneath a grid and enhanced heat transfer at the grid location from steam cooling may result in lower cladding temperatures beneath grids. The lower temperature could result in cladding strain below the fuel mobility strain threshold established in Ref. [4]. Such a condition could separate parts of a fuel rod from the burst node and thus reduce the amount of fuel that could be released from a burst fuel rod. Experimental demonstration of this effect is being considered but testing condition is likely very complicated since the cooling condition changes rapidly in a LOCA.

An EPRI developed thermohydraulic code, GOTHIC, was used to evaluate fuel fragment dispersal within the RCS and containment. The code has the capability to track individual fuel fragments once released from a burst fuel rod. The evaluation used fuel fragments size distribution based on mass assay reported in Ref. [3] and the simulation was biased to release more finer fragments at the start of a rod burst. The evaluation utilized a mathematical blockage model and showed significant mass could be blocked by the spacer grids. Fuel fragments that escape the fuel assembly may deposit in other parts of the RCS, but mostly end up in the containment. As may be expected, with grid blockage more fuel fragments are transported to the containment when starting fragments, e.g. fragments leaving a burst rod, are smaller. Without grid blockage, most of the fuel fragments are carried by steam to the containment. Simulations are in progress to evaluate fuel fragments dispersal inside a containment. Initial runs without fuel fragment loading suggests timing of rod burst may have a significant impact on the dispersal pattern inside the containment. Steam velocities in the containment are much higher in the first few seconds after the break and this could result in a wider dispersal pattern compared to rod bursts later in the transient when the steam velocity decreases.

Some of the models used in the GOTHIC simulation, such as the blockage algorithm at grid locations and particle terminal velocity are not based on actual fuel fragments geometries. Experimental verification of these models is in progress. The terminal velocity of some of the larger fuel fragment sizes will be measured in air and the equivalent hydraulic diameter will be calculated for different fragment weights. The water flow velocities needed to carry fuel fragments of different size on the containment floor or major component surface will also be measured to evaluate potential accumulation of fuel fragments at the sump.

To evaluate possible blockage at the grid, a flow experiment with a 3x3 fuel bundle with two spacer grids will be conducted. The experiment will use crushed non-irradiated $UO_2$ fuel fragments that are sorted to into size groups (bins) described in Ref. [3]. Model calculations show fuel fragments carried by steam could impact a grid at significant velocities and recent drop tests of post-LOCA fuel fragments showed sub-fragmentation is likely. The drop test results showed approximate 50% of the sub-fragment mass after impact with a spacer grid is contained in the next smaller bin size. Therefore, mass contained in the larger bin sizes will be down converted per experiment results. Fuel fragments will be ejected from a perforated centre rod by air pressure. Air will be used to simulate steam flow in the experiment and flow velocity will be varied per LOCA calculation. The amount of fuel relocation to below the burst node, trapped in the fuel and carried outside of the fuel assembly will be measured. Fuel remaining in the fuel assembly is of primary interest for coolability evaluation. Additional evaluations will be performed with the trapped fuel composition to determine coolant permeability for subsequent coolability evaluations.

## REFERENCES


[1] KOLSTAD, E., et al, "High burnup-up fuel behavior under LOCA conditions as observed in Halden experiments", paper presented at IAEA Technical Meeting on Fuel Behavior and Modelling under Sever Transient and LOCA Conditions, MITO, Japan, October 18-21, 2011.

[2] NUCLEAR REGULATORY COMMISSION, Cladding Swelling and Rupture Models for LOCA Analysis, NUREG-630 (1980).

[3] NUCLEAR REGULATORY COMMISSION, Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-coolant Accident, NUREG-2121 (2012).

[4] BALES, M., CHUNG, A., CORSON, J., KYRIAZIDIS, L., Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup, RIL 2021-13, U.S. Nuclear Regulatory Commission, Washington, DC (2021), ADAMS Accession Number ML21313A145.

# ASSESSING THE IMPACT OF PROTOTYPIC HIGH BURNUP OPERATING CONDITIONS ON FUEL FRAGMENTATION, RELOCATION AND DISPERSAL

N. CAPPS, I. GREENQUIST, A. WYSOCKI, J. HIRSCHHORN*

Oak Ridge National Laboratory
United States of America

* Current affiliation: Idaho National Laboratory, United States of America

Investigating strategies are being elaborated by the US nuclear energy industry to further reduce the cost of energy production for its existing fleet. NPP operating costs are governed by some components typically beyond the operating organization's control. It is not the case for core design optimizations which offer potential operational savings. Nevertheless, some limits constrain those optimizations, including the regulatory limit for fuel enrichment up to 5% $^{235}U$ and safety concerns limiting the rod average burnup to 62 GWd/tU. Renewed efforts have been led by the industry to extend the burnup limit beyond 62 GWd/tU. These efforts will likely require additional safety analyses compared to those currently accepted by the regulatory body. The purpose of this work is to demonstrate a BEPU pin-by-pin high burnup LOCA analysis technique to assess full-core high burnup FFRD and to identify approaches for minimizing or potentially mitigating FFRD through core design optimizations.

The first nuclear power stations licensed in the United States were approved to operate on fuel-to-rod average burnups of approximately 30 GWd/tU. This decision was based on a variety of experimental fuel performance and safety data sources that demonstrated safe operation. Extensive demonstration and testing under normal operating conditions were conducted in power generating stations to characterize relevant performance attributes in support of this limit. Simultaneous integral testing under power–cooling mismatch conditions representative of LOCA and RIA scenarios were also conducted at transient test reactors to identify the dominant fuel failure modes and develop relevant fuel safety criteria. The results were essential for developing empirical fuel performance limits and validating the computer codes to demonstrate design compliance with the experimentally determined safety limits required under commercial operation. The knowledge acquired during these tests is reflected in the fundamental structure of the U.S. NRC regulatory documents, including Ref. [1].

Subsequent work was conducted over the following decades to extend the allowable utilization envelope for $UO_2$ and Zr alloy-based fuels to the current limits (i.e. 62 GWd/tU peak rod average). Burnup extension was made possible by fuel technology improvements, most notably the movement of cladding designs from Zircaloy-4 to Zircaloy concepts containing Nb. Additionally, fuel material responses under postulated severe accident conditions are better understood.

Historically, mechanical testing of cladding alloys (which have been previously irradiated under normal service conditions followed by semi-integral LOCA tests to determine the embrittlement thresholds) determined their design basis limits. Claddings were generally found to embrittle if the peak temperature exceeded ~1,204°C or if the cladding that reacted with steam exceeded 17% of the original cladding thickness [2]. This effort was further expanded when a limited number of integral tests were performed to identify and evaluate the dominant failure modes and remained unchanged as burnup levels increased to 62 GWd/tU. Thus, it was determined that semi-integral or separate effects test data were applicable to further extend the relevant fuel safety limits without changing the fundamental safety criteria or analysis techniques.

The Norwegian Halden reactor (followed by Studsvik semi-integral tests) provided a series of integral tests showing a new response of high burnup fuel pellets during LOCA conditions compared to that seen in previous observations [3]. These tests revealed in particular a limit around 67 GWd/tU below which the fragments observed were large compared to the size of the rupture, with modest relocation from these large fragments into the balloon region. However, for fuel samples irradiated above >67 GWd/tU, the fuel pellets fragments were pulverized into small pieces [3]. An important fraction of those fragments was fine enough so they relocated axially within the balloon region, and then ejected. In those tests (Halden and Studsvik tests), those particles were ejected through the cladding rupture into the coolant. The effect of FFRD invalidated the historical assumptions, specifically those related to core coolability and criticality.

In response to these results, U.S. NRC commissioners wrote a letter to their staff identifying a need to take these phenomena into account to assess the acceptability of future fuel designs [4]. This work builds on previously developed methodologies [3, 5] to assess core-wide FFRD susceptibility, based on fuel performance parameters under steady state conditions, possibly contributing to FFRD susceptibility. Those parameters evaluation aims to support future high burnup LOCA tests and to inform strategies intended to reduce or potentially mitigate FFRD. Moreover, the TRAC/RELAP Advanced Computational Engine (TRACE) will use a subset of data representing the high burnup operation envelope to evaluate thermohydraulic transients. Those data will be used subsequently for transient fuel performance analysis using the BISON code. The results obtained will be used to inform future LOCA tests programmes helping to determine cladding rupture susceptibility and the parameters that increase it, but also FFRD susceptibility in case of rupture prediction. The final goal of this work is then to illustrate via a BEPU methodology the core-wide FFRD susceptibility assessment and also to inform high burnup fuel design approaches in order to minimize or potentially mitigate the susceptibility of FFRD phenomena.

This analysis is based on a Westinghouse four-loop PWR plant using 193 nuclear fuel assemblies in the core for a rated power of 3,626 MWth, with an ambient pressure containment design. The current fuel management strategy allows an 18-month fuel cycle length. However, Southern Nuclear Company and ORNL model a transition scheme to fuel cycle lengths up to 24 months [5]. The fuel management includes a design using fresh fuel, once-burned and twice-burned fuels. The twice-burned assemblies are positioned in such places where the power ratings and temperatures are the lowest, i.e. on the core's periphery. The methodology developed for both steady state operation and for LBLOCA transients allows the characterization of high burnup fuel rods. As a first step, the VERA environment (Virtual Environment for Reactor Applications) was used to model the transition from an 18-month to a 24-month fuel cycle, followed by steady state 24-month fuel cycles. For every rod in the core, temperature and power histories were calculated by VERA. Then, a representative subset of rods was selected statistically and modelled using the fuel performance code BISON (using finite element method). The temperature predictions were then used as initial conditions for the TRACE code to simulate a LBLOCA transient. Those predictions were then used for a second set of BISON simulations to predict fuel performance during the transient. These BISON results were statistically analysed to predict the susceptibility to FFRD of the full core during a LBLOCA event.

A series of multiphysics simulations was conducted during a postulated LBLOCA event, to predict the FFRD susceptibility for high burnup fuel of an LWR. The number of rods was reduced over the course of these simulations, from 32,944 high burnup rods using the VERA environment to 242 rods in the transient using the BISON code.

The BISON transient simulations used two models for cladding failure: the strain rate criterion and the Chapman correlation. Those models were used to predict cladding burst occurrence during LBLOCA transients. Other models applied allowed to predict the fuel fragmentation amount surrounding the bursts regions and, therefore, the susceptibility for FFRD. The error margins for these predictions were also identified to estimate ranges for the FFRD susceptibility.

FFRD susceptibility estimation could reach 5,392 kg of fuel. It is important to note, however, that this corresponds to a worst-case scenario, built upon another worst-case scenario. Both models, used to predict cladding failure and pulverization, are meant to be conservative, and the total amount of fuel dispersal indicated before corresponds to the highest value predicted amongst six model combinations. Moreover, the LBLOCA transient corresponds also to a worst-case accidental scenario postulated when burnups were highest, at the end of the cycle. Such an accident is relatively unlikely.

## REFERENCES


[1] NUCLEAR REGULATORY COMMISSION, Domestic Licensing of Production and Utilization Facilities, 10 CFR 50, US Govt Printing Office, Washington, DC (2006).

[2] NUCLEAR REGULATORY COMMISSION, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactor, 10 CFR 50.46, US Govt Printing Office, Washington, DC (2006), https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0046.html.

[3] CAPPS, N., R. SWEET, B.D. WIRTH, A. NELSON, Development and demonstration of a methodology to evaluate high burnup fuel susceptibility to pulverization under a loss of coolant transient, *Nucl. Eng. Des.* 366 (2020) 110744.

[4] NUCLEAR REGULATORY COMMISSION, Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident, NUREG-2121 (2012), https://www.nrc.gov/docs/ML1209/ML12090A018.pdf.

[5] CAPPS, N., A. WYSOCKI, A. GODFREY, B. COLLINS, R. SWEET, BROWN, N., S. LEE, N. SZEWCZYK, S. HOXIE-KEY, Full Core LOCA Safety Analysis for a PWR Containing High Burnup Fuel, *Nucl. Eng. Des.* 379 (2021) 111194, https://doi.org/10.1016/j.nucengdes.2021.111194.

**STATUS AND PLANNED ACTIVITIES FOR THE SCIANTIX CODE APPLICATION TO HIGH BURNUP FUELS**

D. PIZZOCRI, T. BARANI*, A. MAGNI, G. ZULLO, L. LUZZI
Politecnico di Milano, Department of Energy, Nuclear Engineering Division
Italy

* Current affiliation: Commissariat à l'Energie Atomique et aux Energies Alternatives, France

The SCIANTIX code capabilities to describe fission gas behaviour in HBU fuels have been progressively extended since the release of the code [1]. The current version of SCIANTIX can be used both as stand-alone for the simulation of separate effect experiments and coupled with fuel performance codes as a fission gas behaviour module. The scale at which SCIANTIX operates is the scale of fuel grains, at which the assumption of uniform temperature, fission rate, and hydrostatic stress are reasonable. This working hypothesis requires caution when it is applied in the rim of pellets in LWR, in which the self-shielding of thermal neutrons generates high gradients of fission rate.

Recent developments of SCIANTIX involved (i) restructuring the code with an object-oriented programming logic, enabling the way for faster and easier further inclusion of physical models and software maintenance, (ii) completely separating the parts of the code devoted to defining material properties, implementing behavioural models, and solving the associated differential equations. This separation allows for once and for all verification of the numerical solvers and for the application of the same behavioural model (e.g. fission gas diffusion) in different fuel matrixes (e.g. non-restructured uranium dioxide and HBS).

Developments implemented to improve the code capabilities of HBU fuels are detailed in Ref. [2]. It includes:

- Addition to SCIANTIX of HBS fuel material, incorporating properties for restructured fuel.
- HBS formation combining an empirical threshold based on effective burnup [3–5] and KJMA model to describe the fraction of fuel undergoing restructure [6].
- An empirical model defines HBS porosity evolution, which is predicted as a function of local burnup [7].
- Porosity distribution evolution based on a physics-based model [8] derived from a Fökker-Planck approximation of pore-size evolution [9].
- The HBS pores size is derived semi-empirically (function of the porosity and the number density of pores), as for similar meso-scale codes [10–11].

Figure 1 shows a set of representative model results, which are compared to recent experimental data [12]. Figure 1c confirms the semi-empiric nature of the model, as the porosity increase is clearly proportional to the local effective burnup up to 15% (the SCIANTIX result has not been adjusted to the set of experimental results used for this comparison). Figure 1a shows an increase in pore number density from 50 to 100 GWd/tU (nucleation of pores is correlated to the formation of the HBS), followed by a decrease above 100 GWd/Ut (pore interconnection as the average pore radius increases).

The physics-based (considering gas atoms per pore) evolution with burnup of the HBS pore-size distribution is also included in the updated model (Fig. 1d). This physics-based feature can be extended in the future to include the number of vacancies per pore (which will allow to predict a pressure distribution, and then to model fuel fragmentation).

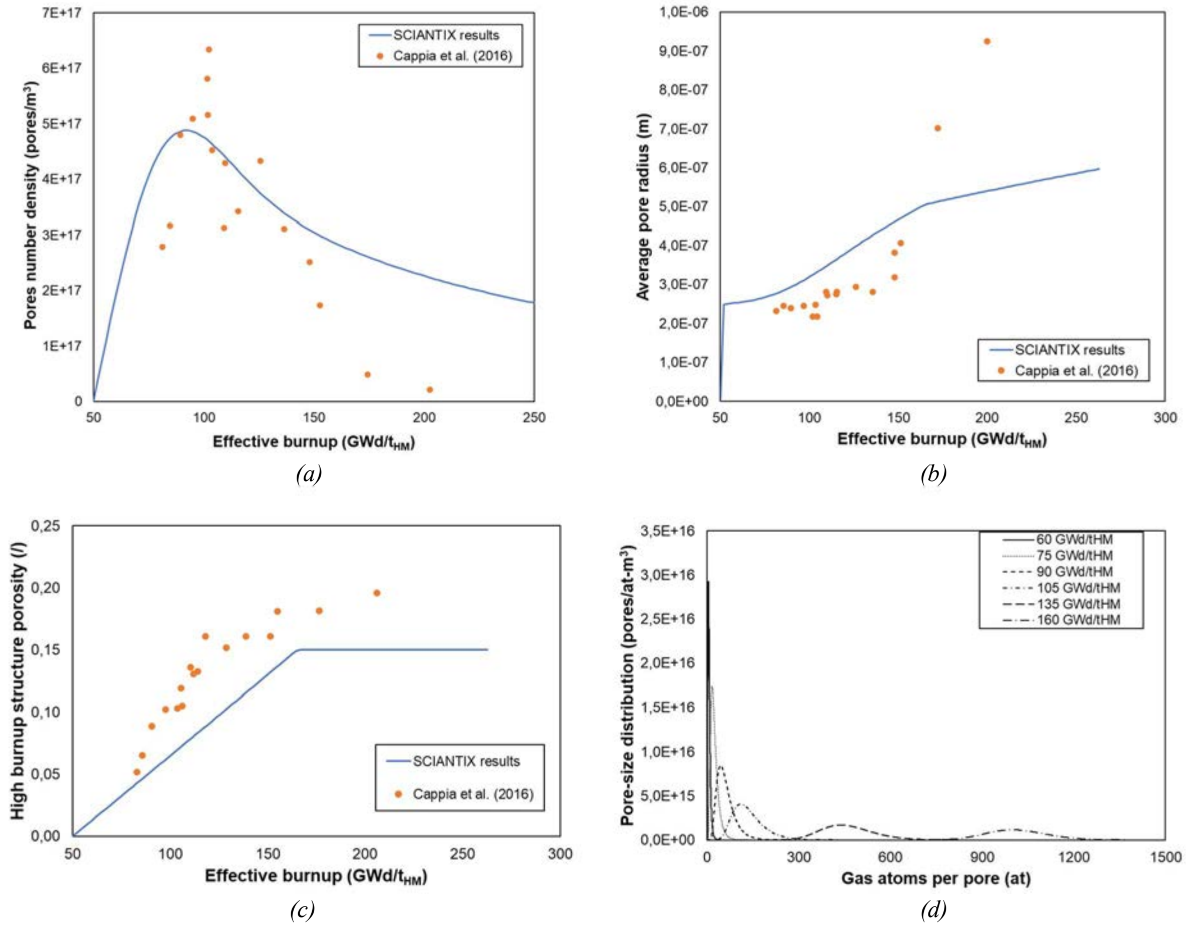



*FIG. 1. Representative results of the behavioural model describing HBS in SCIANTIX. The model includes, (a), a description of the pore number density based on the KJMA model of HBS formation rate and pore interconnection, (b) and (c), a semi-empirical description of the pore radius based on local HBS porosity proportional to the local effective burnup, and (d), a physics-based description of the evolution of the pore-size distribution.*
*Data from [11]. Figure courtesy of Davide Pizzocri, Politecnico di Milano.*

This updated version of SCIANTIX includes HBS formation and the fuel porosity evolution at HBU. These capabilities allow direct coupling within fuel performance codes (numerical robustness, verification and validation, computational time).

Further developments include the already mentioned extension towards a description of the pore–pressure distribution, which is key towards fuel fragmentation, and:

- Overcoming the empirical nature of the description of porosity, which is achieved with the physics-based description of the vacancy concentration in HBS pores.
- Overcoming the semi-empirical nature of the KJMA model of HBS formation via the modelling of the dislocation network evolution [13], paired with considerations of the effect grain size in postponing the formation itself (e.g. in Cr doped fuels [14]).

These models' developments are currently being targeted in synergy with international partners and in the framework of international research projects.

## REFERENCES


[1] PIZZOCRI, D., BARANI, T., and LUZZI, L., SCIANTIX: A new open-source multi-scale code for fission gas behaviour modelling designed for nuclear fuel performance codes, *J. Nucl. Mater.* 532 (2020) 152042. https://doi.org/10.1016/j.jnucmat.2020.152042.

[2] ZULLO G., PIZZOCRI D., LUZZI L., The SCIANTIX code for fission gas behaviour: Status, upgrades, separate-effect validation, and future developments, *J. Nucl. Mater.* 587 X (2023) 154744. https://doi.org/10.1016/j.jnucmat.2023.154744

[3] HOLT, L., SCHUBERT, A., VAN UFFELEN, P., WALKER, C.T., FRIDMAN, E. and SONODA, T., Sensitivity study on Xe depletion in the high burn-up structure of $UO_2$. *J. Nucl. Mater.* 452 (2014), 166–172, https://doi.org/10.1016/j.jnucmat.2014.05.009.

[4] KHVOSTOV, G., NOVIKOV, V., MEDVEDEV, A. and BOGATYR, S., "Approaches to Modeling of High Burn-up Structure and Analysis of its Effects on the Behaviour of Light Water Reactor Fuels in the START-3 Fuel Performance Code", paper presented at Water Reactor Fuel Performance Meeting (WRFPM), Kyoto, Japan, 2-6 October 2005.

[5] PIZZOCRI, D., CAPPIA, F., LUZZI, L., PASTORE, G., RONDINELLA, V.V., and VAN UFFELEN, P., A semi-empirical model for the formation and the depletion of the high burnup structure in $UO_2$ fuel, *J. Nucl. Mater.* 487 (2017) 23–29. https://doi.org/10.1016/j.jnucmat.2017.01.053.

[6] BARANI, T., PIZZOCRI, D., CAPPIA, F., LUZZI, L., PASTORE, G., and VAN UFFELEN, P., Modeling high burnup structure in oxide fuels for application to fuel performance codes, part I: High burnup structure formation, *J. Nucl. Mater.* 539 (2020), 152296. https://doi.org/10.1016/j.jnucmat.2020.

[7] LASSMANN, K., WALKER, C.T., VAN DER LAAR, J., LINDSTRÖM, F., Modelling the high burnup $UO_2$ structure in LWR fuel, *J. Nucl. Mater.* 226 (1995) 1, https://doi.org/10.1016/0022-3115(95)00116-6.

[8] BARANI, T., PIZZOCRI, D., CAPPIA, F., PASTORE, G., LUZZI, L., and VAN UFFELEN, P., Modeling high burnup structure in oxide fuels for application to fuel performance codes. Part II: Porosity evolution, *J. Nucl. Mater.* 563 (2022), 153627. https://doi.org/10.1016/j.jnucmat.2022.

[9] PIZZOCRI, D., PASTORE, G., BARANI, T., MAGNI, A., LUZZI, L., VAN UFFELEN, P., PITTS, S., ALFONSI, A., and HALES, J.D., A model describing intra-granular fission gas behaviour in oxide fuel, *J. Nucl. Mater.* 502 (2018) 323–330. https://doi.org/10.1016/j.jnucmat.2018.02.024.

[10] KREMER, F., DUBOURG, R., CAPPIA, F., RONDINELLA, V.V, SCHUBERT, A., VAN UFFELEN, P. and WISS, T., "High Burn up Structure formation and growth and fission product release modelling: new simulations in the mechanistic code MFPR-F". Proc. Int Conf. TopFuel2018, Prague, September 30–October 4 (2018).

[11] LASSMANN, K., SCHUBERT, A., VAN UFFELEN, P., GYORI, C., and VAN DE LAAR, J., TRANSURANUS Handbook, Copyright © 1975-2022, Joint Research Centre, Karlsruhe, Germany, 2022.

[12] CAPPIA, F., PIZZOCRI, D., SCHUBERT, A., VAN UFFELEN, P., PAPERINI, G., PELLOTTIERO, D., MACIAN-JUAN, R., and RONDINELLA, V.V., Critical assessment of the pore size distribution in the rim region of high burnup $UO_2$ fuels. *J. Nucl. Mater.* 480 (2016) 138–149, https://doi.org/10.1016/j.jnucmat.2016.08.010.

[13] VESHCHUNOV, M.S., and SHESTAK, V.E., Model for evolution of crystal defects in $UO_2$ under irradiation up to high burn-ups, *J. Nucl. Mater.* 384 1 (2009) 12–18. https://doi.org/10.1016/j.jnucmat.2008.09.024.

[14] NOIROT, J., DESGRANGES, L., and LAMONTAGNE, J., Detailed characterisations of high burn-up structures in oxide fuels. *J. Nucl. Mater.* 372 2–3 (2008), 318–339.

https://doi.org/10.1016/j.jnucmat.2007.04.037.

# A REVIEW OF HIGH BURNUP FUEL BEHAVIOUR DURING NORMAL OPERATION AND ACCIDENTAL TRANSIENTS

V.I. ARIMESCU
Nufology Plus LLC.
United States of America

The term 'HBU fuel' is defined relative to the maximum burnup achieved and/or allowed during operation in NPPs. It can be observed that approximately a doubling of the discharged burnup was achieved in the first half of the nuclear industry history, to the current licensed maximum burnups of 62 MWd/kgU (fuel rod average burnup approved by the U.S. NRC); therefore, what was considered HBU 30-40 years ago is now normal burnup.

Recently, the nuclear industry has initiated activities to further extend the operational burnup range, beyond the currently approved limit. This entails research and testing of fuel behaviour at extended HBU to confirm adequate fuel performance. This means the same level of fuel reliability as for the current burnup range and no new phenomena triggered by higher than current burnup, which would require more research to understand the phenomenology and quantify the potential new processes. This is needed because of the necessity of performing design analyses to demonstrate adequate fuel behaviour during normal operation as well as during anticipated operational occurrences and accident conditions.

