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REDUCTION OF RADIOLOGICAL ACCIDENT CONSEQUENCES

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# Contents

## Abbreviations

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|---|---|
| e-ATF | Enhanced-Accident Tolerant Fuel |

# 1 Introduction

Following the Fukushima-Daiichi accident, significant international research and development efforts have been initiated to enhance Accident Tolerant Fuel (e-ATF) cladding materials for Light Water Reactors (LWRs). These initiatives aim to improve safety margins for both severe and design basis accidents while maintaining commercial performance. The prolific international research on this topic demonstrated many advantages in the use of e-ATF, but quantitative analysis on what gain that can be expected from e-ATF are not well documented. To bridge the gap between academics results and industrial implementation, the Task 5.3 of the European R2CA aims to provide high expertise analyses and numerical tools with the ambitious objective to evaluate the scaling up of e-ATF to industrial scale.

The first step was to provide relevant bibliographic survey [1] and [2] with the objective to identify e-ATF physical properties to feed fuel codes by providing a common and base of material properties. Several international entities have provided comprehensive overviews of research outcomes. A first detail review of international project addressing e-ATF was provided by JRC and a focus on outcomes of alumina-forming alloys like FeCrAl, ODS steels, SiC-SiC