The following new or enhanced effects phenomena related to HBU fuel behaviour have been identified:

(a) Decreasing fuel thermal conductivity, a.k.a. burnup degradation of fuel thermal conductivity;
(b) Enhanced FGR during normal operation and thermal transients;
(c) HBS formation;
(d) Fuel–cladding bonding;
(e) Cladding corrosion and hydrogen uptake;
(f) Zirconium alloys irradiation stress-free growth;
(g) Fuel fragmentation.

The decrease of fuel thermal conductivity/diffusivity with burnup was known, however it took a relatively long time to acquire the irradiation data to quantify the phenomenon; it is a general feature of nuclear fuel research that the time span from initiation of a specific R&D programme to interpretation of the results and implementation in production is from several years to over a decade in some cases, mostly because of the time needed to perform the irradiation tests and the associated hot cell post-irradiation examinations (PIE).

Presently, the impact of irradiation damage and accumulation of fission products in the uranium dioxide matrix on thermal diffusivity is well understood and quantified in the fuel codes. Before reaching this stage, conservative biases were applied to fuel thermal models to compensate for the lack of accounting for the burnup degradation of fuel thermal diffusivity. This issue is a perfect example for an efficient way to handle a "known unknown" related to fuel behaviour, namely a cautious increase of operation space, e.g. burnup in our case, by a combination of comprehensive R&D and LTA and/or LTR programme to confirm the R&D results.

A related issue was the HBS formation at the pellet periphery and its further inward extension with increasing burnup after a burnup threshold. The large and numerous pores formed in the

HBS zone cause a drop in thermal conductivity. However, it was noticed that HBS formation led to a cleanup of the matrix form the fission products that have a deleterious impact on fuel thermal conductivity, hence on balance, the impact of HBS on fuel thermal performance is low. Among other potential negative effects of HBS, the impact on FGR was resolved by detailed hot cell PIE; combined X-ray fluorescence and EPMA and later secondary ion mass spectrometer studies clarified that most of the gas in the HBS region is contained in the large HBS pores and negligible quantities of FGR come from the HBS zone during normal steady state or transient operation.

Different approaches have been used to account for the observed FGR enhancement at HBU. To describe the proposed approaches, a summary of the phenomenological picture of FGR which emerged from the large body of experimental and theoretical studies, will be presented.

The main features of the evolution of the effective diffusion coefficient, $D_{eff}$, with burnup can be summarized as follows. At the beginning of the irradiation there is a decrease of the effective diffusion coefficient due to formation of intragranular bubbles. During this stage, both bubble radius and bubble concentration are increasing and thus gas atom capture by intragranular bubbles is increasing with the result of diminishing $D_{eff}$. Later, as irradiation progresses, trap saturation occurs, as the bubble population stabilizes both in size and concentration (as a dynamic equilibrium between creation by gas atom capture and destruction by fission fragment re-solution) and some bubble growth can take place, notwithstanding the effect of varying fission rate, which generally decreases with burnup in fuel operation in NPPs. Therefore, $D_{eff}$ reaches a minimum which is then followed by an increase as irradiation continues to higher burnups.

The burnup enhancement of FGR is most likely associated with the first phase described above. One of the proposed explanations is the hypothesis of gas atom diffusion coefficient augmentation at HBU. To that end, the increase of effective diffusion coefficient at higher burnups was attributed to the saturation of bubble capture strength and at the same time the enhancement of the other short-circuit diffusion mechanisms. One of the main candidates is the dislocation diffusion which is expected to play the major role at higher burnup and low to moderate temperatures as irradiation damage accumulates in the lattice. The role of the dislocation diffusion at lower temperatures is also the main reason for a two-term intrinsic diffusion coefficient. Other approaches hypothesized a saturation concentration limit of gas in the matrix that when attained at HBU increases the flux of gas atoms to the GB, or to the large pores of the HBS zone; however, the underlying physical mechanism was not clearly identified.

The gas accumulated on the grain boundaries (below venting threshold) is available for another FGR mechanism, namely cracking, or fracture FGR, which consists of macro or micro cracking of the fuel that intersects grain boundaries loaded with gas and release it. In some cases, the fracture can be along grain boundaries, which will enhance this type of cracking FGR. Because in many cases fuel fracture occurring during transients is fast, the cracking FGR is rapid and occurs more like a burst; therefore, this FGR process was called (transient) burst FGR.

This burst release is generally associated with micro cracking during thermal transients, such as those associated with LOCA events. The Japanese post-irradiation annealing tests of pellet fragments were interpreted based on a presumed correlation between burst FGR and fine fragmentation. The rationale was based on the large number of such post-irradiation transient heating tests, which revealed several stages of FGR, with the first one appearing as a spike of burst release at low temperatures. Although the higher temperature peaks showed diffusional characteristic kinetics, the mid-temperature peak showed mixed burst-diffusion appearance in several tests. This was interpreted as the low temperature burst being caused by micro cracking, while the one at higher temperature by GB saturation and burst release of the GB inventory.

The RIA-type of transient tests performed in test reactors, such as NSRR and CABRI in combination with hot cell PIE have shown that most of the FGR measured after the test originated from the GB inventory of the central to mid-radius part of the pellet.

Therefore, the FGR during the transient is dependent not only on the transient parameters but also on the fuel state as resulting from prior base irradiation. Unidimensional correlations between FGR and transient magnitude, as characterized by deposited enthalpy for RIA transients, or maximum temperature, are therefore incomplete, albeit highly conservative.

A relatively large database of transient FGR exists for RIA-type transients performed in test reactors. The provenience of the gas atoms for the burst FGR was determined as the GB inventory for mid burnups and, also gas on HBS pores for high and very high burnups. An overall trend of increasing FGR with transient magnitude (expressed as deposited enthalpy) was noticed, although a large spread in data was remarked, e.g. the largest transient FGR occurred for both mid-burnup and high burnup cases, while the highest FGR at low deposited enthalpy was achieved for mid-burnup cases.

The only measured FGR for a LOCA-type test was reported for a test in which cladding rupture was avoided and the measured value of ~ 18 % (of which 2-3% was released after shutdown), which is in the range of the RIA transient FGR dataset. The thermal transients caused by a RIA and a LOCA have similarities as well as differences, mainly related to the maximum temperature and heating rate. Also, the radial temperature profile during the temperature excursion is different because the different causal mechanisms, namely, very fast heat generation within the pellet in the RIA case, while drastic reduction of the heat outflow because of depressurization causing DNB in the LOCA case. This leads to a different stress evolution in the outer pellet rim zone.

The thermal stress evolution in the LOCA case is different, as the cladding temperature quickly rises when DNB occurs, and clad distension occurs soon thereafter. This leaves the pellet almost stress-free, as the temperature radial profile flattens, with the central temperatures dropping and peripheral temperatures rising. Therefore, none, or much fewer macro radial cracks develop in the rim zone in the LOCA case, but microcracking is as prone to occur as in the RIA case.

The still unresolved aspect of transient FGR is its timing, namely, the evolution of FGR during the accidental transient scenario, especially for LOCA. The puncturing data obviously reflect the total FGR as existent at the end of the transient. It is of interest to know the kinetics of the FGR process and the timing of the FGR bursts, in order to resolve the question whether transient FGR contributes or not to fuel rod pressurization during a LOCA transient and thus could potentially enhance propensity to ballooning and burst. In that respect, separate effects tests, which are representative enough of actual conditions, and in which FGR is monitored online could provide insights into FGR evolution; such tests have been initiated, but more studies are needed.

EXPERIMENTAL PROGRAMMES SUPPORTING THE DEVELOPMENT OF HBU NUCLEAR FUELS

(Session III)

**Chairperson**

**T. TVERBERG**

Norway

# RESTRUCTURING AND FISSION GAS BUBBLES IN HIGH BURNUP $UO_2$ FUELS

J. NOIROT[1], R. DOWEK[1,2], I. ZACHARIE-AUBRUN[1], T. BLAY[1], M. CABIE[3], M. DUMONT[4]

[1] Commissariat à l'Energie Atomique et aux Energies Alternatives
France

[2] Aix-Marseille University, Institut Matériaux Microélectronique Nanosciences de Provence
France

[3] Aix Marseille University
France

[4] Arts et Métiers Institute of Technology
France

## 1. INTRODUCTION

When submitted to LOCA type temperature histories, high burnup PWR (pressurised water reactors) $UO_2$ fuels tend to crack or to form small fragments, and to partially release fission gases. Consistently with other works such as heating tests in the LECA-STAR CEA hot cell facility [1–2] showed that most of the cracking and gas release occurred in the fuel parts where a high density of micrometric or quasi-micrometric bubbles were found, in particular in the HBS on the pellet's periphery and in the central volume [3–5]. In addition to the high density of micrometric bubbles, the HBS is characterized by a major restructuring of the original grains into thousands of sub-micrometric to micrometric grains. Recent post-irradiation examinations (PIE) [6–9] have shown that the central volumes of the pellets were also experiencing a restructuring of the original grains into smaller sub-grains, where the orientations of the new sub-grains spread around the original orientation of their father grain, and that bubbles originally thought intragranular and spherical were in reality on the new grain boundaries and can have complex shapes. These observations shone a new light on previously puzzling results [10–11].

This recent progress was allowed by the increasing use of improved micro-examination techniques, in particular EBSD and FIB-SEM 3D examinations (focussed ion beam, scanning electron microscope). These techniques provide local crystal orientation maps that can be used to quantify the restructuring progress, and 3D geometrical information on the bubbles.

## 2. RESULTS

The slides presented during the IAEA Technical Meeting and recently published in Refs [12–14] gave an overview of results obtained on a set of PWR $UO_2$ samples at various burnup rates, including Cr doped $UO_2$ fuels with large grains.

The EBSD examinations of the HBS on the pellets periphery mainly confirmed and refined older results showing how the HBS restructuring starts first by the formation of sub-grains with orientations close to that of the original grains followed by the formation of randomly oriented sub-grains [15]. Pole figures in detailed areas in the HBS on the pellets' periphery and slightly further from the periphery, where partial HBS restructuring is evidenced, show that HBS starts as local misorientations, turning to sub-grains with low angle boundaries. The orientation of the sub-grains progressively moves away from the original orientation, and this eventually leads to random orientations. 3D FIB-SEM examination in the HBS volumes shows the complexity of the HBS bubble surfaces, with the emerging small grain boundaries on these surfaces. It also shows bubbles formed by interconnection of growing HBS bubbles.

In the central volumes, restructuring with sub-grains orientation spreading around that of their original grain was detected by EBSD for all the fuels with a cross section average burnup higher than 60 GWd/tU, for the standard $UO_2$ samples with original grains sizes close to 10 µm, as well as for the Cr doped $UO_2$ with ~60 µm grains. The new sub-grains have, on the polished surface, an equivalent circular diameter (ECD) of around 1.1 µm whatever the original grain size. The EBSD maps exhibit various degrees in this restructuring, with radial differences within each sample as well as differences between these samples. A set of indicators were used to quantify the progress in the restructuring. Two of these indicators were the average mis2Mean and the misorientation line lengths. For each studied field, the average mis2Mean is the average angle, for all original grain in the field, of the average orientation difference between each point in the grain and the mean orientation within this grain. This indicator was applied on the central areas, where the original grains appear clearly on the maps, but cannot be applied on the HBS on the pellets' peripheries, where the original limits of the original grains are unclear or have completely disappeared. The misorientation line length, expressed in $\mu m^{-1}$ (that might be easier to understand as $\mu m/\mu m^2$) is the measurement of the length of all the lines along which a local misorientations higher than 1° is detected (be it a closed line forming a sub-grain or an apparently open line) divided by the total surface of the field. These indicators show the progress in the restructuring with the increasing burnup of the standard $UO_2$, all with a maximum on the periphery of the central zone, and with a sharp limit between the restructured volume and the surrounding un-restructured fuel. In the case of the sample examined at the highest burnup, 73 GWd/tU, the decrease in the restructuring, from this maximum, when approaching the pellet's centre, exhibits a minimum around 0.1R (0R is the centre 1R the pellet's periphery) while the indicators' values found at 0R are higher, close to those found around 0.3R. In addition, for the Cr doped $UO_2$ examined at 63 GWd/tU, the maximum in the restructuring was found at 0R and there was a continuous decrease towards the periphery of the restructured volume. Also, the progress in the restructuring was higher at 0R for this Cr doped fuel at 63 GWd/tU than on the periphery of the restructured sample for all standard $UO_2$ in the same burnup range.

The 3D FIB-SEM characterization of the bubbles in the central volumes showed that in the samples and radial positions where the restructuring is low, the bubbles mainly appear as flat and simple. This is the case at 0R in the standard $UO_2$ at 61 GWd/tU and at 0.3R in the Cr doped $UO_2$ at 63 GWd/ tU. Where the restructuring is higher, the bubbles are larger. They can remain rather flat, as at 0.3R in the standard $UO_2$ at 61 GWd/ tU where large flat bubbles interconnect, or get thicker and more complex as in the standard $UO_2$ at 73 GWd/ tU or, at 0R, in the Cr doped $UO_2$ at 63 GWd/ tU. By comparing 3D results and the restructuring progress it appears that, for these samples, the local porosity and the fraction of complex bubbles tend to increase with the restructuring progress.

Despite local interconnection of the bubbles, forming the complex bubbles, no long-distance interconnected bubble networks were found inside the original grains, along the sub-grain boundaries.

At 0R in the Cr doped $UO_2$ at 63 GWd/ tU, the 3D FIB-SEM examination allowed to determine the local distribution of the bubbles and of the metallic fission product precipitates in a volume including two GB sections and a short grain edge section. It showed layers of about 1 µm around the grain boundaries depleted in bubbles and in metallic fission product precipitates. On the grain boundaries it showed an accumulation of metallic fission product precipitates but only few bubbles and in particular no intergranular interconnected bubbles.

FIB and SEM characterization, combined with EPMA and secondary ion mass spectrometer measurements, providing the bubble content, both as average values and for selected single bubbles provided fission gas density in bubbles. The lowest densities (the highest molar volumes) were found is the situations where the intragranular bubbles (inside the original grains, but intergranular relatively to the sub-grains) had grown thicker (the Cr doped $UO_2$ 63 GWd/ tU at 0R and even more the standard $UO_2$ 73 GWd/ tU on the periphery of the central volume. Expressed as the number of tri-vacancy (Schottky defects) per fission gas atom in these bubbles, the atomic volumes measured ranged between 2 Ω/atom and 8 Ω/atom.

Works on the fracture properties in these restructured zones using FIB-SEM cantilever mechanical testing and pursuing the work started as described in Ref. [16], and transmission electron microscope (TEM) examinations conducted on thin foils prepared using the same FIB-SEM, are still ongoing.

## 3. CONCLUSION

The data produced in this work are to be used to improve high burnup $UO_2$ models in their design and validation.

A general understanding of this central restructuring could be that a certain burnup and a certain temperature level are necessary to observe such a phenomenon, but that higher temperatures, by allowing increased defect annealing and partial FGR, moderate this restructuring. However,

- The often sharp limit between the restructured volume and its un-restructured surrounding,
- The discontinuous radial evolution in the standard $UO_2$ highest burnup sample, with a local maximum at 0R,
- The differences in the restructuring progress and on the radial profile features between the Cr doped and the standard $UO_2$,

are reasons for considering the possibility that local temperatures, linear powers and burnups might not be, at least directly, the only parameters to be considered.

These characterizations and complementary observations on the original grain boundaries also raise questions on the actual mechanisms of fission gas transfer to the grain boundaries and along the grain boundaries and therefore on FGR at high burnup in case of limited linear heat rates.

All these results show the complexity of the phenomena involved in the restructuring and their dependence to the fuel type and the irradiation conditions. We therefore encourage other hot laboratories worldwide to work on the central restructuring in high burnup fuels, using increasingly accurate techniques, and to share theirs results, to cover a wider variety of fuels, of irradiation situations and of examination approaches.

## REFERENCES


[1] NARUKAWA, UDAGAWA, T.Y., "Study on mechanism and threshold conditions for fuel fragmentation during loss-of-coolant accident conditions", Proc. Int Conf. TopFuel2021, Santander (Spain), 24–28 October (2021).

[2] YAN, Y., HARP, J., LINTON, K., BROWN, N., "Post test examinations of a LOCA sample from an irradiated high burnup PWR M5 fuel rod", Proc. Int Conf. TopFuel2021, Santander (Spain), 24–28 October (2021).

[3] MARCET, M., PONTILLON, Y., DESGRANGES, L., NOIROT, J., LAMONTAGNE, J., AUBRUN, I., PASQUET, B., VALOT, C., COGNON, H., BLANPAIN, P., "Contribution of High Burn-up Structure to fission gas release under transient conditions", Proc. Int Conf. TopFuel2009, Paris, France, 2009.

[4] NOIROT, J., BLAY, T., LAMONTAGNE, J., FAYETTE, L., PONTILLON, Y., PUJOL, X., "Size and radial origin of fragments formed while heating a 83 GWd/tU PWR fuel up to 1200 °C", paper presented at LOCA Workshop, Fuel fragmentation, relocation and dispersal (FFRD) – experimental basis, mechanisms and modelling approaches, Aix-en-Provence (France), 2015.

[5] NOIROT, J., PONTILLON, Y., YAGNIK, S., TURNBULL, J.A., TVERBERG, T., Fission gas release behaviour of a 103 GWd/tHM fuel disc during a 1200°C annealing test, *J. Nucl. Mater.* 446 (2014) 163–171.

[6] NOIROT, J., ZACHARIE-AUBRUN, I., BLAY, T., Focused ion beam–scanning electron microscope examination of high burn-up $UO_2$ in the center of a pellet, *Nucl. Eng. Technol.* 50 (2018) 259–267.

[7] GERCZAK, T.J., PARISH, C.M., EDMONDSON, P.D., BALDWIN, C., TERRANI, K.A., Restructuring in high burnup $UO_2$ studied using modern electron microscopy, *J. Nucl. Mater.* 509 (2018) 245–259.

[8] CAPPIA, F., WRIGHT, K., FRAZER, D., BAWANE, K., KOMBAIAH, B., WILLIAMS, W., FINKELDEI, S., TENG, F., GIGLIO, J., CINBIZ, M.N., HILTON, B., STRUMPELL, J., DAUM, R., YUEH, K., JENSEN, C., WACHS, D., Detailed characterization of a PWR fuel rod at high burnup in support of LOCA testing, *J. Nucl. Mater.* 569 (2022) 153881.

[9] JÄDERNÄS, D., CORLEONI, F., PURANEN, A., GRANFORS, M., LYSELL, G., TEJLAND, P., LUTZ, D., HALLSTADIUS, L., "Microstructural and chemical characterization of ramp tested additive fuel", Proc. Int Conf. TopFuel2013, Charlotte, NC (2013).

[10] NOIROT, J., NOIROT, L., DESGRANGES, L., LAMONTAGNE, J., BLAY, T., PASQUET, B., MULLER, E., "In Fission Gas Inventory in PWR High Burnup Fuel: Experimental Characterization and Modeling", paper presented at ANS LWR Fuel Performance, Orlando, Florida, USA, 2004.

[11] NOIROT, J., ZACHARIE-AUBRUN, I., DESGRANGES, L., HANIFI, K., LAMONTAGNE, J., PASQUET, B., VALOT, C., BLANPAIN, P., COGNON, H., High Burnup Changes in $UO_2$ fuels irradiated up to 83 GWd/t in M5 Claddings, *Nucl. Eng. Technol.* 41 (2009) 155–162.

[12] DOWEK, R., CAGNA, C., NOIROT, J., ZACHARIE-AUBRUN, I., BLAY, T., HANIFI, K., BIENVENU, P., ROURE, I., M. CABIÉ, DUMONT, M., Determination of the pressure in micrometric bubbles in irradiated nuclear fuels, *J. Nucl. Mater.* 543 (2021) 152591.

[13] NOIROT, J., DOWEK, R., ZACHARIE-AUBRUN, I., BLAY, T., CABIÉ, M., DUMONT, M., Restructuring in high burn-up pressurized water reactor $UO_2$ fuel central parts: Experimental 3D characterization by focused ion beam—scanning electron microscopy, *J. Appl. Phys.* 132 (2022) 195902.

[14] ZACHARIE-AUBRUN, I., DOWEK, R., NOIROT, J., BLAY, T., CABIÉ, M., DUMONT, M., Restructuring in high burn-up $UO_2$ fuels: Experimental characterization by electron back-scattered diffraction, *J. Appl. Phys.* 132 (2022) 195903.
[15] BENGTSSON, S., "Examination of sub-grain formation in high burnup $UO_2$ fuel using the EBSP method", paper presented at IAEA Technical committee meeting on advances in pellet technology for improved performance at high burnup, IAEA-TECDOC-1036, Tokyo (Japan), 1996.
[16] HENRY, R., ZACHARIE-AUBRUN, I., BLAY, T., CHALAL, S., GATT, J.M., LANGLOIS, C., MEILLE, S., Fracture properties of an irradiated PWR $UO_2$ fuel evaluated by micro-cantilever bending tests, *J. Nucl. Mater.* 538 (2020) 152209.

# EXPERIENCE WITH IRRADIATION TESTING IN RELATION TO HIGH BURNUP INVESTIGATIONS AT THE OECD HALDEN REACTOR PROJECT

T. TVERBERG
Institute for Energy Technology
Norway

## 1. INTRODUCTION

The Halden Reactor started in the late 1950's, and until its eventual closure in 2018, the research conducted there has made important research contributions on in-pile behaviour of water-cooled reactor fuels and materials. The establishment of the OECD HRP facilitated comprehensive international research programmes related both to fuel and cladding, as well as materials related phenomena. It included the ability to instrument fuel rods that allowed for concurrent on-line monitoring of major parameters like fuel centre temperature, fuel rod pressure, fuel axial displacement, cladding deformation on both axial and radial direction etc. While an important part of such tests started with fresh fuel, HBU phenomena became increasingly important. A major milestone was in the 1980's when the technology for refabrication and instrumentation of commercially irradiated fuel rods (the so-called re-instrumentation of irradiated fuel) was developed for employment in the hot cells at the Kjeller facilities of the Institute for Energy Technology [1–2]. The introduction of external loop systems, facilitating the possibility of simulated LWR thermal hydraulic conductions with controlled water chemistry, was also an important development. This was particularly the case for cladding corrosion experiments and studies related to the irradiation assisted stress corrosion cracking [3], which became an increasingly important part of the research activities at the Halden reactor, but also for studies of phenomena related to the interaction of pellet and cladding at HBU. The present work aims at giving a brief overview of, with the main emphasis to experiments devoted to study fuel behaviour at HBU.

## 2. HIGH BURNUP FUELS TESTING METHODOLOGIES IN THE HRP OVER THE YEARS

Instrumented fuels tests on HBU fuel were conducted either starting from fresh fuel, irradiated over several years while continuously collecting data on properties like fuel temperature, rod pressure and FGR, pellet cladding interaction through cladding elongation and/or cladding diameter sensors. Fuels in this type of irradiation would invariably have enrichments higher than 5% (typically 10–13 wt. % $^{235}U$ for tests with BU >90 MWd/kgU) in order to achieve HBU in a reasonable time frame. Of note in this respect were also the fuel disc test type where instead of fuel pellets, the fuel would be stacked between molybdenum discs for rapid burnup accrual. These tests could often have fuel with enrichment up to 20% facilitating a burnup accumulation of 100 MWd/kg in a few calendar years. While these disc irradiation tests would also be instrumented, and in many cases accumulate data on e.g. fuel volume change and FGR as function of burnup, their main purpose would typically be to produce HBU fuel samples (in some cases in excess of 100 MWd/kg) for later studies in hot cells using advanced techniques [4].

Another type of HBU fuels testing was developed during the 1980's, where irradiated fuel segments from commercial irradiations in LWRs were transported to the Kjeller hot cells for re-fabrication and instrumentation for test programmes in the Halden reactor. The subsequent re-irradiation in the Halden reactor would for such re-instrumented tests regularly also be conducted in LWR experiment loops where thermal hydraulic and water chemistry conditions typical of PWR and/or BWR and/or WWER (or even CANDU) reactor conditions could be simulated.

## 3. EXAMPLES OF CONDUCTED TESTS ON FUEL AT VERY HIGH BURNUP

### 3.1. Halden reactor (HBWR) irradiations (fresh pellet fuel)

As mentioned above, pellet tests starting from fresh fuel were typically done with fuel higher than 5% enriched fuel pellets, and conducted in HBWR rigs (240°C, 34 bar $D_2O$) allowing for highly instrumented typically in a six rod configuration. The objectives of such long term irradiations would be integral testing of with highly instrumented rods Fuel rod instrumentation in such long term irradiation tests would typically include:

- Fuel temperature (measured thanks to fuel thermocouples, TF), rod pressure, fuel elongation, clad elongation, clad diameter;
- gas line + gamma spectroscopy.

In order to be able to achieve HBU within a reasonable time frame, rods in such tests would also usually be of higher than 5% enrichment (typically 10–13 wt. % $^{235}U$ for tests with BU >90 MWd/kgU). A list of some HBU experiments starting with fresh fuel is provided in Table 1.

TABLE 1. EXAMPLES OF HRP HBU HBWR IRRADIATIONS

| Test ID | Max. BU [MWd/kgU] | Enr. [wt.% $^{235}U$] | Test objectives | Instrumentation | Remarks |
|---|---|---|---|---|---|
| IFA-504 | 90 | 10 | Gas flow test. FGR measurements. Study effect of fill gas composition on fuel T. | TF, EC, γ-spectr. | $UO_2$ variants, diff. gr. sizes, hollow and solid pellets |
| IFA-515.10 | 85 | 11.5 / 13 | Investigate the thermal behaviour of Gd fuel as compared to $UO_2$ fuel | ET (fuel centre T) | $UO_2$ + Gd |
| IFA-519.9 | 90 | 13 | To evaluate the effect of load follow on FGR | PF | $UO_2$, diff. gr. sizes |
| IFA-533.2 | 90 | 9.88 | Investigate fuel thermal conductivity degradation at high BU | TF | Rods pre-irr. in IFA-409 |
| IFA-550.8 | 65 | 10 / 11 | To investigate the effect of load follow operation on PCMI. | EC, DG | |
| IFA-676 | 75 | 10 | Comparative test of large gr. and std. gr. hollow pellet (WWER) $UO_2$ fuel, and WWER Gd fuel. | TF, EF, PF, EC | FGR data at HBU. |

Instrumentation:
TF: Fuel thermocouple — EF: Fuel elongation — PF: Rod inner pressure
ET: Expansion thermometer (Fuel centre T) — EC: Cladding elongation — DG: Diameter gauge

4. IFA-519.9 (HRP)

An example of a HBU irradiation starting from fresh fuel was the IFA-519.9 experiment, where rods with different grain size $UO_2$ fuels, that had first been irradiated in the IFA-429 rig, were re-instrumented with pressure transducers to study the effects of fuel grain size and pellet–clad gap on FGR at extended burnup. Periods with power cycling (facilitated through the use of a $^3$He coil incorporated into the rig) was also included in this irradiation. Some main observations from the IFA-519.9 experiment were [1]:

- Reduced FGR from large grain fuel;
- Most efficient at modest power and low FGR (10%);
- At high power and high release, an increase in grain size is less effective;
- Power cycling did not show enhanced FGR.

An example of FGR data collected from the IFA-519.9 experiment is provided in Ref. [1].

### 4.1. Refabricated LWR segments

Experiments with re-fabricated fuel segments from fuel irradiated in commercial LWR plants, were an important addition to the previous irradiation capabilities at Halden from the 1980's. The ability to re-instrument with fuel thermocouples in the refabrication process from the 1990's, that also facilitated fuel temperature monitoring for such refabricated rods was particularly important. Over the years, a wealth of data from re-instrumented test segments from HBU fuel rods have been collected (examples of HBU irradiations with re-instrumented fuel irradiated in LWR plants are provided in Table 2). Topics covered include studies on fuel thermal conductivity degradation with burnup, as well as FGR and PCMI at extended burnup. A special test programme to study the extent of cladding lift-off — and its potential impact of fuel temperature — from extended periods of high overpressure was also conducted in the IFA-610 test series, where rods were instrumented with gas lines to provide high rod overpressure (and also gamma spectroscopy for FGR studies), together with fuel thermocouples, cladding elongation sensors [5, 6].

TABLE 2. EXAMPLES OF HRP HBU IRRADIATIONS WITH RE-INSTRUMENTED LWR FUEL

| Test ID | Max. BU [MWd/kgU] | Fuel Type | Test objectives | Instrumentation |
|---|---|---|---|---|
| IFA-534.14 | 62 | $UO_2$ various gr. size | PCMI | EC |
| IFA-597.2/.3 | 70 | $UO_2$ | FGR + PCMI | TF, PF, EC |
| IFA-610 series | 60 - 80 | $UO_2$ and MOX, PWR, BWR, WWER | Lift-off / overpressure | TF, EC, $\gamma$ -spectr., hydr. diam./gas flow |
| IFA-700.4 | 69 | $UO_2$ | FGR + PCMI | TF, PF, EC |
| IFA-720.2 | 63 | 8% Gd | FGR + PCMI | TF, PF, EC |
| IFA-720.3 | 60 | Cr doped + $UO_2$ | Comparative study of FGR of Cr doped + $UO_2$ | TF + PF |

Instrumentation:
TF: Fuel thermocouple EF: Fuel elongation PF: Rod inner pressure

ET: Expansion thermometer (Fuel centre T)  EC: Cladding elongation  DG: Diameter gauge

## 5. LWR FUEL IRRADIATION: IFA-597.3 (HRP)

An example of a test with re-instrumented LWR fuel was the IFA-597.3 experiment, where BWR rods irradiated to ~> 65 GWd/tU in a commercial reactor were re-instrumented with fuel thermocouples and fuel rod pressure gauges or cladding elongation detectors to study FGR and PCMI at HBU. A data point for onset of FGR (>1%) is shown in Fig. 1, together with other data collected in Halden tests as well as from other experimental programmes, that were used for establishing the Halden (Vitanza) 1% FGR threshold [7] (also included in Fig. 1). The observation from the IFA-597.3 experiment was onset of >1% FGR at a fuel centre temperature of 950°C at an oxide burnup of 59 MWd/kg$UO_2$, which is about 100°C below the Halden threshold curve extrapolated to this burnup.

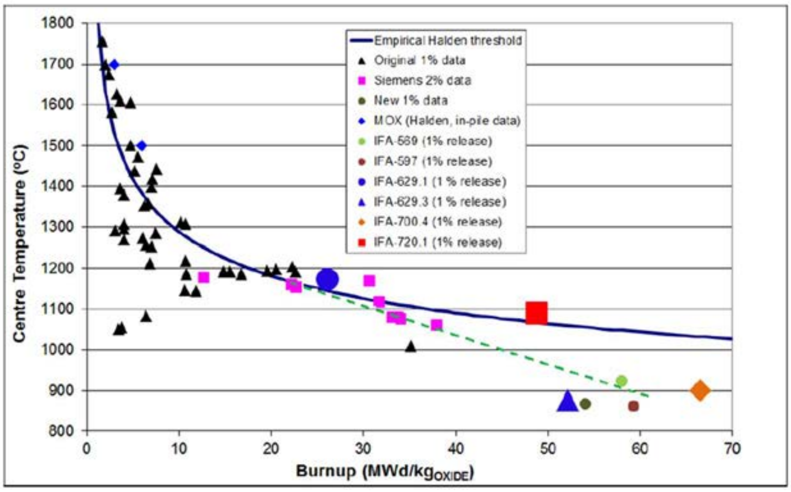



*FIG. 1. Halden (Vitanza) 1% FGR threshold and data collected at different burnups (courtesy of HRP [8]).*

### 5.1 Fuel disc irradiations

A special type of Halden irradiation for studies of fuel at extended burnup, were the fuel disc irradiation designs. In these irradiations, the fuel consisted of thin (typically 1 mm thick) discs with increased enrichment (up to 19.8% $^{235}U$). These discs would be stacked between thicker Mo discs and mounted in fuel rods with Zr alloy cladding. This design facilitated conditions for rapid fuel burnup accrual at near isothermal irradiation conditions. The main objective of such fuel disc tests was often to produce HBU samples for advanced post-irradiation examinations (PIE) at external laboratories, but individual rods in these experiments would usually also be instrumented for monitoring in-pile behaviour during burnup accumulation. Fuel rod instrumentation for such tests would typically consist of:

- Fuel thermocouples (TF) for monitoring fuel disc irradiation temperature:
    - Rods with TF measurements would in several cases also include gas lines which when connected to gamma spectroscopy equipment allowed for measurements of radioactive FGR through the irradiation.
- Fuel rod pressure transducers (PF) to monitor FGR and fuel volume change:

- Fuel stack elongation detectors (EF) to monitor fuel volume change:

A list of fuel disc irradiations to HBU is provided in Table 3.

TABLE 3. SELECTED HBU DISC IRRADIATIONS

| Test ID | Max. BU [MWd/kgU] | Enr. [wt.% $^{235}U$] | Test objectives | Instrumentation | Remarks |
|---|---|---|---|---|---|
| IFA-569 (EPRI/NFIR) | 77 | 19.8 | Produce uniform irradiated fuel samples for HBU | TF, g-spectr. | $UO_2$ + Gd |
| IFA-601 (CRIEPI/HBRP) | 100 | 19.8 | Produce uniform irradiated fuel samples for HBS studies (PIE) | TF, g-spectr. | $UO_2$ + Gd |
| IFA-655 (HRP) | 130/200 | 19.8 | Study HBU effects of FGR and fuel volume change. Transient test studies performed in some rods after irradiation completion | TF, EF, PF, g-spectr. | $UO_2$ + MOX |
| IFA-649 (EPRI/NFIR) | 50, 75, 100 | 19.8 | Produce uniform irradiated fuel samples for HBS studies (PIE) | TF, PF | $UO_2$ variants + MOX |

Instrumentation:
TF: Fuel thermocouple EF: Fuel elongation PF: Rod inner pressure
ET: Expansion thermometer (Fuel centre T) EC: Cladding elongation DG: Diameter gauge

## 6. IFA-655 (HRP)

An example of a fuel disc experiment for studies of HBS fuel behaviour was the IFA_655 experiment. This experiment consisted of 12 short length rods, each with 25 discs (8.d mm disc diameter; 1 mm disc height) divided into two clusters with 6 rods each. In each cluster, 4 rods $UO_2$ rods (19.8% enrichment) and 2 MOX rods (14% Pu) were irradiated. The $UO_2$ fuel was consisted of standard and large grain $UO_2$ (two of each in each cluster) while the MOX rods were of a homogeneous and heterogeneous type (one each in each cluster). Target irradiation temperatures were ~500°C and 700°C. Instrumentation consisted of fuel temperature and gas lines for the upper cluster rods, and fuel pressure and elongation for the lower cluster rods. Discharge burnups were ~130 MWd/kgHM for the $UO_2$ rods, and ~110 MWd/kgHM for the MOX rods. Two of the $UO_2$ rods and both MOX rods from the lower cluster were later subjected to transient testing in a separate rig for studies of FGR from HBS fuel. Figure 2 shows data from the fuel elongation sensors in the lower cluster rods through the full irradiation (only data at hot stand-by conditions are shown). The data indicate enhanced swelling rates for all fuel variants above a burnup of ~70 MWd/kgHM.

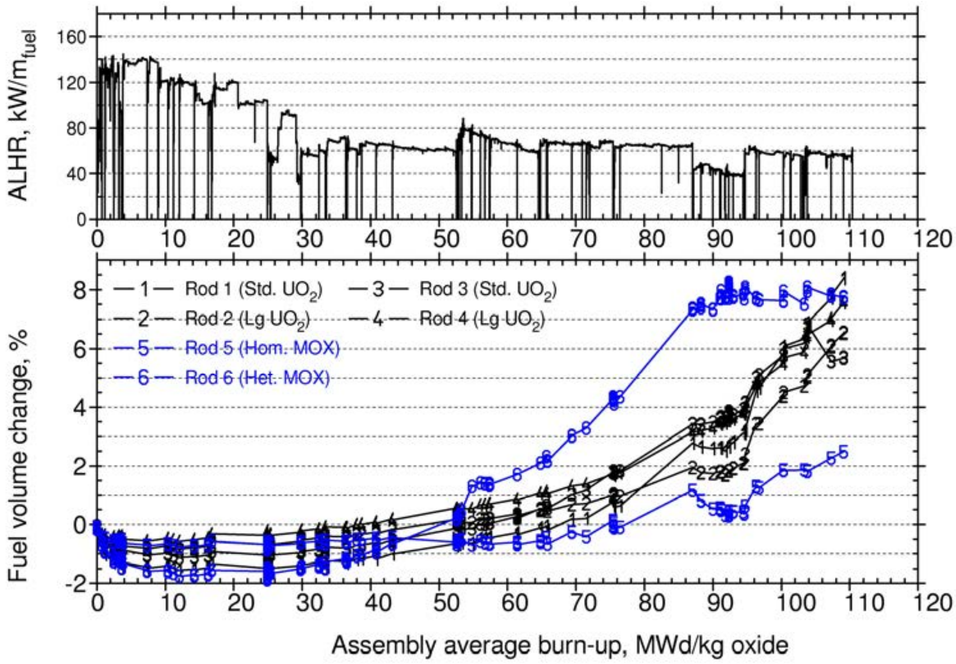



*FIG 2. Fuel volume change deduced from fuel stack elongation measurements in IFA-655 (courtesy of HRP [9]).*

### 6.1. LOCA studies of HBU fuel: the IFA-650 LOCA test series

The IFA-650 LOCA test series was incorporated in the HRP programme as a contribution to international efforts to re-examine the LOCA safety criteria and to verify their continued validity for high burnup fuel and new types of cladding materials. The test series was designed to look at in-reactor effects that are different from those obtained in out-of-reactor tests. The primary objectives were defined as [10]:

1. Measure the extent of fuel (fragment) relocation into the ballooned region and evaluate its possible effect on cladding temperature and oxidation.
2. Investigate the extent (if any) of "secondary transient hydriding" on the inner side of the cladding above and below the burst region.

A third objective was activated in 2009 and applied to the tests executed since then:

3. Measure the release of iodine and caesium from failed fuel in LOCAs.

Aside from three commissioning tests with fresh fuel, a total of 13 tests with pre-irradiated LWR fuel were conducted between 2003 and 2017, with burnup ranging from ~45 MWd/kgU to above 90 MWd/kgU. The tests (all with $UO_2$ fuel) that were carried out in the IFA-650 test series consisted of:

- Three commissioning and system check tests (with fresh fuel);
- Seven tests with PWR HBU fuel ;
- Four tests with BWR medium and/or HBU fuel ;
- Two tests with WWER medium and/or HBU fuel.

The test scheme for the IFA-650 LOCA tests consisted of a pressure flask (connected to a water loop) and a single fuel rod inserted in it. To simulate decay heat in the fuel rod, a low level of nuclear power generation was used and an electrical heater surrounded the rod to simulate the heat from neighbouring rods, as in a real fuel bundle.

In general, the rod instrumentation during an IFA-650 LOCA test would include two cladding thermocouples at the upper part of the rod (at the same elevation, 90° apart), a thermocouple at the lower part, two heater thermocouples at different elevations, a cladding extensometer and a rod pressure sensor. For power calibration and axial power distribution purposes, the test rig contained coolant thermocouples at inlet and outlet, as well as axially distributed vanadium neutron detectors.

## 7. MAIN OBSERVATIONS FROM THE IFA-650 LOCA TEST SERIES

Some main observations from the IFA-650 LOCA test series in terms of fuel fragmentation, relocation and dispersal are from the HBU tests in this test series are listed below:

- Fuel fragmentation:
  - The extent of fine fragmentation is strongly correlated with burnup.
  - From the PIE results it was found that only rods with burnup >60 MWd/kgU showed an appreciable amount of fine fragmentation:
    - It was found that a clad distension above 5% combined with an estimated pellet surface temperature above 870–930°C during LOCA was necessary for extensive fuel fragmentation to take place.
- Fuel relocation:
  - The extent of fuel relocation is correlated with the amount of fine fragmentation.
  - Regions with coarse fragments had less propensity for relocation, even when clad distension was large enough to allow for fuel to move into it.
- Fuel dispersal:
  - Due to how the IFA-650 test series was designed and executed, the amount of fuel dispersal during the tests was not quantifiable from the available examination techniques like e.g. post-test gamma scanning [11].
  - Nevertheless, some important information can be extracted:
    - Appreciable fuel dispersal was not seen for fuel with burnup < 60 MWd/kgU;
    - Where photographs were taken of dispersed fuel, these invariably showed predominantly very fine fragments.

In addition to the above observations related to FFRD, the plenum pressure was measured during all tests, and data from these can assist in modelling (transient) FGR during the LOCA event [12, 13]. In particular, one test (IFA-650.14), was a non-burst test where post-test rod puncturing data are also available as valuable data point for validation of transient FGR calculations.

### 7.1. Overview of data sets available at IAEA and/or NEA (IFPE/DATIF)

A number of datasets from Halden Reactor irradiations — in the frame of the HRP or in bilateral projects — have been made available to the nuclear community either through the NEA IFPE (or DATIF) [14, 15] database or as datasets used in IAEA code benchmark exercises like e.g. FUMEX [16, 17] and FUMAC [18]. Tables 4 and 5 list some of the available datasets.

TABLE 4. SOME HALDEN REACTOR HBU DATA SETS IN IFPE

| | |
|---|---|
| NEA-1599 | IFPE/FUMEX-I, Data from OECD HRP for FUMEX-1 (Fuel Modelling at Extended Burnup) |
| NEA-1546 | IFPE/IFA-429, FGR, Thermal Behaviour $U0_2$ Fuel, Halden Reactor |
| NEA-1488 | IFPE/IFA-432, FGR, Mechanical Interaction BWR Fuel Rods, Halden |
| NEA -1860 | IFPE/IFA-519.9, Three PWR rods irradiated to 90 MWd/kg $UO_2$ |
| NEA-1549 | IFPE/IFA-533.2, Fuel Thermal Behaviour at HBU, Halden Reactor |
| NEA-1684 | IFPE/IFA-534.14 R1, FGR as a function of burnup at high power (52-55 MWd/kg) |
| NEA-1548 | IFPE/IFA-535, FGR, Power Ramps, HBU Fuel |
| NEA-1803 | IFPE/IFA-585, In-Reactor Creep Behaviour of Zircaloy-2 and Zircaloy-4 under Variable Loading Conditions |
| NEA-1772 | IFPE/IFA-597-MOX, Hollow and solid MOX rods experiments |
| NEA-1685 | IFPE/IFA-597.3, centreline temperature, FGR and clad elongation at high burnup (60-62 MWd/kg) |
| NEA-1861 | IFPE/IFA-629.1, The Re-irradiation of MIMAS-MOX Fuel in IFA-629.1 |
| - | Also, the various IAEA FUMEX modelling exercises included several data set from Halden irradiations |

TABLE 5. HALDEN REACTOR LOCA (IFA-650) DATA SETS

| | |
|---|---|
| NEA-1862 | IFPE/IFA-650.1 & 2, LOCA testing at Halden, Two experiments, IFA-650 series |
| FUMAC | IAEA-TECDOC-1889 Fuel Modelling in Accident Conditions (FUMAC) |

## REFERENCES


[1] WIESENACK, W., Review of Halden reactor project high burnup fuel data that can be used in safety analyses, *Nucl. Eng. Des.* 172 (1997) 83–92.

[2] WIESENACK, W., et al., The OECD Halden reactor project fuels testing programme: methods, selected results and plans, *Nucl. Eng. Des.* 207 (2001) 189–197.

[3] KARLSEN, T.M., Achievements and further plans for the OECD Halden reactor project materials programme, *Nucl. Eng. Des.* 207 (2001) 199–206.

[4] KINOSHITA, M., et al., "High Burnup RIM Project (II) Irradiation and Examination to Investigate RIM-Structured Fuel", paper presented at ANS Meeting on Light Water Reactor Fuel Performance, Park City, Utah, 2000.

[5] VITANZA, C., Overview of the OECD–Halden reactor project. *Nucl. Eng. Des.* 207 (2001) 207–221. 10.1016/S0029-5493(00)00384-8.

[6] KHVOSTOV, G. et al., Analysis of selected Halden overpressure tests using the FALCON code, *Nucl. Eng. Des.* 310 (2016) 395–409.

[7] VITANZA, C. et al., "Fission gas release from $UO_2$ pellet fuel at high burn-up", paper presented at ANS meeting, Portland, OR, 1979.

[8] KLECKA, L., Summary report of the fission gas release experiments in IFA-629/700/720, internal report, Halden Reactor Project, HWR-1012, 2011.

[9] TVERBERG, T., The High Burn-up Disk Irradiation Test, IFA-655: Final Report on the In-pile Performance, internal report, Halden Reactor Project, HWR-837, 2008.

[10] OECD NUCLEAR ENERGY AGENCY, Report on Fuel Fragmentation, Relocation and Dispersal, OECD 2016, No. NEA/CSNI/R(2016)16 (2016).

[11] BRANKOV, V. et al., Characterization of the relocated and dispersed fuel in the Halden reactor project LOCA tests based on gamma scan data, *Nucl. Eng. Des.* 300 (2016) 97–106.

[12] KHVOSTOV, G., Modelling effects of transient FGR in LWR fuel rods during a LOCA, *J. Nucl. Mater.* 559 (2022) 153446.

[13] KHVOSTOV, G. et al., Some insights into the role of axial gas flow in fuel rod behaviour during the LOCA based on Halden tests and calculations with the FALCON-PSI code, *Nucl. Eng. Des.* 241 (2011) 1500–1507.

[14] OECD NUCLEAR ENERGY AGENCY, International Fuel Performance Experiments (IFPE) database (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_36358/international-fuel-performance-experiments-ifpe-database.

[15] OECD NUCLEAR ENERGY AGENCY, DATabase for IFpe (DATIF) (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_33943/database-for-ifpe-datif.

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel modelling at extended burnup: report of the Co-ordinated Research Programme on Fuel Modelling at Extended Burnup – FUMEX, IAEA- TECDOC-998, IAEA, Vienna (1998).

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling at extended burnup (FUMEX-II) – Report of a Coordinated Research Project, IAEA-TECDOC-1687, IAEA, Vienna (2012).

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling in Accident Conditions (FUMAC) – Final Report of a Coordinated Research Project, IAEA-TECDOC-1889, IAEA, Vienna (2019).

# JAPAN ATOMIC ENERGY AGENCY'S STUDIES ON HIGH BURNUP LWR FUEL BEHAVIOUR UNDER REACTIVITY-INITIATED ACCIDENT CONDITIONS

Y. TANIGUCHI, Y. UDAGAWA
Japan Atomic Energy Agency
Japan

## 1. INTRODUCTION

The Japan Atomic Energy Agency (JAEA) has performed extensive research programmes to better understand the transient behaviour of LWR fuels under RIA conditions. RIA simulated pulse irradiation tests on HBU LWR fuels, irradiated in commercial reactors, were conducted at the NSRR, including under the Advanced LWR Fuel Performance and Safety (ALPS) research programme followed by the subsequent ALPS-II programme (launched in 2010). These experimental programmes have thus far added more than 20 data points to the NSRR experiment database and extended its burnup range to 84 GWd/tU. In these programmes, the primary interest has been on the influence of burnup extension on PCMI failure limit and transient FGR.

Based on the NSRR database on more than 1000 fresh fuel tests and about 100 irradiated fuel tests including the extensive HBU fuel test campaign (ALPS + ALPS-II), it was found that the burnup extension significantly affects the PCMI failure limit, heat transfer and transient FGR as described below:

- PCMI failure limit (see Fig. 1)
  - Cladding hydrogen embrittlement is the primary factor; H content and hydride morphology are important parameters as understood from the comparison between Tests HBO-1 and VA-1 and Tests VA-5 and LS-1, respectively [1-4].
  - The effect of clad initial temperature (onset of transient) is also significant: increase in failure limit with ductility enhancement due to cladding temperature rise as understood from the comparison between Tests VA-1 and VA-3 and Tests VA-5 and VA-7 [4].
  - The effect of irradiation damage accumulated in the cladding matrix is not clear [1].
  - The MOX fuel tests at the NSRR did not present clear signs of a MOX-specific effect i.e. additional PCMI loading by gas-induced fuel pellet transient swelling: see the results of Tests BZ-1 and BZ-2 [3].

- Heat transfer
  - Cladding surface heat transfer is significantly enhanced, probably owing to surface wettability enhancement by irradiation, and peak cladding temperature is suppressed [5].

- Transient FGR [4]
  - Transient FGR continuously increases with burnup above ~30 GWd/tU, reaching ~40% above 70 GWd/tU. Enthalpy dependency, suggested in the earlier studies, has become less evident through transient FGR data accumulation (see Fig. 2).

## 2. REMAINING KNOWLEDGE GAP AND ONGOING CHALLENGE

JAEA is planning to address one of the knowledge gaps related to transient FGR. While the burnup effect on transient FGR is evident, many datapoints of the transient FGR database are to be interfered by the failure event (e.g. FCI) and the evaluation has fully relied on post-irradiation examinations (PIE): there is quite limited knowledge of FGR kinetics such as release rate and gas communication in the rod axial direction. Furthermore, there are limited data for HB MOX and doped fuels, and no data for ATF. To address this knowledge gap, Fission Gas Dynamics test [6] focusing on FGR and/or migration behaviour has been developed in collaboration with IRSN, as a complementary effort to the integral RIA tests of NSRR, CABRI and expected TREAT experiments.

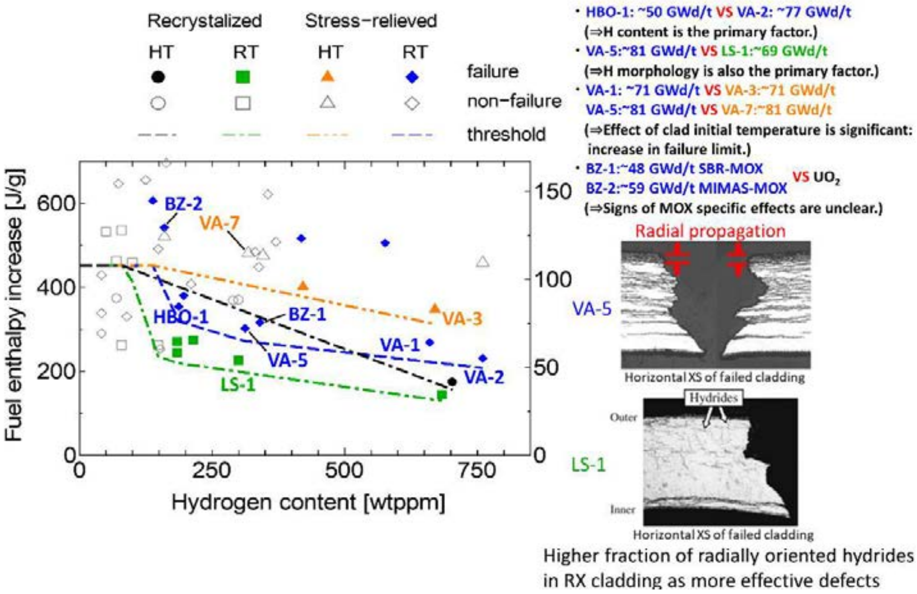
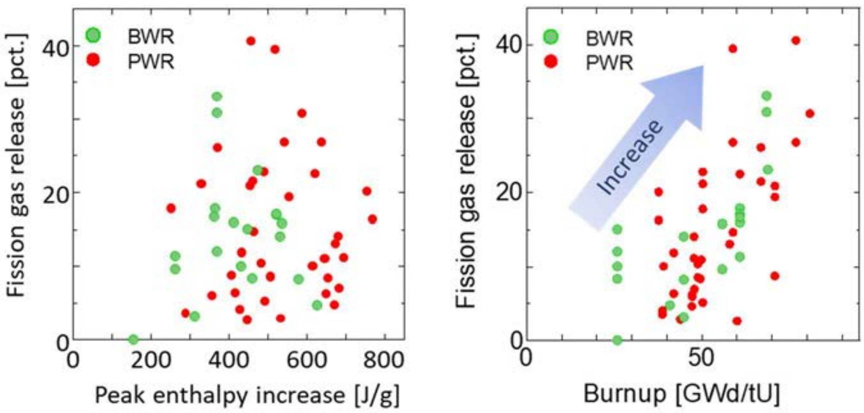



*FIG. 1. PCMI failure thresholds: fuel enthalpy increases at PCMI failure expressed as a function of cladding hydrogen content (courtesy of Yoshinori Taniguchi, Japan Atomic Energy Agency).*




*(a)* *(b)*

*FIG. 2. Relationship between FGR data of the existing RIA-simulated experiment data and (a) peak enthalpy increase or (b) burnup (courtesy of Yoshinori Taniguchi, Japan Atomic Energy Agency).*

## REFERENCES


[1] UDAGAWA, Y., SUGIYAMA, T. and AMAYA, M., Thresholds for failure of high-burnup LWR fuels by Pellet Cladding mechanical interaction under reactivity-initiated accident conditions, *J. Nucl. Sci. Technol.* 56 (2019) 1063–1072.

[2] FUKETA, T., SASAJIMA, H. and SUGIYAMA, T., Behavior of High-burnup PWR Fuels with Low-tin Zircaloy-4 Cladding under Reactivity-initiated-accident Conditions, *Nucl. Technol.* 133 (2001) 50–62.

[3] SUGIYAMA, T., UMEDA, M., FUKETA, T., SASAJIMA, H., UDAGAWA, Y., and NAGASE, F., Failure of high burnup fuels under reactivity-initiated accident conditions, *Ann. Nucl. Energy* 36 (2009) 380–385.

[4] MIHARA, T., UDAGAWA, Y., SUGIYAMA, T. and AMAYA, M., Mechanical failure of high-burnup fuel rods with stress-relieved annealed and recrystallized M-MDA cladding under reactivity-initiated accident conditions, *J. Nucl. Sci. Technol.* 58 (2019) 872–885.

[5] UDAGAWA, Y. SUGIYAMA, T., SUZUKI, M. and AMAYA, M., “Experimental analysis with RANNS code on boiling heat transfer from fuel rod surface to coolant water under reactivity-initiated accident conditions”, INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA-TECDOC-CD-1775 (2013).

[6] UDAGAWA, Y., SUGIYAMA, T. and AMAYA, M., “Recent Research Activities Using NSRR on Safety Related”, Proc. Int. Conf. ICAPP 2016, San Francisco, CA, USA (2009).

# INTEGRAL BEHAVIOUR OF HIGH BURNUP FUEL ROD UNDER LOCA: EFFECT ON STEAM OXIDATION, INTEGRAL FFRD EXPERIMENT, AND REGULATORY IMPLICATIONS

Y. LEE, D. KIM, H. YOOK
Seoul National University
Korea

This work presents recent Seoul National University (SNU) research findings that may critically advance the current understanding of the integral behaviour of HBU fuel rods under LOCA and associated regulatory implications. Firstly, the effect of HBU represented by pre-transient oxide and hydrogen on Zircaloy steam oxidation is discussed with regard to experimental results and the developed Zircaloy steam oxidation model (TRANOX). The rationale and potential limitations of the prevailing HBU oxidation regulatory audit method that uses the Cathcart-Pawel correlation with the corrected cladding thickness are discussed (Fig. 1).

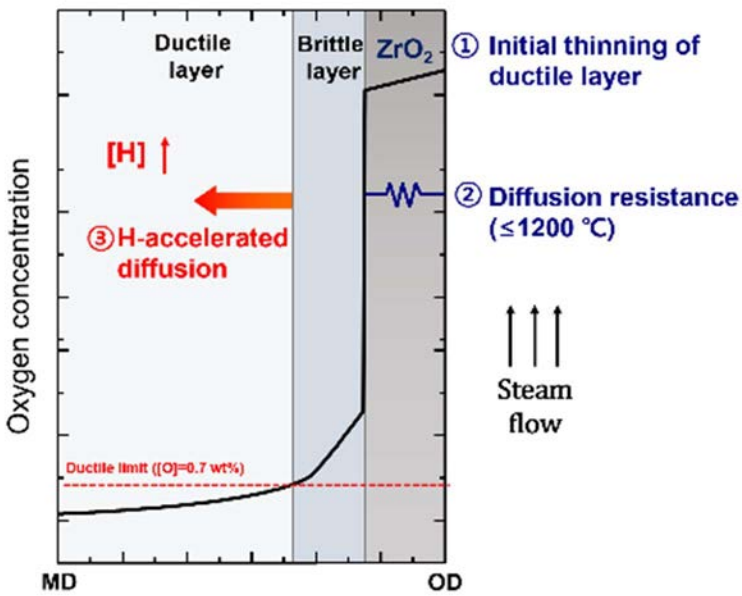



*FIG. 1. Schematic of combined effects of pre-transient oxide and hydrogen on HBU Zircaloy cladding steam oxidation (from [1], courtesy of D. Kim, Seoul National University).*

Secondly, SNU's out-of-pile integral fuel rod LOCA experiments are presented. Using induction heating for pressurized fuel rods containing surrogate pellets (i.e. whole pellet or crushed powder form, Fig. 2) that undergo the entire process of ballooning, burst, steam oxidation, and reflood quenching (Fig. 3), this study reports results in terms of the dispersed fraction of powdered surrogate pellets with respect to various conditions such as fuel rod internal pressure, and relative size of burst hole and fragmented pellet.

*(a)* *(b)*

*FIG. 2. Surrogate pellets loaded in cladding: (a) Whole pellet load and (b) Crushed powder form loaded in cladding (courtesy of Youho Lee, Seoul National University).*

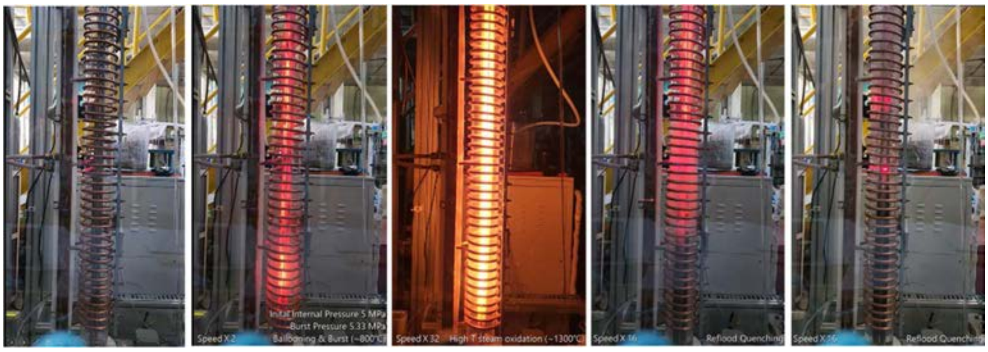



*FIG. 3. SNU's integral fuel rod LOCA experiment for FFRD phenomena investigation: ballooning-burst-steam oxidation- reflood quenching.*

Along with FFRD experiments, the integral behaviour of fuel rod is presented (Fig. 4). Based on the experimentally obtained results, preliminary regulatory implications on the audit technology of FFRD are discussed.

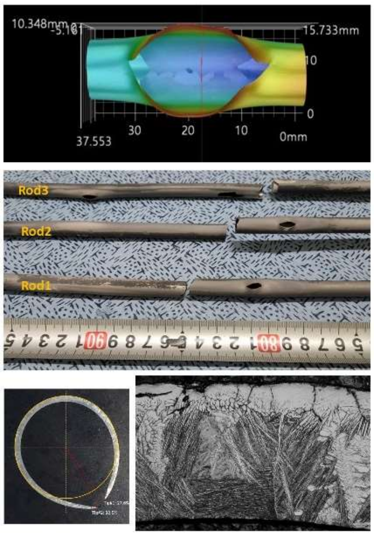



*FIG. 4. Post characterization and analysis of integral fuel rod LBLOCA experiment.*

Lastly, the origin of the difference between the post-LOCA ductility-based criteria (i.e. U.S. NRC and Korea) and fracture-based criteria (i.e. Japan) for HBU fuel is discussed with highlights on hydride-induced Zircaloy embrittlement. The MARS-GIFT integrated analysis revealed that burnup-insensitive fracture-based criteria are primarily due to delayed hydride precipitation upon quenching. The comparison of two different regulatory perspectives demonstrates both strength and potentially excess conservatism associated with the ductility-based regulation for HBU fuels.

## REFERENCES

[1] KIM, D. et al., Study of high burnup effect on steam oxidation of Zircaloy and its regulatory implications via the development of a pre-transient oxide model of TRANOX, *J. Nucl. Mater.* **567** (2022) 153801, https://doi.org/10.1016/j.jnucmat.2022.153801.

**COMBINED TREAT-LOC AND SATS LOCA EXPERIMENT PLAN: INTEGRAL LOCA EXPERIMENTS ON HIGH BURNUP FUELS**

R. ARMSTRONG, C. JENSEN, C. FOLSOM, N. WOOLSTENHULME, D. KAMERMAN, F. CAPPIA, D. WACHS
Idaho National Laboratory
United States of America

N. CAPPS, J. HARP, K. LINTON
Oak Ridge National Laboratory
United States of America

## 1. INTRODUCTION

This extended abstract serves as a summary of the U.S. Department of Energy (DOE) Advanced Fuels Campaign combined TREAT-LOC and SATS LOCA experiment plan. The full plan can be found in Ref. [1].

The TREAT loss-of-coolant (LOC) and HBU experiment series, along with the SATS HBU experiment series, are integral LOCA experiments planned under DOE's Advanced Fuels Campaign programme. These experiments aim to support burnup extension needs by addressing identified R&D priorities in order to achieve an improved understanding of HBU FFRD during LOCA events. Data generated under this experiment plan will be used to inform future R&D and model development and further validate and confirm existing models.

This experimental plan was designed to address knowledge gaps and opportunities identified through a review of existing knowledge on LOCA-induced FFRD, available publicly. The test programme employs a unique combination of in- and out-of-pile experimental approaches and state of the art facilities to provide a clear connection to the existing integral and semi-integral LOCA experiment database. The primary goal of the programme is to investigate the impact of prototypic HBU fuel/cladding thermomechanical behaviours under postulated LWR LOCA conditions that have not yet been fully studied. These conditions correspond with prototypic DEH (decay energy heat-up) and SEH (stored-energy heat-up) conditions. Importantly, the unique capability of the Transient Reactor Test (TREAT) will enable the first ever evaluation of the impact of stored energy heat-up conditions on HBU fuels. This programme, including independent experimental systems but also the development of a database to support fuel performance modelling tools, and the employment of world-leading advanced materials characterization and in-situ diagnostics to evaluate FFRD, will emphasize the development of an improved understanding of key experimental phenomena. The results generated by the programme will provide novel data to support modelling development and validation and will represent a significant advancement in evaluating prototypic conditions, in addition to informing the technical basis for LOCA-induced FFRD.

## 2. PURPOSE AND OBJECTIVES

Nuclear- and furnace-heated LOCA experiments based on a first-of-a-kind experimental approach are planned in order to address outstanding questions regarding the FFRD phenomena in HBU fuels [2–3]. The experimental methods were specifically designed to consider the impact of prototypical conditions on relevant FFRD phenomena, and with reference to the classical LOCA experiment database conducted on HBU fuels. One key aspect of the new methods pertains to the prescribed heat-up rate of the cladding and fuel temperature profiles, in recognition of the need to address SEH- and DEH-driven LOCA scenarios by applying both nuclear fission induced internal and external heating approaches. Notably, SEH conditions and the full nuclear heating of test fuel have never been explored in regard to HBU fuels. Thus, the primary objectives of the dual-facility experimental programme are to systematically achieve the following:

1. Establish world-leading integral LOCA experiment capabilities by tying back to existing databases and leveraging integral hot cell furnace and in-pile experimental testing approaches;
2. Evaluate the integral impacts of SEH and DEH conditions (depending on the HBU core design) on FFRD and transient FGR during LOCAs;
3. Measure key behavioural phenomena in situ (e.g. cladding deformation and burst dynamics, transient FGR, fuel relocation and dispersal, and cladding balloon surface temperature) to reduce the uncertainty in phenomena interdependencies (e.g. temperature profile and plenum pressure/volume), while also allowing for first-of-a-kind model development and validation;
4. Perform LOCA tests on material samples, with detailed relevant microstructural characterization both pre- and post-testing;
5. Provide expanded LOCA datasets for model development and code validation in regard to relevant HBU fuels near important burnup thresholds.

To achieve these objectives, the experimental programme will conduct experiments on fuel segments irradiated to HBU levels in commercial NPPs. The experiment goals require the use of a semi-integral LOCA furnace, SATS (located at ORNL), and the integral in-pile LOCA capability developed for TREAT at INL. These tests will be supplemented by targeted separate-effects testing and characterization using the post-irradiation examination capabilities of hot cell laboratories located at various U.S. national laboratories.

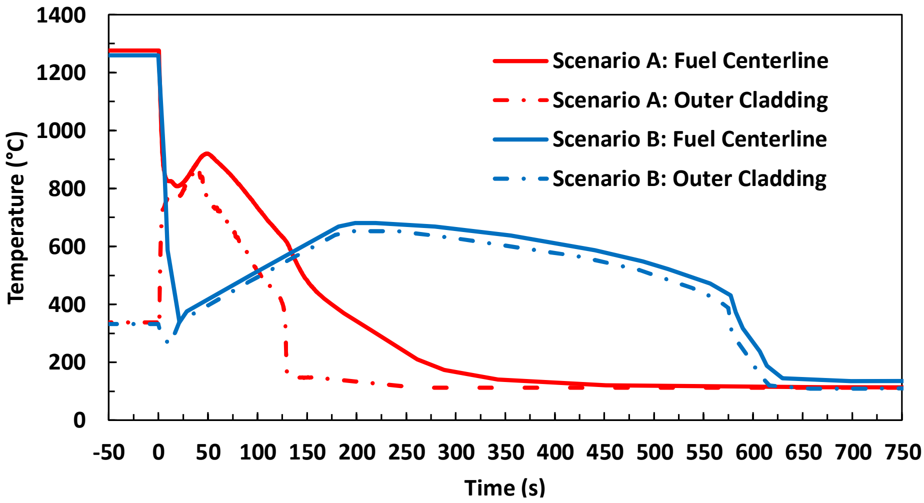
## 3. SEH VS. DEH LOCA CONDITIONS

The Advanced Fuels Campaign LOCA design teams invested heavily in ascertaining prototypic or licensing-relevant conditions for FFRD in LB-LOCAs in order to translate them for use in experiment design and test programme development. As with any experiment or model, the assumptions made to represent the true application (i.e. full-plant LOCA simulation) are of paramount importance and were thus scrutinized heavily. The general conclusions of these studies indicate that the LOCA conditions (i.e. fuel and/or cladding temperature and cladding wall pressure differential dynamics) experienced by fuel rods may vary rather widely, depending on many factors. However, certain key behaviours observed under PWR LBLOCA conditions may notably affect FFRD, including the impacts of SEH as compared to DEH conditions. These classifications are defined to highlight potentially important differences in temperature spatial/temporal evolution and rate effects.

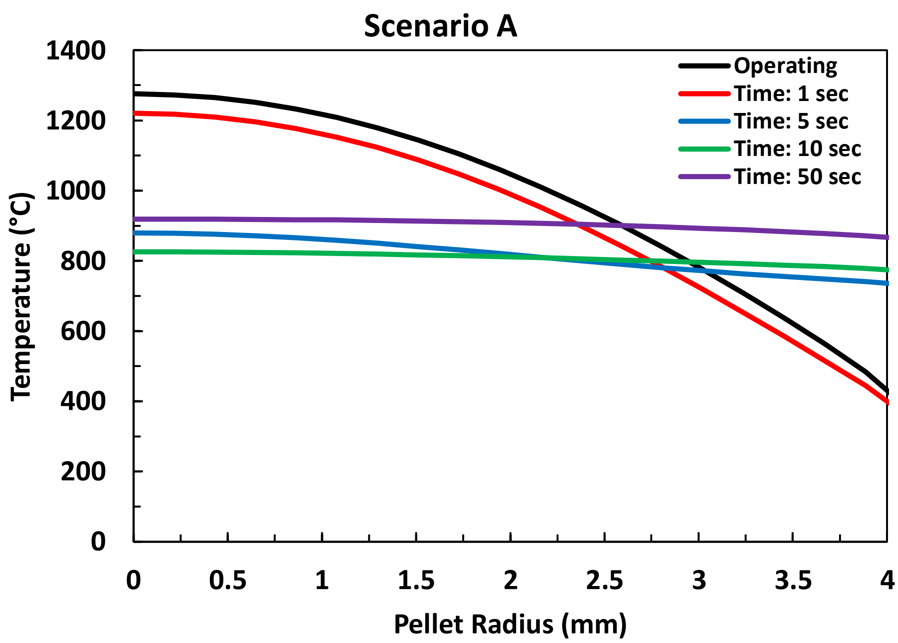
While DEH conditions always exist in LOCAs, they may be masked by much more dominant SEH effects during the first 10–20 seconds of the transient event. The level of SEH contribution appears dictated by several factors, though the dominant ones include the pre-accident linear heat rate and the fuel burnup [4] and blowdown cooling parameters. The former is also an important input to decay heat levels, which play an important role in determining the peak cladding temperatures reached during LOCAs. SEH conditions were found to generally correspond to higher peak cladding temperatures.

To illustrate the scenarios described above, Fig. 1 provides LBLOCA temperature prediction examples that specifically contrast the SEH and DEH conditions. The fuel and cladding temperature histories from the two LBLOCA simulations are included. For both scenarios, Fig.

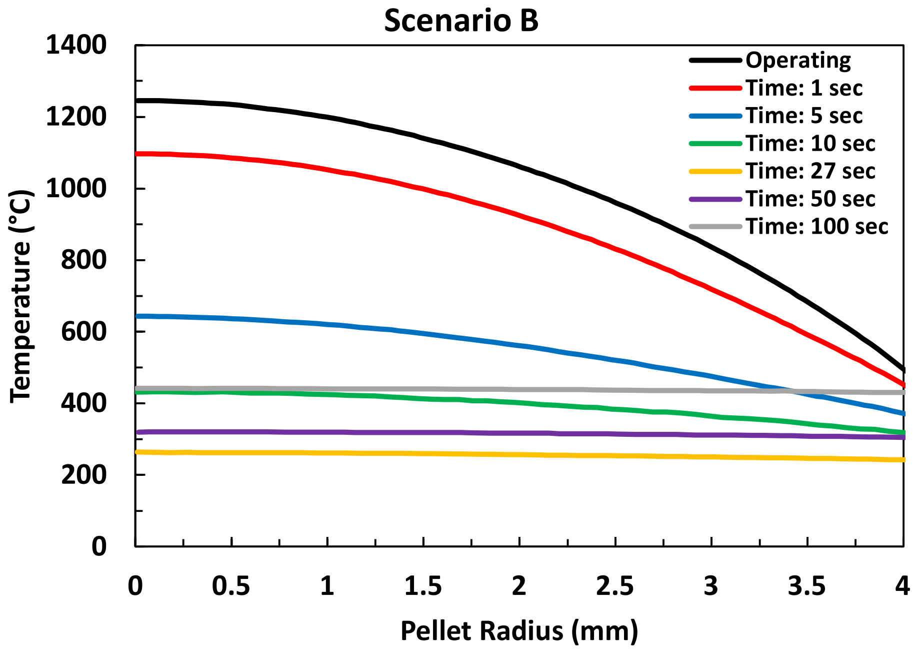
1 also shows the radial temperature profile of the fuel at various times throughout the LOCA. The Scenario A simulation results stem from a coupled BISON and TRACE LBLOCA analysis of a generic EOL Westinghouse 4-loop PWR with an HBU-cycle core design. This work was performed by INL to support development of the in-pile Transient Water Irradiation System for TREAT (TWIST) LOCA vehicle by identifying representative LOCA scenarios of interest. Scenario B stems from work performed by ORNL [5], who used RELAP5-3D and BISON to simulate a LBLOCA in an EOL Westinghouse 4-loop PWR featuring an HBU core design. In both scenarios, the fuel is operating at the same linear heat rate of ~21 kW/m prior to the LOCA, generating nearly identical radial temperature profiles through the fuel; however, the temperature response of the fuel rod following the LOCA vastly differs between the two scenarios.




*(a)*




*(b)*




*(c)*

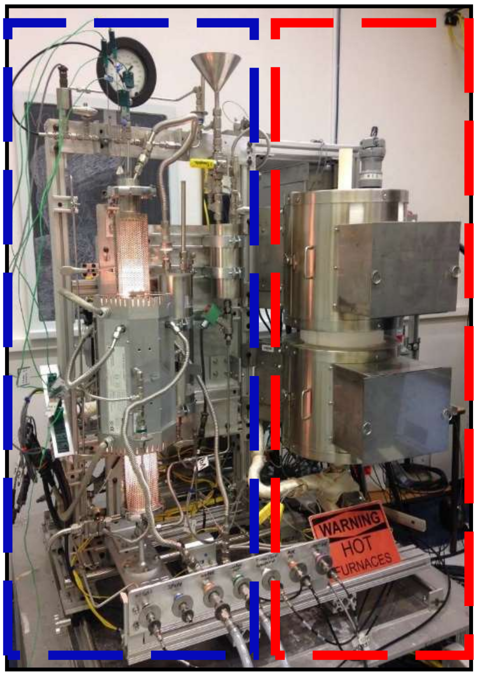
*FIG. 1. Examples of relevant LOCA heat up conditions, illustrating SEH vs. DEH (courtesy of Robert Armstrong, INL): Fuel and cladding temperatures as a function of time for Scenario A, with a strong SEH effect, and for Scenario B, with no observed SEH contribution (a) ; Radial temperature profiles in the fuel at different points in time during the transient progression for Scenario A (b) and for Scenario B (c).*

In Scenario A, the coolant in the core flashes to steam within the first few seconds after pipe rupture, greatly reducing the coolant's heat transfer capability. This results in a rapid redistribution of the fuel's stored energy, causing temperatures in the central region of the fuel to decrease at a rate of ~100°C/s, while the cladding heats up at this same rate. After this SEH-driven heat-up, core flow reversals decrease the fuel rod temperature for a period of time, followed by a second, slower, DEH-driven heat-up of the fuel rod. In Scenario B, the coolant's heat transfer capabilities directly following pipe rupture remain high enough to completely remove the stored energy from the fuel, such that the fuel rod is cooled to approximately the coolant temperature. In this scenario, there is no SEH-driven temperature peak, and the subsequent ~5°C/s temperature ramp is only due to the decay heat in the fuel.

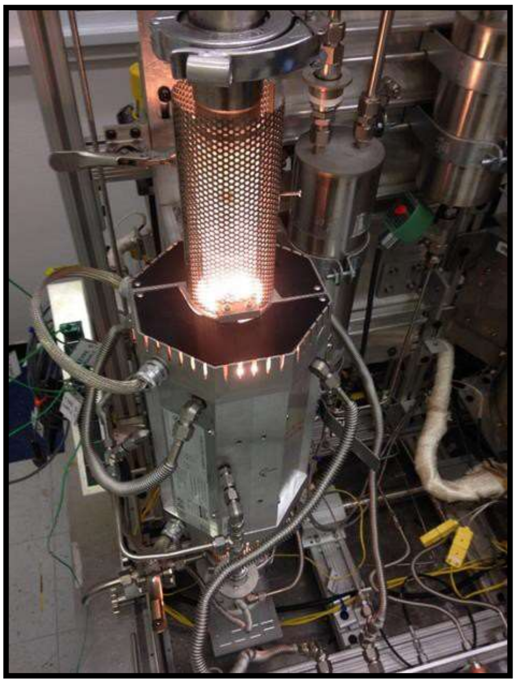
Since the potential impacts of these different conditions may be of great importance for FFRD, they are a main focus of this experimental plan, but also need to be considered in the context of specific events. Significant SEH influence may be absent in smaller break LOCA or BWR LOCA events [6]. The lack of data presently available makes the influence of heating ramp rates difficult to assess, especially when using models that lack appropriate validation and fail to account for all the necessary mechanisms.

## 4. SEVERE ACCIDENT TEST STATION

The SATS system was initially developed to conduct high temperature steam oxidation testing of unirradiated candidate accident tolerant fuel cladding concepts at ORNL [7]. Leveraging this experience, the out-of-cell SATS capabilities were replicated and modified for hot cell operation. The system has been exercised through extensive commissioning exercises and has performed multiple experiments on HBU fuels [7]. SATS consists of two modules: the DBA module and the beyond DBA module. Fig. 2 provides an overview of the two-module test station prior to insertion into the hot cell. Fig 2a shows the integral LOCA test apparatus (blue outline) and the high temperature test furnace (red outline). A zoomed-in view of the integral LOCA test apparatus is shown in Fig. 2b. The SATS LOCA integral test train can house a fuel rod segment up to 30-cm long designed to be externally heated up to 1200°C. The facility uses an infrared radiation furnace under high internal rod pressure (maximum ~20 MPa at 300°C). High pressure argon gas generates the internal pressure, and a digital output pressure transducer is used for monitoring.




(a)



(b)

*FIG. 2. (a) SATS, consisting of two modules in a single unit: outlined in blue, there is the DBA integral LOCA testing and outlined in red is the module devoted to beyond DBA high temperature testing; (b) this picture focuses on the LOCA integral test apparatus ([8] courtesy from N. Capps, INL).*

Cladding temperature control and monitoring are extremely important for understanding FFRD performance and for code validation. Analyses have shown that the cladding and fuel temperatures will differ from each other at high heating rates of 100°C/s (corresponding to SEH conditions), whereas under DEH conditions (5°C/s), fuel temperatures remain flat (~25°C across the fuel pellet). Slower heating rates consistent with DEH conditions produce greater uniformity between the cladding and fuel temperatures. Therefore, each test rodlet is equipped with up to four thermocouples: one positioned 50 cm below the midplane; one positioned 5 cm above the midplane; and two at the specimen axial midplane, 180° apart; and. Quick heat-up occurs for short samples, but an integrated, well-instrumented control system allows to avoid a temperature overshoot. Upgrades to SATS are underway to increase the cladding heating rate from the current range of 5–17°C/s to 45–60°C/s. The upgrades are also expected to reduce the axial and azimuthal temperature gradients.

## 5. TREAT AND THE TREAT LOCA DEVICE

TREAT offers the unique flexibility to serve many fuel and reactor designs by enabling a modular experiment strategy, as well as a highly agile control rod system for tailoring desired power histories for experiments [9]. Special experimental devices have been (and are currently) under development to aid in performing integral LOCA experiments and related separate-effects studies [10]. TREAT is an adiabatic, transient-shaping reactor with an air cooled core. In terms of thermal capacity, the core is limited to a total deposition energy of near 2500 MJ, with the ability to deliver that total over a period ranging from ms to several minutes. Relative to other transient reactors, TREAT is especially well suited to drive HBU low-enriched specimens to nearly any power level.

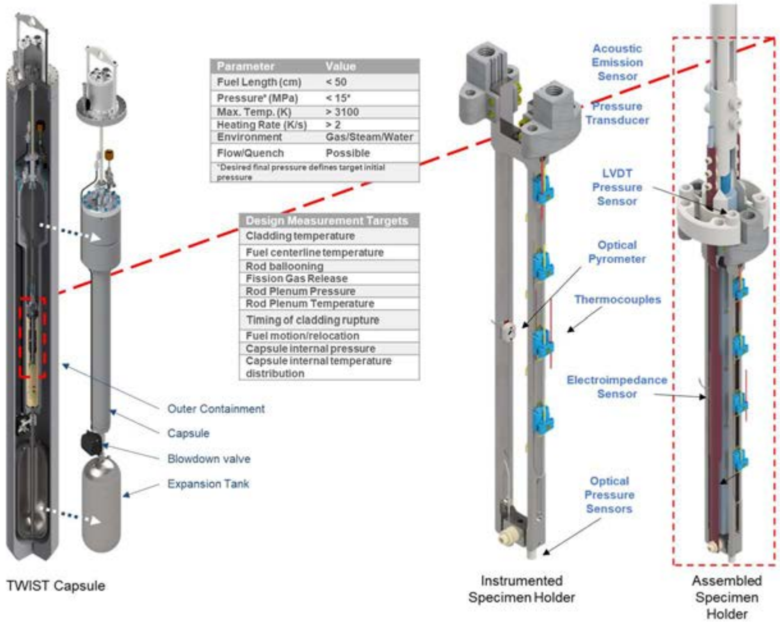



*FIG. 3. Schematic overview, key specifications, instrumentation measurement targets, and test train overview of the TREAT LOCA device, TWIST.*

With static-gas and -water capsules being routinely used in TREAT, a static-water blowdown capsule (i.e. TWIST) is being developed to enable control of relevant heat transfer boundary conditions in order to create a testing capability that specifically addresses fuel behaviours and FFRD during LOCAs. Fig. 3 gives a schematic overview of the TWIST capsule design, with tables listing its capabilities. A detailed evaluation study for a similar first generation TREAT device is given in [11]. The primary difference in that study is a smaller capsule that has since been expanded with TWIST to accommodate longer length specimens (i.e. up to 50 cm in fuelled length) and to afford improved instrumentation options.

The TREAT TWIST design experiment strategy represents a unique opportunity for LOCA testing (one not previously explored for HBU fuels), with the primary objective being to use unique instrumentation to investigate thermal histories, starting from full power conditions. Fig. 3 is a schematic representation of the nominal design conditions for a TREAT experiment, for which each segment may be tailored to suit the specific objectives. The capsule relies on careful selection of specimen power input to reach nucleate boiling conditions and prototypic linear heat rates (so as to generally not overheat beyond EOL conditions) in the specimen prior to opening an expansion tank valve, while simultaneously reducing the reactor power. This simplistic approach creates a suitable balance between heat input and heat rejection in order to control the rate of temperature change across the fuel/cladding radii. For example, the relative timing of liquid blowdown, the rate of blowdown, and the power reduction in the rod can be fine-tuned to achieve specific targeted conditions. This approach was developed to represent certain LOCA events of great importance to LWR limiting conditions and has been shown to provide the flexibility to achieve that goal. The reflood capability is not currently considered an immediate testing requirement and has thus not yet been incorporated into the TWIST capsule, but could be added with basic modifications. The capsule is designed to provide a range of boundary conditions corresponding to LWR LOCAs.

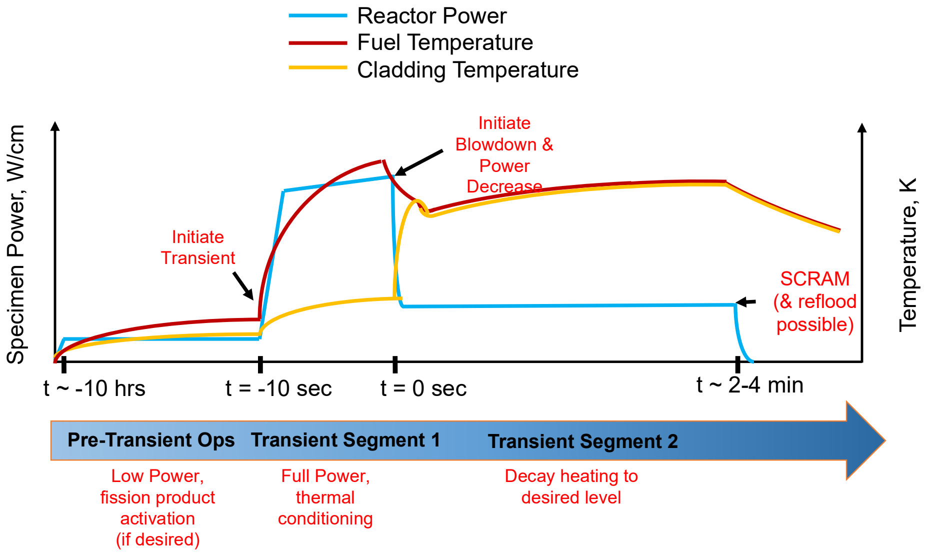



*FIG. 4. Schematic representation of the TREAT LOCA experiment sequence in the TWIST device.*

## 6. TREAT LOC-HBU AND SATS-HBU SERIES

Utilization of both TREAT and SATS represents a key opportunity and feature afforded by the LOCA test programme. More specifically, though the two experimental approaches are fully independent, they feature some overlap in terms of the available test conditions, thereby reducing uncertainties and allowing for enhanced interpretation of results in an optimal manner. Materials that feature similar burnups and are of identical origin (sister materials) or composition will, after consideration of their operational histories, be used to systematically reduce the associated uncertainties. The companion hot cell facilities are also independent and offer state of the art characterization capabilities, thus allowing for increased throughput between both institutions. Table 1 provides an overview of the planned LOC-HBU and SATS-HBU experiments.

TABLE 1. JOINT LOC-HBU AND SATS-HBU EXPERIMENT TEST MATRIX OVERVIEW. EACH LINE REPRESENTS A PAIRED EXPERIMENT CONDUCTED AT TREAT AND SATS. APPLICABLE DIFFERENCES IN TEST CONDITIONS ARE NOTED BY DUAL INPUTS

| Test ID | Seg. Burnup (GWd/t) | PCT (K) | Max. Temp. Ramp Rate (K/s) | Fuel Length (cm)* | Rod Free Volume (cc) | Purpose |
|---|---|---|---|---|---|---|
| LOC-1<br>SATS-1 | ~65 | 1173 | 5 | 25 | 15 | HBWR IFA 650.10/15 and SCIP test 36U-N05 tieback, simulate “classic” Halden/furnace condition |
| LOC-2<br>SATS-2 | ~65 | 1173 | <100<br>50 | 25 | 15 | SEH heat-up comparison to test #1 |
| LOC-3<br>SATS-3 | ~75 | 1173 | 5 | 25 | 15 | SCIP tieback with higher burnup |
| LOC-4<br>SATS-4 | ~75 | 1173 | <100<br>50 | 25 | 15 | SEH vs. DEH comparison with higher burnup (comparison test #3) |
| LOC-5<br>SATS-5 | ~75 | 1173 | <100<br>5 | 25 | 15 | Evaluate different failure conditions; target failure of the rod at a distinct point in the heatup/blowdown vs. refill phases |
| LOC-6<br>SATS-6 | ~75 | 1173 | <100<br>5 | 25 | 15 | Evaluate non-failure conditions; target conditions similar to #3–5, without rod burst (tFGR data with no burst and no burst effects) |
| LOC-7<br>SATS-7 | ~85 | 1173 | <100<br>5 | 25<br>25 | 15 | Very HBU |
| LOC-8<br>SATS-8 | ~75 | 1173 | <100<br>5 | 50<br>25 | 15 | Length effects, plenum size, axial gas communication effects, SCIP complements |
| LOC-9<br>SATS-9 | ~75 | 1173 | <100<br>5 | 50<br>25 | 2.5 | Length effects, plenum size, axial gas communication effects, SCIP complements |

* Fuel length will be limited to the distance between grid spacers, or would include a grid spacer if present in the commercial irradiation (likely applicable for 50-cm-long specimens in TREAT). Most semi-integral furnace tests have been conducted on segments with a length of nearly 30 cm.

The test matrix defines several planned experiments that will be conducted at a nominal rate of 2–3 per facility, per year, and are envisioned to provide a balance between affording a significant data generation rate to the R&D community for interpretation and still allowing the opportunity to incorporate learning from previous experiments. Ultimately, the throughput will likely be determined by the available resources (and demand). Currently, INL is expected to receive the first test materials — which will be collected from HBU commercial fuel — from ORNL and/or Byron Nuclear Generating Station in the spring and/or winter of 2023, with testing beginning in 2024. Materials available from other commercial sources and international facilities are also being considered, as the latter can provide direct linkage to R&D programme results. As the test programme progresses, new insights may impact the current testing plan.

## 7. SUMMARY

This extended abstract summarizes the DOE Advanced Fuels Campaign combined TREAT-LOC and SATS LOCA experiment plan. The full plan can be found in [1]. The experimental programme aims to support burnup extension needs by addressing identified R&D priorities in order to achieve an improved understanding of FFRD of HBU fuel during LOCA events. The data generated under this plan will be used to further validate and confirm existing models and to inform future R&D and model development. Experiments will be performed on commercially irradiated HBU fuels in the semi-integral SATS furnace at ORNL, as well as in-pile in the TREAT LOCA device at INL. This approach will systematically evaluate LOCA experiments under full nuclear heated (in-pile) vs. externally heated (furnace) conditions.

## REFERENCES

[1] JENSEN, C., ARMSTRONG, R., FOLSOM, C., WOOLSTENHULME, N., KAMERMAN, D., CAPPIA, F., WACHS, D., CAPPS, N., HARP, J., LINTON, K., Combined TREAT-LOC and SATS LOCA Experiment Plan, internal report, Idaho National Laboratory, INL/RPT-22-69915, 2022.

[2] NUCLEAR ENERGY INSTITUTE, The Economic Benefits and Challenges with Utilizing Increased Enrichment and Fuel Burnup for Light Water Reactors, Nuclear Energy Institute White Paper, NEI, (2019).

[3] BALES, M., CHUNG, A., CORSON, J., KYRIAZIDIS, L., Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup, RIL 2021-13, U.S. Nuclear Regulatory Commission, Washington, DC (2021), ADAMS Accession Number ML21313A145.

[4] TERRANI, K., et al., The effect of fuel thermal conductivity on the behavior of LWR cores during loss-of-coolant accidents, *J. Nucl. Mater.* 448 1-3 (2014) 512–519, https://doi.org/10.1016/j.jnucmat.2013.09.051.

[5] CAPPS, N., et al., Full Core LOCA Safety Analysis for a PWR Containing High Burnup Fuel, internal report, Oak Ridge National Laboratory, ORNL/TM-2020/1700, 2020.

[6] RAYNAUD, P., and PORTER, I., "Predictions of Fuel Dispersal During a LOCA," Paper no. 100026, Proceedings of Water Reactor Fuel Performance Meeting, Sendai, Japan, (2014), https://www.researchgate.net/publication/263965796_Predictions_of_Fuel_Dispersal_During_a_LOCA.

[7] SNEAD, M., et al., Severe Accident Test Station Design Document, internal report, Oak Ridge National Laboratory, ORNL/TM-2015/556, 2015.

[8] CAPPS, N., et al., Integral LOCA fragmentation test on high-burnup fuel, *Nucl. Eng. Des.* 367 (2020) 110811, https://doi.org/10.1016/j.nucengdes.2020.110811.

[9] PARRY, J., Foreword Special issue on Restarting the Transient Reactor Test Facility, *Nucl. Technol.* 205 10 (2019), https://doi.org/10.1080/00295450.2019.1650604.

[10] WOOLSTENHULME, N.E., et al., Development of Irradiation Test Devices for Transient Testing, *Nucl. Technol.* 205 10 (2019) 1251–1265, https://doi.org/10.1080/00295450.2019.1590072.

[11] WOOLSTENHULME, N.E., et al., "Thermal-hydraulic and Engineering Evaluations of New LOCA Testing Methods in TREAT," *Nucl. Technol* 207 5 (2021) 637–652, https://doi.org/10.1080/00295450.2020.1807280.

# NEA ACTIVITIES SUPPORTING IMPROVED UNDERSTANDING OF HIGH BURNUP FUEL BEHAVIOUR

M. BALES, T. IVANOVA, V. ROUYER, D. JACQUEMAIN, M. BEILMANN
OECD Nuclear Energy Agency

## 1. INTRODUCTION

The NEA's mission is to assist its member countries through international cooperation in maintaining and developing the scientific, technological and legal bases required for a use of nuclear energy for peaceful purposes under safe, environmentally sound and economical conditions. Two key mechanisms for facilitating this international cooperation are NEA's expert communities and joint projects. On the subject of nuclear fuel behaviour, key expert communities include the WGFS under the Committee on the Safety of Nuclear Installations (CSNI) and the EGRFP under the Nuclear Science Committee (NSC). Joint projects that have historically and are currently providing insight on nuclear fuel behaviour include the HRP [1], the FIDES-II programme [2] and the Studsvik Cladding Integrity Project [3].

## 2. ACTIVITIES OF THE NEA EXPERT COMMUNITIES

### 2.1 Working Group on Fuel Safety (WGFS)

The main mission of the WGFS is to advance the current understanding and address cross-cutting issues related to fuel behaviour in accident conditions. The WGFS aims to facilitate international convergence in fuel safety issues, including experimental approaches as well as interpretation and use of experimental data, and other relevant information. Specifically, on the subject of HBU fuel behaviour, one recently publication and one ongoing activity can be highlighted.

The recently published State-of-the-art Report on Nuclear Fuel Behavior under RIA Conditions, NEA 7575 [4] is an update of the 2010 report (NEA 6847 [5]) considering significant expansion (revisions, corrections and supplements to data and information) of the knowledge base on fuel behaviour under RIA. The report discusses the implications of HBU microstructure on RIA performance and adds two new chapters on reviewing regulatory acceptance criteria and on applicability and transferability of test data.

The recently concluded task under WGFS on Fuel Safety Implications of Extended Enrichment and High Reactivity/High Suppression Core Designs will conclude with the publication of a status report. The status report aims to consolidate information regarding fuel enrichment limits and utilize WGFS expertise to evaluate possible fuel safety implications of extended enrichment (5–8%) and high reactivity/high suppression core designs. The report will also provide recommendations for collaborative analytical and experimental research on this topic. The task group recognized that the recent interest within the nuclear industry to extend enrichment beyond 5% is closely motivated by the desire to increase burnup. However, the task group examined the fuel safety implications of extended enrichment independent of coincident fuel safety implications of extended burnup. Nevertheless, it is interesting to note a few of the task group's findings. First, they recognized that starting with increased enrichment means rods have greater reactivity at HBU, which may challenge failure limits in some accident scenarios (e.g. RIA). Therefore, the task group's final recommendations included that additional tests on behaviour of extended enrichment fuel during power ramps or RIA would be useful. Second, they recognized that core designs with extended enrichment will likely require increased use of burnable absorbers. Burnable absorbers often have negative impact on fuel performance, specifically, lower thermal conductivity of Gd-fuel and increased rod internal pressure due to He release from $ZrB_2$. Therefore, the task group's final recommendations also included that additional experiments on Gd doped and extended enrichment fuel under DBA conditions, particularly at HBU would be useful.

### 2.2 Expert Group on Reactor Fuel Performance (EGRFP)

The EGRFP performs tasks specifically associated with fuel performance aspects, including present and future nuclear power systems, focusing on normal operating conditions. One mechanism for deepening their understanding of fuel performance is code benchmarks. Further, a key activity of the EGRFP is the identification and preservation of appropriate experimental data.

In the area of code benchmarks, the EGRFP recently concluded a benchmark on PCMI [6] and recently launched a benchmark on burst FGR [7]**.** The objectives of the burst FGR benchmark include:

- Improving understanding of burst FGR amongst NEA member organisations;
- Facilitating improvements of existing code models to address burst FGR, and the development of new models for fuel performance;
- Facilitating reduction in modelling uncertainties of FGR predictions and hence improvement in their accuracy, margin increase to fuel performance limits (e.g. rod internal pressure), and also an increase in plant flexibility.

On the subject of data preservation, the EGRFP is responsible for the IFPE [8]; a data collection on nuclear fuel performance experiments devoted to code development and validation. This public domain data collection is developed jointly by the NEA and the IAEA. In recent years, significant investment has been made to facilitate easier electronic distribution in a single package and increase usability of the data collection through a software tool called DATIF, the DATabase for IFpe [9]. Considering HBU fuel behaviour specifically, IPFE includes a few datasets >60 GWd/tU.

## 3. JOINT PROJECTS

### 3.1 Halden Reactor Project (HRP)

The HRP ran for over 60 years and in that time ran a large number of experiments on HBU fuel. This legacy data has high value for code validation to HBU levels. Following the reactor closure in 2018, the HRP approved transfer of data for preservation and administration. The NEA is currently seeking to implement an integrated, cost-effective approach for managing key data sets for fuel safety in the future, considering developments made for IFPE and DATIF.

### 3.2 Framework Irradiation Experiments (FIDES-II)

The NEA FIDES-II project supports the fuel and material experimental needs of all stakeholders (nuclear safety regulatory bodies, technical support organisations, research institutions and industry) and preserves experimental knowledge for future generations. FIDES-II allows a connection for research facilities through joint experimental programmes, forming a global network to perform high priority experiments. One joint experimental programme in particular, the HERA joint experimental project, is focused on improving understanding of HBU fuel behaviour. The HERA experiments are being conducted at the TREAT reactor (INL) and NSRR reactor (operated by JAEA). The objectives of the experiments are to (1) quantify the impact of pulse width on fuel performance, (2) investigate performance of modern HBU fuel in RIA transients at representative pulse widths, (3) identify available cladding failure margin in modern claddings and (4) identify any safety impacts from operation of extending the burnup of $UO_2$ fuels.

### 3.3 Studsvik Cladding Integrity Project (SCIP)

The SCIP has been conducted, so far, in four phases, with each phase having a separate focus. Common to all phases are advanced out-of-pile experiments on irradiated nuclear fuels and claddings in its laboratories, complimented by advanced microscopy and other characterization technologies. SCIP-III (2014-2019) focused on LOCA issues, in particular on FFRD. The project assessed the influence of different parameters such as burnup, cladding strain, temperature, rod internal pressure but also free volume and of microstructural effects. SCIP-IV (2019-2024) includes studies on the behaviour of fuel during a LOCA as well as fuel behaviour under dry storage conditions. Data from both phase three and four is key to understanding FFRD behaviour, which is critical to understanding the behaviour and boundaries of acceptable performance of HBU fuel under LOCA conditions.

## 4. CONCLUSION

The expert communities and joint projects organized under the auspices of the NEA have produced, and continue to produce, key contributions on the subject of nuclear fuel behaviour. As our members increase their focus on the development of improved understanding of HBU fuel behaviour, NEA's expert communities and joint projects have proved to be key mechanisms to advance their knowledge and understanding.

## REFERENCES


[1] OECD NUCLEAR ENERGY AGENCY, Halden Reactor Project - Fuels and Material (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_24970/halden-reactor-project-fuels-and-material.

[2] OECD NUCLEAR ENERGY AGENCY, Second Framework for Irradiation Experiments (FIDES-II) (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_70867/second-framework-for-irradiation-experiments-fides-ii.

[3] OECD NUCLEAR ENERGY AGENCY, Studsvik Cladding Integrity Project (SCIP) (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_25445/studsvik-cladding-integrity-project-scip.

[4] OECD NUCLEAR ENERGY AGENCY, State-of-the-art Report on Nuclear Fuel Behaviour Under Reactivity-initiated Accident Conditions (RIA SOAR) (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_74371/state-of-the-art-report-on-nuclear-fuel-behaviour-under-reactivity-initiated-accident-conditions-ria-soar.

[5] OECD NUCLEAR ENERGY AGENCY, Nuclear Fuel Behaviour under Reactivity-initiated Accident (RIA) Conditions (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_14526/nuclear-fuel-behaviour-under-reactivity-initiated-accident-ria-conditions?details=true.

[6] OECD NUCLEAR ENERGY AGENCY, Pellet-Cladding Mechanical Interaction (PCMI) Benchmark (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_15168/pellet-cladding-mechanical-interaction-pcmi-benchmark.

[7] OECD NUCLEAR ENERGY AGENCY, Burst Fission Gas Release (FGR) Benchmark of the Expert Group on Reactor Fuel Performance (EGRFP) (oecd-nea.org),

https://www.oecd-nea.org/jcms/pl_70791/burst-fission-gas-release-fgr-benchmark-of-the-expert-group-on-reactor-fuel-performance-egrfp.
[8] OECD NUCLEAR ENERGY AGENCY, International Fuel Performance Experiments (IFPE) database (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_36358/international-fuel-performance-experiments-ifpe-database.
[9] OECD NUCLEAR ENERGY AGENCY, DATabase for IFpe (DATIF) (oecd-nea.org), https://www.oecd-nea.org/jcms/pl_33943/database-for-ifpe-datif.

# RECENT ADVANCED POST-IRRADIATION EXAMINATIONS OF HIGH BURNUP $UO_2$ FUELS AT IDAHO NATIONAL LABORATORY

F. CAPPIA, D. FRAZER, T. R. PAVLOV, K. WRIGHT, K. BAWANE, B. KOMBAIAH, F. TENG
Idaho National Laboratory, Characterization and Advanced Post Irradiation Examination Division,
United States of America

R. DAUM
Electric Power Institute
United States of America

C. JENSEN, D. WACHS
Idaho National Laboratory, Nuclear Fuels and Materials Division
United States of America

## 1. INTRODUCTION

The changes in physical and chemical properties that nuclear fuel endures under extended in-reactor service have been identified as one of the critical parameters contributing to the fuel fine fragmentation [1–2], one of the biggest outstanding technical issues that are a knowledge gap and performance concern for burnup extension [3].

The development of a mechanistic criterion for fuel fragmentation and pulverization is the ultimate goal in a science-based approach to fuel performance. With the increased complexity of the physics-based models and their improved ability to spatially resolve chemical and microstructural variations in the fuel [4–5], it is imperative that experimental techniques can obtain information on changes in physical and chemical properties with the same spatial accuracy to provide a robust feedback process.

New analytical material science techniques have been made available in the last two decades to study highly radioactive materials, such as HBU fuels. The application of techniques such as micro-mechanical testing and spatially resolved thermal property measurement techniques now enable the study of these properties of HBU fuels as a function of the fuel radius, producing the necessary inputs to validate meso-scale physics-based models. In this study, application of micro-tensile testing [6] and local thermo-reflectance method [7] are particularly demonstrated on a HBU $UO_2$ fuel. These techniques, combined with traditional post-irradiation examination and advanced microscopy, create a powerful toolbox to investigate the property degradation of HBU fuels.

## 2. MATERIALS AND EXPERIMENTAL TECHNIQUES

The sample used in this work belongs to a HBU rod irradiated between 1997 and 2004 for four eighteen-months cycles to provide M5® cladding performance data at HBU [8]. The pellet average burnup was ≈80 GWd/tU calculated from the EPMA data [9].

EBSD orientation maps were collected using the EDAX Hikari Super EBSD detector of the Plasma Focused Ion Beam using a step of 0.1 μm and processed using the EDAX TSL OIM™ 8.0 software. The EBSD mapping along the fuel radius revealed three regions in the fuel pellet: a first one from the fuel centre up to approximately half of the radius where the grains were internally subdivided with low angle grain boundaries. A second region from the mid radius up to approximately 85–90% of the radius exhibited grains similar to the as-fabricated ones and the peripheral region showed the progressive formation of the HBS [9].

Micro-tensile bars 2 µm x 2 µm x 10 µm in size were prepared from grains with target orientation {111} and {100}, with and without internal sub grain boundaries, using the Plasma Focused Ion Beam. Additional details of the sample preparation procedure can be found in Ref. [10]. Microtensile tests were performed using a push-to-pull device. Values of the fracture strength were calculated by dividing the measured force at fracture by the area of the bars measured during fabrication.

For the local thermal conductivity and diffusivity measurements, the thermal conductivity microscope installed in a shielded glovebox at the Irradiated Materials Characterization Laboratory at INL was employed. A complete description of the system and measuring protocol can be found in Ref. [7].

## 3. RESULTS AND DISCUSSION

Figure 1 shows the summary of the experimental fracture strength values obtained from the sample in comparison with the literature data from non-irradiated and low-burnup samples [11, 12]. This initial dataset shows that at HBU the intergranular fracture strength is comparable to the fracture strength at lower burnup, while the intragranular fracture strength is significantly degraded compared to the low burnup counterpart. The difference could be due to the extensive accumulation of intragranular dislocation observed in the sample [9] that weakened the interior of the grains at HBU.

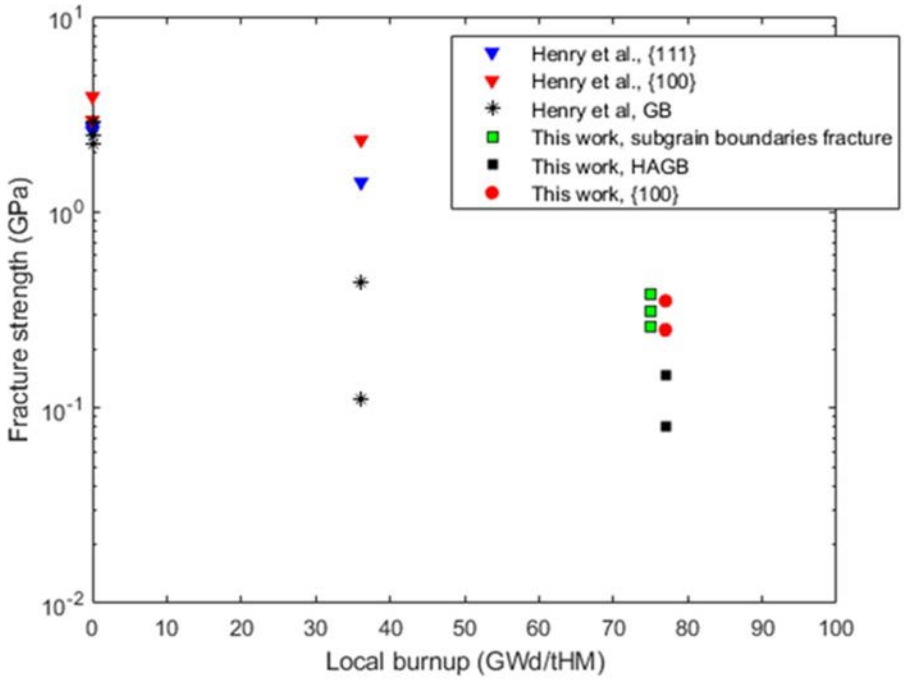



*FIG. 1. Local fracture strength as a function of local burnup.*

Figure 2 shows the radial variations of thermal conductivity obtained either directly from the thermal conductivity microscope's measurements or derived by the measured thermal diffusivity measurements. It is shown that the thermal conductivity remains approximately constant from the pellet centre to the pellet mid-radius followed by a gradual decrease in these properties towards the pellet periphery. This trend is consistent with the measured inventory of fission gases such as Xe measured by EPMA in the sample [9], which precipitate in nanometric bubbles with a limited quantity present in matrix solution in a three vacancy Schottky cluster. Such bubbles and point defects are effective phonon scattering sites and contribute to the degradation of thermal conductivity. Furthermore, the accelerated decrease in thermal diffusivity and conductivity towards the pellet periphery is in line with the significant increase in porosity and burnup [9]. These measurements, coupled with the microstructural characterization performed on the sample in [9], can be used to validate meso-scale thermal conductivity models such as the one reported in Ref. [5].

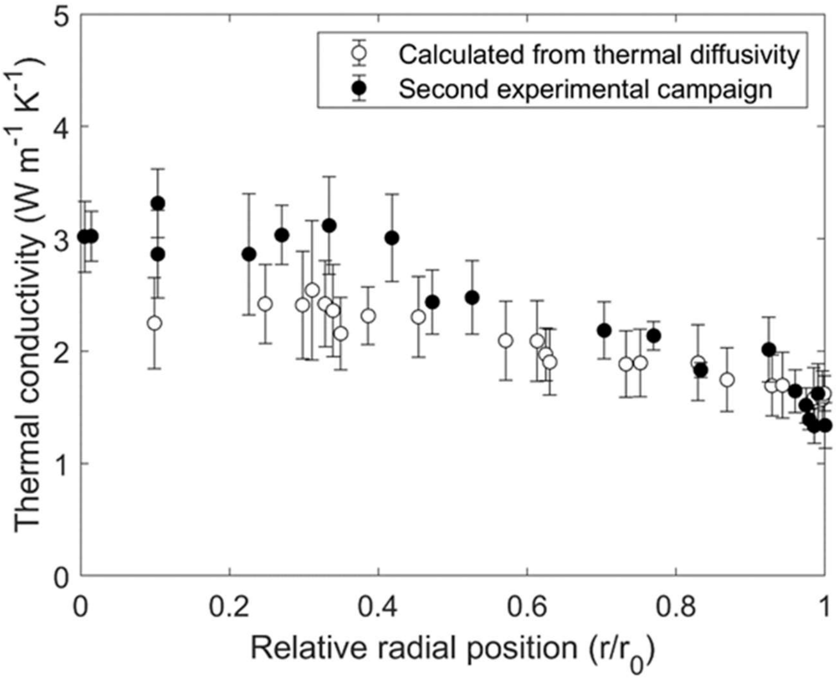



*FIG. 2. Radial profile of the measured thermal diffusivity of the sample.*

## 4. CONCLUSION AND PERSPECTIVES

The current study demonstrated the deployment of state of the art local characterization techniques to obtain new experimental data on HBU fuels, with spatial resolution previously not accessible. The current approach, coupled with extensive microstructural characterization, paves the way for exploring the influence of irradiation-induced modifications on fundamental material properties such as mechanical and thermal properties needed to develop and validate physics-based fuel performance modelling. Systematic synergy between these experiments and modelling efforts are expected to be employed in the future to address the knowledge gaps for burnup extension of nuclear fuels.

## REFERENCES


[1] TURNBULL, J.A., YAGNIK, S.K., HIRAI, M., STAICU, D.M., WALKER, C.T., An Assessment of the Fuel Pulverization Threshold During LOCA-Type Temperature Transients, *Nucl. Sci. Eng.* 179 (2015) 477–485, 10.13182/NSE14-20.

[2] CAPPS, N., JENSEN, C., CAPPIA, F., HARP, J., TERRANI, K., WOOLSTENHULME, N., WACHS, D., A Critical Review of High Burnup Fuel Fragmentation, Relocation, and Dispersal under Loss-Of-Coolant Accident Conditions, *J. Nucl. Mater.* 546 (2021) 152750, https://doi.org/10.1016/j.jnucmat.2020.152750.

[3] GEELHOOD, K.J., Fuel Performance Considerations and Data Needs for Burnup above 62 GWd/MTU, internal report, Pacific Northwest National Laboratory, PNNL-29368, 2019.

[4] JIANG, W., HU, T., AAGESEN, L.K., BISWAS, S., GAMBLE, K.A., A phase-field model of quasi-brittle fracture for pressurized cracks: Application to $UO_2$ high-burnup microstructure fragmentation, *Theor. Appl. Fract. Mech.* 119 (2022) 103348, https://doi.org/10.1016/j.tafmec.2022.103348.

[5] BAI, X.M., TONKS, M.R., ZHANG, Y., HALES, J.D., Multiscale modeling of thermal conductivity of high burnup structures in $UO_2$ fuels, *J. Nucl. Mater.* 470 (2016) 208–215, https://doi.org/10.1016/j.jnucmat.2015.12.028.

[6] HOWARD, C., JUDGE, C.D., HOSEMANN, P., Applying a new push-to-pull micro-tensile testing technique to evaluate the mechanical properties of high dose Inconel X-750, *Mater. Sci. Eng.* 748 (2019) 396–406, https://doi.org/10.1016/j.msea.2019.01.113.

[7] PAVLOV, T.R., MIDDLEMAS, S.C., MILLER, B.D., CAPPIA, F., COLE, J.I., GIGLIO, J., Understanding the local thermal conductivity evolution of neutron irradiated U3Si2 dispersion fuel via state-of-the-art thermo-reflectance measurements, *J. Nucl. Mater.* 557 (2021) 153280, https://doi.org/10.1016/j.jnucmat.2021.153280.

[8] GARNER, G.L., HILTON, B.A., MADER, E., “Performance of Alloy M5® Cladding and Structure”, Proc. 2007 Int. LWR Fuel Performance Meeting, San Francisco, 2007.

[9] CAPPIA, F., WRIGHT, K., FRAZER, D., BAWANE, K., KOMBAIAH, B., WILLIAMS, W., FINKELDEI, S., TENG, F., GIGLIO, J., CINBIZ, M.N., HILTON, B., STRUMPELL, J., DAUM, R., YUEH, K., JENSEN, C., WACHS, D., Detailed characterization of a PWR fuel rod at high burnup in support of LOCA testing, *J. Nucl. Mater.* 569 (2022) 153881, https://doi.org/10.1016/j.jnucmat.2022.153881.

[10] CAPPIA, F., FRAZER, D., TENG, F., BAWANE, K., JENSEN, C., WACHS, D., YUEH, K., DAUM, R., “Fracture properties and advanced characterization of high burnup fuel relevant to fragmentation during accident conditions”, Proc. Int Conf. TopFuel2021, Santander (Spain), 24–28 October (2021).

[11] HENRY, R., ZACHARIE-AUBRUN, I., BLAY, T., CHALAL, S., GATT, J.M., LANGLOIS, C., MEILLE, S., Fracture properties of an irradiated PWR $UO_2$ fuel evaluated by micro-cantilever bending tests, *J. Nucl. Mater.* 538 (2020) 152209, https://doi.org/10.1016/j.jnucmat.2020.152209.

[12] HENRY, R., ZACHARIE-AUBRUN, I., BLAY, T., TARISIEN, N., CHALAL, S., ILTIS, X., GATT, J.M., LANGLOIS, C., MEILLE, S., Irradiation effects on the fracture properties of $UO_2$ fuels studied by micro-mechanical testing, *J. Nucl. Mater.* 536 (2020) 152179, https://doi.org/10.1016/j.jnucmat.2020.152179.

TECHNOLOGY DEVELOPERS' EXPERIENCE

(Session IV)

**Chairperson**

**J. ZHANG**

Belgium

# FUEL MANAGEMENT STRATEGY AND PLAN FOR INCREASING FUEL BURNUP

Y. JIAO
Nuclear Power Institute of China
China

## 1. NUCLEAR POWER IN CHINA

The Chinese government announced at the 75th United Nations General Assembly that China will reach the carbon peak by 2030 and realize carbon neutrality by 2060. In order to promote a green transformation of economic and social development, improve the industrial structure, and formulate plans for carbon peak in the energy, petrochemicals, transportation and other sectors, and accelerate the building of a clean, low-carbon, safe and efficient energy system, strictly control the consumption of fossil energy, and actively develop non-fossil energy, nuclear energy is the main option to provide a stable and clean base-load energy. Taking Gen-II+ million-level reactor units as example, during the whole industrial chain and lifespan, the emission is 11.9g-$CO_2$, eq/(kW·h); without considering irradiated fuel reprocessing and disposal, the emission is 6.2g-$CO_2$, eq/(kW·h).

In China, 54 NPPs are under operation by June 2022, total installed capacity is 55.78 GWe, ranking third in the world. While 23 NPPs are under construction or approved, ranking first in the world.

The total NPPs under operation or construction are the second in the world. In 2022, 10 NPPs were approved. 34 NPPs in China received a full score in the WANO composite index 2021. Full score NPPs account for 70% of the total Chinese NPPs.

It is estimated that the target for nuclear power development in China is to generate around 10% of total energy from installed capacity by 2035 and 15–20% of total energy by 2050.

## 2. FUEL MANAGEMENT IN CHINA

The major refuelling cycle length of an NPP in China is around 18 months, including the combination of two enrichments, 4.45% and 4.95%, and an increasing number of refuelling fresh fuel assemblies compared with a12-mouth fuel management strategy.

The objective of fuel management is to prolong the refuelling period and increase fuel discharged burnup, so that the economy of the NPP and the availability of fuel can be increased.

Currently, most of the Gen- II+ reactor units in China adopt an 18-month refuelling strategy. While the fuel economy of this strategy is relatively low. Certain Gen-II+ also adopt the mixed enrichment and 18-month refuelling strategy, with higher average fuel assembly discharge burnup and better fuel economy. In addition, some of Gen-II+ units adopt a long and short alternate 18-month refuelling strategy. As for WWERs, AP1000 and CNP1000, all of them adopt an 18-month refuelling strategy.

To summarize, the major refuelling cycle length of an NPP in China is around 18 months, while the fuel enrichment is low than 5%. The major fuel assembly burnup limit is 52000MWd/tU, except for AP1000 and WWER. Nevertheless, fuel assembly discharged burnup can reach to 57000MWd/tU in some fuel management strategies.

## 3. 24-MONTH FUEL MANAGEMENT

Some Chinese NPPs are considering adopting a 24-month refuelling strategy. Compared with 18-month refuelling, increasing the cycle length to 24- months brings the benefits of increase power generation and the capacity factor of NPPs, reducing times of shutdown and refuelling overhauls, the cost of major repairs, and radiation exposure of workers. Additionally, adjusting refuelling overhauls to a certain time, is conducive to the rational arrangement of the unit overhaul and increasing the capacity of NPPs.

When enrichment is less than 5%, the number of fresh fuel assemblies (FAs) is increased to enlarge the cycle length, thus, the average discharged burnup decreases, and the fuel economy deteriorates. When enrichment exceeds 5% and the FA burnup limit increases, the number of fresh FA can be effectively reduced, and the fuel economy can be improved at a similar cycle length.

According to the analysis, for the WWER and AP1000 units, there is no need to increase fuel burnup limit when the fuel enrichment is less than 5%. For CNP650 units with 121 fuel assembly cores, the fuel economy is acceptable when the fuel enrichment is less than 5% and fuel burnup limit does not exceed 55000MWd/tU. While for M310 units with 157 fuel assembly cores and CNP1000 with 177 fuel assembly cores, the fuel economy is not ideal when the fuel enrichment is less than 5% and fuel burnup limit does not exceed 55000MWd/tU.

In order to increase the fuel economy of 24-month refuelling strategy, fuel enrichment exceeding 5% and fuel burnup limit exceeding 55000MWd/tU are necessary for M310 and CNP1000 units, which adopt the more prevalent fuel assembly nowadays.

## 4. PLAN FOR INCREASING BURNUP

Currently, approved burnup limit of the widely used fuel assembly is 52000MWd/tU. In 2019, four lead fuel assemblies were reloaded for higher burnup irradiation tests, and the discharged burnup is close to 57000MWd/tU. Before and after the irradiation tests, full poolside inspection was conducted, including for deformation of the FA and oxide film thickness. To increase batched FA burnup limits is still underway. Some ATF types currently under research will increase FA burnup to above 62000MWd/tU, and can be applicated by 2030.

## 5. CONCLUSION

To sum up, the major refuelling cycle length of NPP is around 18 months and the fuel burnup limit is around 52000MWd/tU in China.

To further improve fuel economy, some nuclear power units are considering adopting a 24-month refuelling strategy and increasing the fuel burnup limit. While the regulatory body's safety supervision provides a challenge to increasing the fuel burnup limit. The irradiation test plan is being designed to progressively increase fuel burnup limit.

# ENUSA EXPERIENCE ON HIGH BURNUP DEMONSTRATION PROGRAMMES

N. DONCEL, J.M. ALONSO
ENUSA Industrias Avanzadas S.A. SME
Spain

## 1. BACKGROUND

In Spain, nuclear energy has represented approximately 20% of the total production of electricity in recent years. Spanish companies have participated in the development of the Spanish nuclear programme since the beginning. The Spanish nuclear industry participation also includes international R&D projects related to HBU fuel for advanced nuclear reactors.

The design, manufacture, and supply of fuel to Spanish and international NPPs is made by the public capital company ENUSA (Industrias Avanzadas S.A. SME). ENUSA's aim is to develop innovative nuclear and environmental solutions, which contribute to the sustainable progress of society focused on activities of the nuclear fuel cycle, as follows:

- Management of the fuel supply under the criteria of security and flexibility;
- Engineering services in all aspects of the lifetime of the nuclear fuel (design, operation, transport and dry storage);
- Manufacturing nuclear fuel assemblies for domestic and foreign NPPs for both PWR and VVER as well as BWR.
- On-site services during refuelling programmes (including handling, inspection and repair campaigns), reception of fresh fuel supply, handling services of irradiate fuel and process supervision during the reload (including inspection, repair, characterization, and cleaning);
- Transport of nuclear and other radioactive material (through the subsidiary ETSA).

ENUSA also develops technological capabilities for the sale of equipment for manufacturing fresh fuel and for inspection of irradiated fuel. On the engineering side, ENUSA is responsible for the product design as well as nuclear and safety analysis for Spanish NPPs. The reload activities (design, follow up and support operations), together with the design of spent fuel management methodologies and new products and technology introduction (R&D) are also part of our portfolio for the European PWR and BWR market.

### 1.1 Spanish National Integrated Energy and Climate Plan (ENCP)

The ENCP for 2021–2030 is mandated by the European Union in order to meet EU overall GHG emissions targets. The ENCP addresses the five dimensions of the EU targets: de-carbonization, energy security, energy efficiency, innovation, internal energy markets and research, and competitiveness.

The government released a second version of the plan in September 2020. According to that version, the nuclear power phase-out in Spain is expected to be achieved by 2035. As of November 2022, seven nuclear reactors were in operation, the majority owned and managed by Iberdrola and ENDESA. A sharp decline of the nuclear power capacity is expected, from 7.1 GW in 2020 to 3 GW by 2030.

## 2. HIGH BURNUP DEMONSTRATION PROGRAMMES IN SPAIN

The Spanish nuclear industry has conducted joint experimental programmes with the Japanese PWR industry since the early 1990's to address fuel performance issues such as cladding corrosion resistance, dimensional stability, and FGR (Fission Gas Release).

These efforts have produced substantial amounts of valuable information on the in-reactor performance of fuel materials for current as well as potential future fuel designs.

Characterized fuel rods composed of different cladding materials were irradiated in the Spanish PWR Vandellós II NPP for up to five irradiation cycles achieving values up to 75 MWd/kgU, rod average burnup. An extensive post-irradiation examinations (PIE) programme, both on-site and in hot cells, was conducted over this higher burnup fuel as well as other related data on similar fuel rods at lower burnup thus supporting and strengthening the conclusions.

### 2.1. Vandellós Segmented Fuel Rod Irradiation Program and Extension [2]

The four major phases for the Vandellós Segmented Fuel Rod Irradiation Program were:

(I) Fuel assembly design, licensing, fabrication, and characterization;
(II) Fuel assembly irradiation and on-site examination;
(III) Ramp testing and hot cell PIE;
(IV) Extension of irradiation up to HBU, together with PIE.

Phase I was performed through a collaboration programme between ENDESA, S.A. representing Vandellós II, and ENUSA; with the collaboration of Westinghouse; and the Japanese organizations KEPCO (Kansai Electric Power Co., Inc.) representing other Japanese PWR utilities, and MHI (Mitsubishi Heavy Industries, Ltd.). Phases II and III were conducted under the sponsorship of MITI (Japanese Ministry of International Trade and Industry) by NUPEC (Nuclear Power Engineering Corporation).

The objective of this programme (1991–2001) was to obtain performance data for advanced fuel materials at high assembly burnup (55 MWd/kgU). Thirty two segmented rods were fabricated and loaded into four fuel assemblies to supply a range of fuel pellet and cladding combinations. Each segmented rod consisted of seven segments, jointed by intermediate end plugs. The main design features of the segments were:

(a) Composition of the cladding alloy (MDA, ZIRLO and Low-tin Zircaloy-4);
(b) Cladding texture (conventional and texture controlled claddings);
(c) Pellet grain size (conventional and large grain pellets).

ENUSA manufactured four segment fuel assemblies with eight segmented fuel rods installed per assembly. The segmented fuel rod consisted of seven short segments, designed for extended irradiation and/or power ramp test in the test reactor.

The four cycles of irradiation in Vandellós II were completed in March 1999 up to 53 MWd/kgU (assembly average burnup). On-site inspections were performed after each irradiation cycle, including oxide thickness measurement by Eddy current test and underwater visual inspection. Several segmented rods were withdrawn from fuel assemblies after their second, third and fourth-cycle irradiation, and sent to Studsvik Nuclear AB in Sweden for hot cell examination and ramp testing.

There are several papers presented in different TopFuel meetings which provided a large number of results from on-site examinations, as well as hot cell PIE and ramp testing.

Power ramp tests to simulate transient events were conducted and the test results showed the relationship with the terminal power and power increment during ramp tests with a higher failure threshold than the earlier database, and there was no failure beyond approximately 55 MWd/kgU (peak pellet). The ramp test with a large power increase caused several small surface cracks on the HBU low tin Zircaloy-4 cladding, although the segment did not fail. This information was very valuable for completing the knowledge and modelling of the pellet–cladding interaction (PCI).

### 2.2. High burnup irradiation behaviour in the hot cell [3]

After the four cycles of irradiation and on-site inspection, several segment rods were removed and transported to Studsvik. There, designed facilities, installed in the R2 reactor as self-convection boiling water capsules (BOCA), were used to achieve higher burnup. The BOCA facilities were able to replicate LWR coolant operating pressure, temperature and neutron flux conditions.

Some of the segments were re-irradiated in these BOCA capsules of R2 in Studsvik at a high Linear Heat Rate (LHR) approximately 30 kW/m, up to a maximum of 71 MWd/kgU peak pellet burnup.

To confirm the integrity of segment fuel assemblies, the coolant activity was monitored during irradiation and ultrasonic tests were performed after irradiation, and no leakage was detected. Visual inspections were performed, and no anomalies were detected. The dimensional changes and waterside oxide thickness were measured by the on-site inspection equipment and no anomalies of the advanced materials were detected.

The detailed PIE works in the hot cell were performed to evaluate the irradiation behaviour at HBU. The segment rod length and diameter profile were measured to investigate the dimensional change during irradiation. To investigate the fuel pellet volume change during irradiation, the density of the fuel pellets was measured by the immersion method. The FGR was evaluated by puncture tests on the segments. Results of the waterside oxide thickness measurements as well as hydrogen content were evaluated in this project.

### 2.3. Extension to high burnup in Reactor (Phase IV) [4]

Phase IV consisted of movement of 12 reference fuel rods, with different features, into a fresh fuel assembly to provide experimental data beyond the licensed rod average burnup. One of the challenges of Phase IV was to manage irradiated rods on the site, fulfilling all the safety and operational requirements from the customer and regulatory authorities.

These efforts produced valuable information on in-reactor performance of fuel materials representing current as well as potential future fuel designs. Those well characterized fuel rods, composed of different cladding and pellets materials, were irradiated in Vandellós II for a fifth cycle up to 75 MWd/kgU rod average burnup.

The results show, based on the fuel volume change, that no acceleration in solid swelling occurs within the burnup range of the data, up to 82 MWd/kgU (peak pellet burnup), despite the formation of the fully developed HBS in the pellets' periphery. The measured gas release fraction under normal operation was within expected limits, i.e. up to 8% for 70 MWd/kgU average rod burnup. No significant gas release occurred despite the HBS presence in the periphery of the fuel pellets.

Finally, the fuel pellet stability up to HBU was demonstrated, as along the whole length of the rod, no dish filling or axial gaps were encountered [3].

### 2.4. High burnup Vandellós Irradiation Programme [5-6]

In 2000, ENUSA and the ANAV (PWR utility Asociación Nuclear Ascó-Vandellós II) jointly launched the HBU Irradiation Programme (Programa de Alto Quemado) aiming to understand the HBUs available margin and high residence times using an RFA (Robust Fuel Assembly), which was the standard PWR product at that time, with a 17x17 array and 9.50 mm diameter fuel rods.

Four RFA assemblies were irradiated in Vandellós II up to an assembly average burnup beyond the licensed value to analyse both the fuel rods and also skeleton (guide thimbles and grids) performance at those conditions.

The RFA HBU fuel assemblies were irradiated in Vandellós II during four consecutive 18-month cycles, reaching 68 MWd/kgU assembly average burnup, after 2030 effective full power days, and following a demanding operating history representative of the one expected in commercial operation.

After each cycle, the four HBU fuel assemblies were inspected for assembly length, grid width, shape (bow, tilt, and twist) and distance between rods (after the 3rd and 4th cycle). After the fourth cycle, additional inspections were carried out to assess grid to rod fretting and grid cell size, and some vanes were cut from the outer straps of the uppermost ZIRLO™ mid grids and sent to hot cells.

Post irradiation examination (PIE) of HBU assemblies irradiated for four 18-month cycles in Vandellós II has shown that the skeleton (grids and thimbles) of the RFA fuel design is capable to support burnups up to 68 MWd/kgU (assembly average burnup) with adequate margin.

The HBU Vandellós II programme allowed to assess the existing margin at HBU for the following parameters: fuel assembly growth, grid growth, fuel assembly deformation, channel spacing, grid to rod fretting, grid cell size, corrosion and hydriding.

Eight rods from the HBU programme were selected to be characterized in a PIE, in the range of 64.0–74.5 MWd/kgU average rod burnup. Four other standard fuel rods irradiated in Vandellós II for two 18-month cycles (at high power achieving a value around 52 MWd/kgU rod average burnup) completed the PIE programme.

Therefore, this characterized fuel was operated under high duty conditions, as high power, and HBU and moreover, these additional twelve fuel rods were selected to cover several other features as different pellet densities from standard to high densities on $UO_2$ fuel rods, and a wide range of gadolinia concentrations from 2% to 8%.

In addition, other examinations completed the hot cell programme such as rod visual inspection, clad metallography, clad corrosion and hydriding measurements and fuel rod growth. The information generated was used as feedback for the fuel performance models, and incorporated into the ENUSA database.

### 2.5. Almaraz New Cladding Programme

During the early 2000's, ENUSA arranged a new fuel irradiation programme focused on the development and validation of alternative new cladding allows with enhanced corrosion resistant and good dimensional stability.

Finally, two subprogrammes were agreed:

- The J-Alloy Program (three types of cladding alloys) in collaboration with KEPCO (representing Japanese PWR plants), ENDESA (representing Spanish PWR Plants), MHI and ENUSA;
- The Alloy-X Program (five variants of cladding alloys) in collaboration with Westinghouse, ENDESA and ENUSA.

ENUSA designed and manufactured between 2005 and 2007, 8+2 RFA LTAs with 36 special fuel rods per LTA. They were irradiated in Almaraz 2 NPP in two stages:

- 8 LTAs (2007-2011) up to licensed burnup limits;
- 2 LTAs (2011-2015) to achieve 68 MWd/kgU rod average burnup.

An extensive PIE, including hot cell exams, were conducted for these rods.

Furthermore, some J-Alloy cladding tubes were subjected to a joint research programme with OECD's Halden Reactor Project (HRP) [7] to evaluate the corrosion resistance of cladding tubes.

## 3. ENUSA PARTICIPATION

ENUSA, in these programmes, held different roles and responsibilities:

- Manufacturing: for qualification of new materials and processes, traceability, pre-characterization and documentation.
- Design: verification of the applicability of codes and methods and specific development (if needed) to address material variants and/or operation beyond commercial limits.
- Licensing: All aspects are subjected to the safety authorities and the conditions and controls, to accomplish the irradiation, are discussed with them.
- Irradiation: Close follow-up and documentation.
- PIE (both on-site and hot cell):
  - Development of techniques and devices for reliable and accurate on-site characterization.
  - Arrangements of on-site required fuel services: handling, re-constitution, inspection.
  - Arrangements for supply of irradiated materials (fuel rods, component parts) to a hot cell laboratory.

## 4. FUTURE INITIATIVES: ADVANCE TECHNOLOGY FUEL

The accident at the Fukushima Daiichi NPP in March 2011 highlighted the benefits for improvement in the current cladding material. Since then, the nuclear industry, research centres and governmental organizations started numerous R&D programmes focused on the development of ATF. ATF is based on the development of new pellet and cladding designs that provide accident tolerance while maintaining or improving the performance of the original fuel, and which, in turn, allow fuel cycle lengthening at lower cost.

One of the most studied ATF concepts is the coating of the zirconium alloy cladding by deposition of a protective layer on its surface. A framework cooperation agreement between Westinghouse and ENUSA in 2018–2020 was a first step for the implementation of Westinghouse EnCore® Fuel technologies in Europe.

As a continuation of this collaboration, both companies in agreement with Electrabel and Tractebel participate in a demonstration programme of Cr coated LTRs under irradiation into Unit 4 of the Doel NPP in Belgium.

There was a precedent in the Byron-2 LTRs programme, which introduced LTRs into demonstration assemblies with cold spray Cr coated claddings in 2019. The Doel-4 Cr coated Cladding Demonstration Program aims to irradiate four RFA-2 OPT XLR assemblies containing 32 Cr coated LTRs with EnCore® technology to enable the collection of first-hand information on the operational behaviour of this novel fuel.

After the first irradiation cycle, an on-site visual inspection of the four LTAs was performed in November 2021, the integrity of the LTR coatings was confirmed and all the LTAs were reloaded for a second cycle of irradiation [10].

## REFERENCES


[1] FORO NUCLEAR. Spanish Nuclear Industry 2022. Online catalogue, https://www.foronuclear.org/en/resources/publications/spanish-nuclear-industry-catalogue-2022/.

[2] IRISA, Y., MURAI, K., GOTO, K., SUZUKI, S., SABATÉ, R., ALONSO, J.M., “Segmented Fuel Rod Irradiation Program on Advanced Materials for High Burnup”, Proceedings of 2000 International Topical Meeting on LWR Fuel Performance, Park City, Utah, 10–13 April (2000).

[3] TSUKUDA, Y., KOSAKA, Y., KIDO, T., DOI, S., SENDO, T., GONZÁLEZ, P., ALONSO, J.M., “Performance of Advanced Fuel Materials for High Burnup”, Proceedings, International Conference Nuclear Fuel ENS TopFuel 2003 - ANS LWR Fuel Performance Meeting. Würzburg, Germany, 16-19 March (2003).

[4] SERNA, J.J., TOLONEN, P., ABETA, S., WATANABE, S., KOSAKA, Y., SENDO, T., GONZÁLEZ, P., Experimental Observations on Fuel Pellet Performance at High Burnup, *J. Nucl. Sci. Technol.* **43** 9 (2006).

[5] AULLÓ, M., GARCIA-INFANTA, J.M., CHAPIN, D., GAGO, J.L., PONTEJO CALVENTE, A., SANCHEZ BAPTISTA, J., “Post Irradiation Examination of High Burnup Assemblies in Vandellós II”, Paper 098, Proceedings of 2010 LWR Fuel Performance/TopFuel/WRFPM Orlando, Florida, USA, September 26-29 (2010).

[6] ARANA, I., MUÑOZ-REJA, C., CULEBRAS, F., “Post-irradiation Examination of High Burnup Fuel Rods from Vandellós II”, Proceedings of 2012 LWR Fuel Performance/TopFuel/WRFPM, Manchester, United Kingdom, 2–6 September (2012).

[7] NISHIKAWA, S., FUJII, H., TSUCHIUCHI, Y., NOVO SANJURJO, M., ALONSO, J.M., “J-AlloyTM, Advanced PWR Fuel Cladding Material Program of J-AlloyTM Development”, Paper 035, Proceedings of 2010 LWR Fuel Performance/TopFuel/WRFPM Orlando, Florida, USA, September 26-29 (2010).

[8] MUÑOZ-REJA, C., QUECEDO, M., GAGO, J.A., NOVO, M., “PWR Fuel Surveillance in Spain: A Decade of Fuel Examinations”, Paper 95, Proceedings of 2010 LWR Fuel Performance Meeting/TopFuel/WRFPM, Orlando, Florida, USA, September 26-29 (2010).

[9] PASTOR, I., DONCEL, N. “Fuel Rod Performance Code for demanding PWR plants”, Paper No.75832, Proc. 17$^{th}$ Int. Conf. on Nuclear Engineering (ICONE), Brussels, Belgium, 12-16 July (2009).

[10] FALLOT, L., MARTÍNEZ, L., DONCEL, N., WALTERS, J.L., ZHANG, J., MEERT, A., “Poolside Visual Inspection of Cr coated Lead Test Rods at Doel4 after One Irradiation Cycle”, American Nuclear Society (2022), Proceedings of TopFuel2022, Raleigh, NC, October 9-13, 2022.

# HIGH BURNUP FUEL DEVELOPMENT AND IMPLEMENTATION STRATEGY AT WESTINGHOUSE

K. BARBER, J. KOBELAK
Westinghouse Electric Company LLC.
United States of America

Westinghouse is supporting an industry initiative to transition to high energy, high burnup (HEHB) fuel to improve the safety and economics of operating NPPs. In support of this initiative, Westinghouse is commercializing two HEHB EnCore® [1] ATF designs; chromium-coated zirconium alloy cladding with doped $UO_2$ ADOPT™ [2] fuel with greater than 5% $^{235}U$ enrichment in the mid-2020's (short-term) and silicon carbide (SiC) SiGA® [1] cladding with uranium nitride ($U^{15}N$) fuel in the early 2030's (long-term).

Due to the apparent ability of the Cr coated cladding to resist crud build-up and the added tensile strength imparted by the thin Cr coating, an effort was made to define potential benefits over and above those of ATF that would offset the added manufacturing costs and neutron penalty. The benefits with qualitative economic evaluations include, but are not limited to:

- Facilitating 24-month cycles through improved fuel rod accident performance in the form of higher burst temperature, a reduced high temperature metal–water reaction, smaller burst balloon strain, and smaller burst opening improving FFRD performance.
- Increase in cladding strength and resistance to oxidation at expected DNB transient temperatures allowing significant plant uprates in linear heat generation rates and reducing FFRD.

Westinghouse is pursuing a two-step process to increase existing fuel rod average burnup limitations. Step 1, referred to as the incremental burnup extension, increases the allowable fuel rod burnup limit to approximately 68 MWd/kgU for rods located in peripheral assembly locations within the reactor core. The incremental burnup extension primary benefits are improved fuel utilization and reduced backend costs resulting from a lower number of feed assemblies. The topical report describing the incremental burnup extension was submitted to the U.S. NRC in December 2020 and is currently under review.

The second step of the burnup limit increase would increase the allowable fuel rod average burnup to approximately 75 MWd/kgU for all of the fuel rods in the core. The biggest challenge to achieving this second step of burnup increase is addressing the phenomena associated with FFRD. Three different options are being pursued by Westinghouse to address FFRD and the attendant consequences. The first is the development of licensing basis analysis methods to explicitly analyse FFRD and the attendant consequences. The second is supporting the EPRI ALS for FFRD. The third option is to demonstrate that none of the higher burnup fuel rods is susceptible to fine fragmentation rupture during a postulated LOCA. The Westinghouse preferred approach to address the potential for fuel dispersal during a postulated LOCA is via the EPRI ALS.

To develop licensing basis analysis methods for FFRD, Westinghouse is updating its current analysis capabilities. Westinghouse's latest advancement in best estimate LOCA analysis technology is the extension to intermediate break and small break LOCA sizes with the FULL SPECTRUM™ LOCA (FSLOCA™) [3] evaluation model. This evaluation model considers LOCA scenarios resulting from a postulated break in a PWR cold leg. This includes the full spectrum of break sizes, from those not compensated by the normal charging pumps flowrates, up to (and including) a double-ended guillotine rupture.

As part of the FSLOCA methodology, a new Westinghouse thermohydraulic system code was created and named WCOBRA/TRAC-TF2 (WCT-TF2). The code allow a complete and detailed simulation of a PWR for any postulated break size, combining an existing two-fluid, three-field, multidimensional fluid equations used in the vessel from WCOBRA/TRAC (WCT) and an upgraded 1D, two-fluid, six-equation formulation for the two-phase flow. The WCT-TF2 code is currently being updated with capability to analyse important phenomena which are associated with HBU fuel rod response during postulated LOCAs. The updates include, but are not limited to, the following:

1) Cladding deformation and rupture models;
2) Decay heat and kinetics models;
3) Fuel pellet thermal conductivity model;
4) Transient FGR model;
5) Pre-burst axial fuel relocation model;
6) Fuel pellet radial power profile model;
7) Model for dispersal of fine fuel fragments.

The updated WCT-TF2 code will interface with core designs from nuclear design and fuel performance data from fuel rod design to calculate the amount of fuel which could be dispersed from HBU fuel rods into the surrounding coolant during postulated LOCAs over a wide variety of boundary conditions. The potential consequences of the dispersed fuel would then be analysed, with a focus on the following:

1) Radiological consequences;
2) Potential for fuel fragment re-criticality;
3) Potential for sump blockage and concerns related to GSI-191;
4) Fuel fragment coolability;
5) Impact on core coolability.

The first four potential consequences identified above can be analysed separately from the LOCA analysis, using output from the LOCA simulations as input to the consequence analyses. However, the last consequence above needs to be considered as part of the LOCA analysis to demonstrate compliance with the emergency core cooling system acceptance criteria, since demonstration of coolable geometry is one of the criteria in 10 CFR 50.46.

In order to obtain analytical capabilities for FFRD, Westinghouse is partnering with industry to perform testing as well as performing internal testing. The past, ongoing and planned activities include:

- Development of a mature testing plan for the ORNL SATS facility;
- Development of plans for shipments and testing of HBU ATF materials at the INL TREAT facility;
- Execution of Cr coated cladding DNB tests at Westinghouse WALT loop;
- Execution of cladding deformation and rupture tests at HBU conditions;
- Shipment of KKL ADOPT fuel rods to Studsvik for FFRD tests and continued participation in the SCIP;
- Collaboration with EPRI on additional testing plans to address data gaps.

The approach to address fuel dispersal during a postulated LOCA using licensing basis modelling capabilities is understood by industry to require a substantial amount of data in order to develop and validate models capable of predicting the HBU fuel rod response during the transient. Therefore, under the coordination of EPRI, multiple US plant operating organizations, vendors, and industry research organizations collaborated to investigate alternate possible paths for addressing FFRD regulatory expectations (i.e. EPRI ALS). The goal of the EPRI-led collaboration was to identify the most feasible path to enable earlier adoption of higher burnup designs, while avoiding the significant risk associated with first-of-a-kind testing and method development associated with explicit analysis of fuel dispersal. After consideration of various concepts, the one deemed most viable was an approach which would continue to rely on the existing cladding-focused criteria for design basis LOCA analyses, and which would assess the risk associated with HBU cores and the phenomenon of FFRD in a risk-informed manner under the guidance in U.S. NRC regulatory guide (RG) 1.174. Most recently, EPRI has modified the approach to leverage leak-before-break as part of the fuel dispersal consequence analysis.

LTR and LTA programmes to gather data from in-reactor irradiation are also ongoing to support the fuel development and qualification process. Results from pool-side and non-destructive irradiation examination of LTRs inserted into the Byron Unit 2 commercial reactor in 2019 along with pool-side results from Doel Unit 4 in 2020 showed no issues and the coated fuel rods were virtually free from crud buildup. Potential benefits for coated cladding in addition to those provided by corrosion resistance, based on the observations of little or no crud buildup, have been identified. Additional LTRs and LTAs of the EnCore fuel are planned using Cr coated cladding and ADOPT fuel pellets to demonstrate HEHB performance up to 75 MWd/kgU.

Finally, it is noted that Westinghouse has many recent achievements related to enabling higher burnup and the development ATF products. The final U.S. NRC safety evaluation report for ADOPT pellets was issued June 2022. The final U.S. NRC safety evaluation report for AXIOM® [4] cladding, which is a non-ATF product with improved corrosion and growth compared to Optimized ZIRLO™ [5] and ZIRLO® [6] cladding to enable higher burnup, was issued in December 2022. The topical report for burnup extension to 68 MWd/kgU was submitted to the U.S. NRC in December 2020, with the review nearing completion and significant progress made on challenging technical issues. The deposition technique for the Cr coated cladding was down selected via an engineering review process to nitrogen cold spray. The Vogtle HBU higher enrichment license amendment request for the >5% $^{235}U$ LTA programme was submitted to U.S. NRC. Westinghouse is focused on supporting industry in the transition to HEHB fuel and implementation of ATF concepts to improve the safety and economics of operating NPPs.

## REFERENCES


[1] KAROUTAS, Z. et al., “Westinghouse EnCore accident tolerant fuel and high energy program”, Proc. Int Conf. TopFuel2022, Railegh, NC, 9–13 October (2022).

[2] HALLMAN L.H., Westinghouse Advanced Doped Pellet Technology (ADOPT$^{TM}$) Fuel, internal report, Westinghouse Electric Company, WCAP-18482-NP-A Revision 0, September 2022.

[3] WESTINGHOUSE ELECTRIC COMPANY LLC., Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology), internal report, Westinghouse Electric Company, WCAP-16996-NP-A Revision 1, November 2016.

[4] WESTINGHOUSE ELECTRIC COMPANY LLC., Westinghouse AXIOM Cladding in Pressurized Water Reactor Fuel, internal report, Westinghouse Electric Company, WCAP-18546-NP-A Revision 0, March 2023.

[5] WESTINGHOUSE ELECTRIC COMPANY LLC., Optimized ZIRLO™, internal report, Westinghouse Electric Company, WCAP-12610-NP-A & CENPD-404-NP-A, Addendum 1-A, July 2006.

[6] WESTINGHOUSE ELECTRIC COMPANY LLC., VANTAGE+ Fuel Assembly Reference Core Report, internal report, Westinghouse Electric Company, WCAP-12610-NP-A & CENPD-404-NP-A Revision 0, April 1995.

**POST IRRADIATION CHARACTERIZATION OF HIGH BURNUP OXIDE FUELS**

P. MISHRA, B.N. RATH, A. SAMANTA, P.M. SATHEESH, P.K. SHAH, J.S. DUBEY, P. P. NANEKAR
Bhabha Atomic Research Centre, Post Irradiation Examination Division
India

India is pursuing an indigenous three-stage nuclear power programme. The nuclear power programme is guided by the limited resources of natural uranium and the vast thorium resources available in the country. The first stage of the nuclear power programme consists of natural $UO_2$ fuelled PHWRs to generate electricity and produce plutonium. The fuel from the first stage is reprocessed to extract Pu, which will be used in fast reactors of the second stage. Thorium will be bred in the fast reactor, to produce uranium-233, which will be used in combination with thorium in the third stage, thermal breeder reactor. Besides, LWR based on foreign cooperation are also being set up in the country.

Currently, total NPP capacity in India is 6780 MWe with 22 operating NPPs. These include 2 units of 1000 MWe WWERs, 2 units of 160 MWe BWRs and 18 units of PHWRs of different installed capacities. More PHWRs and WWERs are under construction. Most of the reactors are PHWRs and are using natural $UO_2$ based 19 element and 37 element fuel bundles. The core average discharge burnup and maximum design burnup of these fuel bundles is 7000 MWd/tU and 15,000 MWd/tU, respectively.

There is a worldwide trend in the nuclear industry to extend the discharge burnup of fuel assemblies in NPPs because HBU fuels provide advantages such as better utilization of fuel, lesser refuelling rate, and reduction in the volume of fuel bundles, . However, Irradiation of fuel to higher burnup leads to degradation in properties of the fuel and cladding, which may limit the life of the fuel pins. Hence, few natural $UO_2$ fuel bundles were irradiated for extended periods in PHWRs up to the burnup of ~22,000 MWd/tU. In order to characterize the HBU effects in PHWR fuels, detailed post-irradiation examinations (PIE) of these fuel bundles was carried out. PIE to assess the performance of fuels is being carried out at the hot cell facilities of Bhabha Atomic Research Centre for nearly five decades. The PIE techniques used to examine the fuels are visual examination, gamma scanning, profilometry, neutron radiography, fission gas analysis, autoradiography, metallography, fractography and mechanical tests on the cladding.

The PHWR fuel element is designed to operate at a high linear heat rating (LHR), hence experiences higher fuel temperature. This leads to higher FGR, which needs to be accommodated in the fuel pin without fission gas plenum. Hence, FGR is an important parameter, which is measured during PIE for fuel pin performance assessment at HBU. PIE results show higher FGR at extended burnups. Modification in the fuel design parameters are required to mitigate FGR in the fuel pin. Studies on the extent of restructuring in the fuel and fuel centre temperature estimation is done by carrying out microstructural studies of the irradiated fuels. Characterization of fission gas bubble size and density distribution is done by examining surfaces from fractured fuel pieces. Axial and radial fission product distribution in the fuel is assessed from gamma scanning of the fuel pins and autoradiography of fuel sections, respectively. Gamma scanning results are useful to arrive at the burnup distribution in the fuel pin. Additionally, radiochemical burnup analysis of fuel is carried out for accurate burnup determination.

Apart from the changes in fuel during irradiation, mechanical properties of Zircaloy cladding also degrades with irradiation due to the damage from fast neutrons, clad oxidation and hydrogen ingress in the clad matrix. Hence, evaluation of clad ductility is carried out using ring tension tests to ensure clad integrity at HBUs. Tests on the irradiated cladding have indicated adequate residual ductility at extended burnups. Hydrogen content in the cladding is analysed by differential scanning calorimetry. Extent of clad corrosion and hydriding is studied by measuring the thickness of oxide layer on the outer surface of the clad and hydride platelet distribution.

In addition to PIE studies, computer code PROFESS (performance analysis of rod-type oxide fuel elements under steady state) is used to calculate all the fuel pin parameters measured during PIE of irradiated fuel pins.

Various non-destructive and destructive techniques at PIE hot cells of Bhabha Atomic Research Centre and fuel performance codes are being used for performance evaluation and failure analysis of fuels. PIE of HBU fuel bundles has shown satisfactory performance of the fuels from 15,000 MWd/tU to 22000 MWd/tU. It provides confidence to extend the average burnup of PHWR fuel with suitable fuel design modifications.

LICENSING EXPERIENCE AND REGULATORY PERSPECTIVE

(Session V)

**Chairpersons**

**S. GORANOV**
Bulgaria

**J. LEHNING**
United States of America

# SAFETY IMPLICATIONS OF USING HIGH BURNUP FUEL ASSEMBLIES

S. BZNUNI
Nuclear and Radiation Safety Center
Armenia

The nuclear industry carries out intensive R&D to increase nuclear fuel discharge burnup, thereby decreasing the number of fresh fuel assemblies in each reload and improving fuel cycle efficiency.

Currently, the main activities in the development and qualification of HBU fuels are focused on the front end of the fuel cycle. However, implementation of HBU fuel will have significant safety implications on the back end of the fuel cycle, since the increase of the fuel burnup would require fuel assemblies with initial enrichment higher than the established 5% threshold. This would influence validation of the reactor physics, criticality safety, source term, decay heat and shielding codes, since codes themselves and related nuclear data libraries for LWR reactors were validated mainly based on experiments with enrichments less than 5%.

This in turn could impact on the transport and storage of fresh and spent fuel assemblies, leading to either design changes by adding additional neutron absorbers or limiting the number of fuel assemblies that could be loaded into transport containers to address criticality safety. Another option to tackle criticality safety could be the application of the burnup credit; however, the validation of depletion codes (used to feed criticality safety codes with spent fuel isotopic data) was mainly based on spent fuel chemical assay data with a burnup under 62 GWd/tU.

An increase in discharge burnup inevitably would lead to an increase in discharge decay heat, which would require either design change to handle the increased thermal load or increasing cooling time in spent fuel pools. In some cases, despite the increase in discharge burnup leading to a decrease in the amount of discharge fuel assemblies, the growth of essential cooling time could potentially lead to a shortage of available places in cooling pools.

An increase in discharge burnup causes an increase in neutron and gamma dose rates in the transportation and storage of spent fuel assemblies. It would require either design change to implement enhanced shielding or increasing cooling time in spent fuel pools. This also could contribute to the rising shortage of available places in cooling pools.

The extension of the fuel rod burnup can potentially cause more corrosion, hydrogen uptake, irradiation growth of zirconium-based materials, and more FGR from the fuel.

Consequently, it will significantly affect radiological consequence analysis since the amount of fission products will essentially increase. More hydrogen uptake can negatively affect the storage safety of spent fuel assemblies.

In the case of LOCA analysis main concern is related to fuel fragmentation, relocation, and dispersal. According to U.S. NRC fuel fragmentation does occur in HBU fuel, and pellet fragments could disperse from a HBU fuel rod which ruptures during a postulated LOCA. Recently, EPRI developed risk-informed methodology to tackle this issue.

The above issues need to be thoroughly and comprehensively assessed by designers and vendors to have a smooth licensing process.

# ADEQUACY OF SAFETY DESIGN CRITERIA AT HIGH BURNUP FUEL TYPE OF UNIT 5&6 KOZLODUY NPP

S. GORANOV
Bulgarian Nuclear Regulatory Agency
Bulgaria

The presentation at the Technical Meeting provided a summary of how relevant safety design criteria have been satisfied for a fuel transition involving high burnup fuel for a specific reactor design.

## 1. KEY QUALIFICATIONS CONDUCTED

- Update of Chapters 1 and 15 of the safety analysis report for units 5 and 6 based on the new safety analysis performed for fuel TVSA-12 safety operation (TVSA is a hexagonal type of nuclear fuel assembly for WWER NPPs; TVSA-12 is a model of TVSA FA with 12 space grids);
- TVSA-12 fuel design, review and assessment for safety operation of unit 5 and 6 for power operation level up to 3120MW;
- Thermohydraulic analysis for LOCA (lost of coolant accident in the primary circuit) and transients with loss of residual heat removal (RHR) for operational conditions in reactor shutdown states defined by PSA;
- Periodic safety review of unit 5 and 6;
- Limit the effects of secondary circuit water or steam piping breaks in the reactor building;
- Safety Analysis Report of NPP “Kozloduy” units 5&6. Review and assessment in compliance with international requirements;
- Study of thermal potential of warm canal and possibilities to be utilized for internal power supply of units 5&6;
- Analysis and assessment of radionuclides released to atmosphere in the case of design extension conditions at an NPP, and analysis of radiological consequences;
- Methodology elaboration for calculation of thermal power of units 5&6 based on primary and secondary circuit real parameters measurement;
- Power upgrade of units 5&6 up to 104%Nom;
- Carry out an analysis of the consequences of internal flooding.

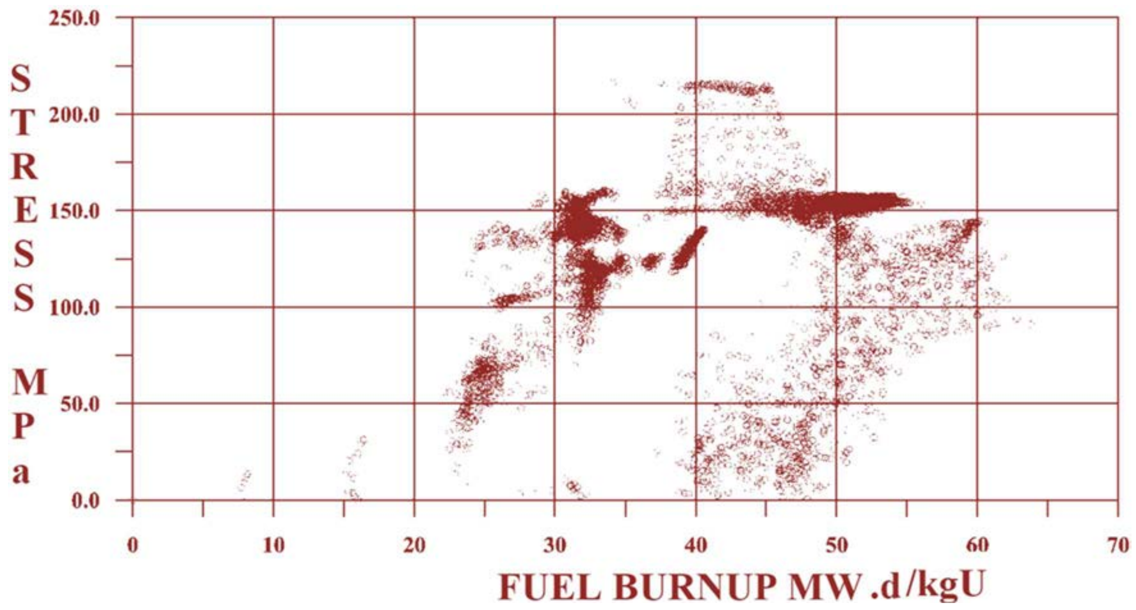



*FIG. 1. Maximum stress of fuel element (FE) cladding depending on fuel burn up in mode of reactor power increasing from 50 %Nnom to 100 %Nnom after long period of time operation (more than 2 weeks) at 50 %Nnom.*

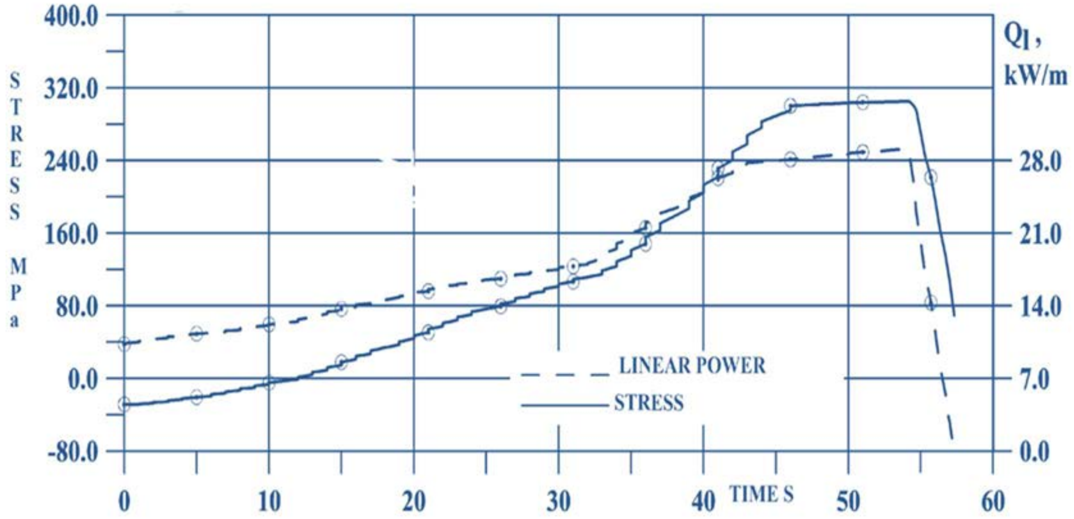



*FIG. 2. Maximum values of the stress of FE cladding depending on linear power and time in mode of uncontrolled withdrawal of a control rod group (end of fuel cycle conditions).*

TABLE 1. LOCA (2´DN 850 MM): FULFILLMENT OF ACCEPTANCE CRITERIA

| Acceptance criteria | TVSA-12 | TVSA |
|---|---|---|
| The fuel rod cladding temperature should not exceed – 1200°C | 977.8°C/864°C<br>FE /FEGa | 1016,2/892<br>FE /FEGa |
| The equivalent cladding oxidation should not exceed a prescribed part of the initial cladding thickness ≤ 17% | 3.24%/4.61%<br>FE /FEGa | 2,6925<br>FE |
| The fuel temperature should not exceed the melting temperature of FE <2800 °C and FEGa temperature < 2370 °C | 1860.4°C<br>1720.3°C<br>FE /FEGa | 1774,8/1500<br>FE /FEGa |
| Total mass of reacted zirconium should not exceed a prescribed part of its total mass in the fuel rods 1% | 0.13 % | 0.165 % |

The verification of the operability and safety of the FE and FEGa (fuel elements containing Gadolinium in the pellets) of TVSA-12 and TVSA have been carried out, according to the selection of design criteria for the corresponding operational modes; a check has been made on the feasibility of the determined design criteria by comparing the predetermined values with the obtained calculated ones (see Table 1 above).

The operability of FE and FEGa is substantiated by checking the fulfilment of a group of design criteria, which ensure the absence of damage to the fuel elements.

In certain operating modes, a short term exceeding of the predetermined design limits (for SC1, or the strength criteria on the stress corrosion cracking (SCC) in the presence of aggressive fission products) was observed, but the satisfaction of the criterion was fulfilled according to the secondary conditions and time limits — not exceeding the incubation period — and the satisfaction of the design criterion SC5 (remaining deformation limit of fuel cladding).

In case of accidents the cooling of the reactor core is ensured, as well as the possibility of post-accident removal and transportation from reactor vessel to spent fuel pool.

The activity in the primary coolant for the sum I131-135 is in accordance with the operational limit for unit 5 and 6 of Kozloduy NPP.

- Factors significantly impacting on the type of FE threshold depressurization are: excess internal pressure under the FE Cladding, sufficient margin of plasticity of the cladding with fuel burnup up to 60 MWd/kg U, oxidation of the external surface of the cladding within 8 µm, oxidation on the inner surface of the cladding less than 20 µm, hydrogenation of the cladding not exceeding 100 ppm, limiting interaction of the fuel with the cladding.
- The formation of several ruptures in fuel cladding with a burnup 60 MWd/kg U is associated with a substantial reduction or disappearance of the gap between the fuel and the cladding;
- It can be concluded that the presence of initial technological defects with a depth of less than 7% of the cladding thickness does not affect the tendency of the FE cladding to SCC. To initiate the corrosion process, the presence of a sufficient amount of iodine

under the cladding is necessary. A conservative calculation of the maximum concentration of free iodine under the FE cladding gives the value 0.2 $mg/cm^2$. The limit value of the SCC at a temperature of 380 ºC for claddings with a technological defect with a depth of up to 35 μm, determined by the results of the calculations, strength limits to $\sigma = 260$ MPa.

# U. S. NRC RESEARCH INFORMATION LETTER ON FUEL FRAGMENTATION, RELOCATION, AND DISPERSAL

J. CORSON
Nuclear Regulatory Commission
United States of America

In December 2021, staff from the U.S. NRC published RIL 2021-13 [1], which provides the staff's interpretation of recent research on FFRD. It focuses on five elements related to FFRD: (1) the conditions at which fuel becomes susceptible to fine fragmentation, (2) the cladding strain above which fuel can relocate axially within the fuel rod, (3) the mass of "dispersible" fuel, (4) transient FGR under LOCA conditions, and (5) the packing fraction of fuel fragments within the balloon region of the fuel rod. Notably, RIL 2021-13 provides the following conservative interpretations of the available data: fuel becomes susceptible to fine fragmentation starting at a pellet average burnup of 55 GWd/tU and to axial relocation once cladding hoop strain exceeds 3%. RIL 2021-13 also provides a conservative model for fuel dispersal based on the aforementioned burnup and strain thresholds, as well as significant uncertainty in the size of the burst opening.

RIL 2021-13 only includes information about the behaviour of uranium dioxide fuel in zirconium alloy cladding; it does not address the potential impacts of adding dopants (e.g. gadolinium, chromium) to the $UO_2$ fuel or of changes to the cladding design (e.g. chromium-coated zirconium alloy cladding). The report acknowledges that FFRD thresholds are defined in terms of surrogate parameters like burnup or cladding strain. There are almost certainly other parameters influencing FFRD, but more research is needed to develop more mechanistic models. Finally, RIL 2021-13 only addresses the behaviour of a single fuel rod under LOCA conditions. However, the information in this report could be applied as part of a full core LOCA analysis to estimate the potential mass that could be dispersed under these conditions. Such estimates could then be used to evaluate the potential consequences of fuel dispersal. U.S. NRC is actively engaged in experimental and analytical research efforts to address some of these limitations, as described in RIL 2021-13 [1] and in a recent Top Fuel paper [2].

## REFERENCES

[1] BALES, M., CHUNG, A., CORSON, J., KYRIAZIDIS, L., Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup, RIL 2021-13, U.S. Nuclear Regulatory Commission, Washington, DC (2021), ADAMS Accession Number ML21313A145.

[2] CORSON, J., CHUNG, A., KYRIAZIDIS, L. and BALES, M., "NRC's Research Information Letter on Fuel Fragmentation, Relocation and Dispersal", American Nuclear Society, Proc. Int. Conf. TopFuel2022, Raleigh, NC, October 9-13 (2022), ADAMS Accession Number ML22151A117.

**ASSESSMENT OF THE U.S. NUCLEAR REGULATORY COMMISSION'S REGULATORY FRAMEWORK FOR THE LICENSING OF HIGH BURNUP FUEL**

J. LEHNING
Nuclear Regulatory Commission
United States of America

The U.S. NRC is preparing to perform independent safety reviews of anticipated licensing applications to permit reactor licensees to increase EOL fuel burnups beyond currently approved limits (e.g. 62 GWd/tU rod-average burnup). Proposals to increase fuel burnup may involve impacts across the nuclear fuel cycle that the U.S. NRC is assessing to assure adequate protection of public health and safety.

In preparation for these anticipated safety reviews, the U.S. NRC has performed an assessment of the existing regulatory framework for the licensing of advanced fuel designs, including HBU fuel, fuel with increased enrichment, and ATF [1]. Key impacts of HBU fuel considered in the assessment include the following:

- Fuel dispersal during a LOCA.
  Relevant issues include core coolability, coolability of dispersed fuel fragments, subcriticality of the core and any accumulations of fuel fragments, assuring no adverse impacts of dispersed fuel on safety equipment, and assuring that dispersed fuel does not unacceptably reduce defence in depth with respect to containing radioactive materials.

- Fuel dispersal during accidents other than the LOCA.
  Non-LOCA events are typically perceived as less challenging with respect to fuel dispersal than the LOCA because the core is expected to remain covered, and impacts may be more localized. However, associated impacts for non-LOCA events have generally not been assessed in as much detail as the LOCA event.

- Cladding embrittlement.
  As fuel burnup increases, existing acceptance criteria in 10 CFR 50.46 may become nonconservative due to such phenomena as hydrogen-enhanced beta layer embrittlement, oxygen ingress from cladding inner surface, and breakaway oxidation. The Commission is deliberating upon a proposed rulemaking (10 CFR 50.46c) that would address these issues [2].

- Radiation dose.
  While some radionuclide concentrations may attain saturation, and while complications may arise from other factors associated with increased fuel burnup (e.g. increases to fuel enrichment, burnable absorber concentration, cycle length), increasing fuel burnup may tend to increase the potential for radiological consequences.

- Fuel storage and transportation.
  Updates to analyses may be necessary in areas such as criticality, decay heat loadings, radiation source terms, and shielding. The U.S. NRC staff is preparing NUREG reports associated with the validation of burnup credit and criticality codes for HBU fuel.

- Environmental impacts.
  The U.S. NRC staff is evaluating ATF, HBU fuel, and increased enrichment relative to environmental impacts specified in 10 CFR 51.52.

While many potential impacts of HBU fuel may likely be addressed by updates to regulatory guidance, the U.S. NRC continues to evaluate potential revisions to regulatory requirements. For example, the Commission has directed that the U.S. NRC staff address fuel fragmentation, relocation, and dispersal in concert with the currently ongoing rulemaking for increased enrichment that is scheduled to be completed in 2026 [3]. The U.S. NRC staff is further preparing for regulatory reviews of industry topical reports describing methods for performing safety analyses for HBU fuel.

In conclusion, responsive to industry plans to propose increased fuel burnup limits, the U.S. NRC is assessing modifications to regulations and guidance across the fuel cycle to promote the safe, consistent, and efficient licensing of HBU fuel. Licensing HBU fuel is a major effort, and significant work remains underway to resolve attendant technical and regulatory issues, particularly those associated with dispersed fuel fragments. The U.S. NRC continues to evaluate changes to regulatory guidance and to regulatory requirements associated with HBU fuel; any regulatory or policy changes would involve deliberation and decision from the Commission.

## REFERENCES


[1] NUCLEAR REGULATORY COMMISSION, Regulatory Framework Applicability Assessment and Licensing Pathway Diagram for the Licensing of ATF-Concept, Higher Burnup, and Increased Enrichment Fuels, ML22014A112 (2022).

[2] NUCLEAR REGULATORY COMMISSION, Draft Final Rule – Performance-Based Emergency Core Cooling System Requirements and Related Fuel Cladding Acceptance Criteria (RIN 3150-AH42), ML15238A947 (2016).

[3] NUCLEAR REGULATORY COMMISSION, Staff Requirements – SECY-21-0109 – Rulemaking Plan on Use of Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors, ML22075A103 (2022).

## LIST OF ABBREVIATIONS

| | |
|---|---|
| ALS | Alternate Licensing Strategy, promoted by EPRI |
| ATF | Accident Tolerant Fuels, Advanced Technology Fuel |
| BEPU | Best Estimate Plus Uncertainty |
| BWR | Boiling Water Reactor |
| CEA | Commissariat à l'Energie Atomique et aux Energies Alternatives (France) |
| CIP | CABRI International Project |
| CRP | Coordinated Research Project |
| CSNI | OECD NEA’s Committee on the Safety of Nuclear Installations |
| DBA | Design Basis Accident |
| DEH | Decay Energy Heat-up |
| DG | Diameter Gauge |
| DNB | Departure from Nucleate Boiling |
| EBSD | Electron Backscattered Diffraction |
| EC | Cladding Elongation |
| EF | Fuel stack Elongation detectors |
| EGRFP | OECD NEA Expert Group on Reactor Fuel Performance |
| EOL | End of Life |
| EPMA | Electron Probe Micro Analyser |
| EPRI | Electric Power Research Institute |
| ET | Expansion Thermometer (to measure fuel centreline fuel temperature) |
| FA | Fuel Assembly |
| FFRD | Fuel Fragmentation Relocation and Dispersal |
| FGR | Fission Gas Release |
| FIB-SEM | Focused Ion Beam, Scanning Electron Microscope |
| GB | Grain Boundary |
| HBS | High Burnup Structure |
| HBU | High Burnup |
| HRP | Halden Reactor Project |
| IAEA | International Atomic Energy Agency |
| IFPE | Integral Fuel Performance Experimental Database (OECD NEA) |
| INL | Idaho National Laboratory (USA) |
| IRSN | Institut de Radioprotection et de Sûreté Nucléaire, France |
| JAEA | Japan Atomic Energy Agency |
| KJMA | Model by Kolmogorov-Johnson-Mehl-Avrami |
| LBLOCA | Large Break LOCA |
| LEU | Low Enriched Uranium (up to 5%) |
| LEU+ | Low Enriched Uranium (from 5 to 10%) |
| LOCA | Loss of coolant accident |
| LTA | Lead Test Assembly |
| LTR | Lead Test Rod |
| LWR | Light Water Reactor |
| MOX | Mixed Oxide Fuel |
| NEA | OECD Nuclear Energy Agency |
| NKSP | Nuclear Key Safety Parameter |
| NPP | Nuclear Power Plant |
| NSRR | Nuclear Safety Research Reactor, Japan |
| NUREG | U.S. Nuclear Regulatory Commission technical report designation |
| ORNL | Oak Ridge National Laboratory (USA) |
| PCMI | Pellet Cladding Mechanical Interaction |

| | |
|---|---|
| PF | Fuel rod Pressure transducers |
| PHWR | Pressurized Heavy Water Reactor |
| PIE | Post-Irradiation Examinations |
| PWR | Pressurized Water Reactor |
| RCS | Reactor Coolant System |
| RFA | Robust Fuel Assembly (Westinghouse LLC.) |
| RIA | Reactivity Insertion Accident |
| RIL | Research Information Letter |
| R&D | Research and development |
| SATS | Severe Accident Test Station, at Oak Ridge National Laboratory (USA) |
| SCIP | Studsvik Cladding Integrity Program |
| SEH | Stored Energy Heating |
| TF | Fuel Thermocouple |
| TREAT | Transient Reactor Test, at Idaho National Laboratory (USA) |
| TWIST | LOCA vehicle for the Transient Water Irradiation System in TREAT |
| U.S. NRC | Nuclear Regulatory Commission (United States of America) |
| VERA | Virtual Environment for Reactor Applications |
| WCR | Water Cooled Reactors |
| WGFS | OECD NEA Working Group on Fuel Safety |
| WWER | Water–Water Energetic Reactor |

## CONTRIBUTORS TO DRAFTING AND REVIEW

| | |
|---|---|
| Arimescu, I. | Consultant, Nufology Plus LLC, United States of America |
| Armstrong, R. | Idaho National Laboratory, United States of America |
| Bales, M. | OECD Nuclear Energy Agency |
| Barber, K. | Westinghouse, United States of America |
| Bznuni, S. | Nuclear and Radiation Safety Center, Armenia |
| Cappia, F. | Idaho National Laboratory, United States of America |
| Capps, N. | Oak Ridge National Laboratory, United States of America |
| Corson, J. | Nuclear Regulatory Commission, United States of America |
| Csontos, A. | Nuclear Energy Institute, United States of America |
| Doncel, N. | ENUSA Industrias Avanzadas S.A. SME, Spain |
| Gassmann, W. | Constellation Nuclear, United States of America |
| Georgenthum, V. | Institut de Radioprotection et de Sûreté Nucléaire, France |
| Goranov, S. | Bulgarian Nuclear Regulatory Agency, Bulgaria |
| Jiao, Y. | Nuclear Power Institute of China, China |
| Khvostov, G. | Paul Scherrer Institute, Switzerland |
| Lee, Y. | Seoul National University, Republic of Korea |
| Lehning, J. | Nuclear Regulatory Commission, United States of America |
| Massara, S. | International Atomic Energy Agency |
| Mishra, P. | Bhabha Atomic Research Centre, India |
| Noirot, J. | Commissariat à l'Energie Atomique et aux Energies Alternatives, France |
| Pizzocri, D. | Politecnico di Milano, Italy |
| Sim, K. | International Atomic Energy Agency |
| Shaw, P. | International Atomic Energy Agency |
| Taniguchi, Y. | Japan Atomic Energy Agency, Japan |
| Tverberg, T. | Institute for Energy Technology, Norway |
| Van Uffelen, P. | European Commission, Directorate General Joint Research Centre |
| Veneau, T | International Atomic Energy Agency |
| Yueh, K. | Electrical Power Research Institute, United States of America |
| Zhang, J. | Tractebel, Belgium |

### Technical Meeting

Vienna, 15–18 November 2022

### Consultancy Meetings

Vienna, 21–23 June 2022
Vienna, 10–12 May 2023

IAEA
International Atomic Energy Agency

No. 26

# ORDERING LOCALLY

IAEA priced publications may be purchased from the sources listed below or from major local booksellers.

Orders for unpriced publications should be made directly to the IAEA. The contact details are given at the end of this list.

## NORTH AMERICA

***Bernan / Rowman & Littlefield***

15250 NBN Way, Blue Ridge Summit, PA 17214, USA

Telephone: +1 800 462 6420 • Fax: +1 800 338 4550

Email: orders@rowman.com • Web site: www.rowman.com/bernan

## REST OF WORLD

Please contact your preferred local supplier, or our lead distributor:

***Eurospan Group***

Gray's Inn House
127 Clerkenwell Road
London EC1R 5DB
United Kingdom

***Trade orders and enquiries:***

Telephone: +44 (0)176 760 4972 • Fax: +44 (0)176 760 1640
Email: eurospan@turpin-distribution.com

***Individual orders:***

www.eurospanbookstore.com/iaea

***For further information:***

Telephone: +44 (0)207 240 0856 • Fax: +44 (0)207 379 0609
Email: info@eurospangroup.com • Web site: www.eurospangroup.com

## Orders for both priced and unpriced publications may be addressed directly to:

Marketing and Sales Unit
International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22530 • Fax: +43 1 26007 22529
Email: sales.publications@iaea.org • Web site: www.iaea.org/publications

24-01641E

International Atomic Energy Agency
Vienna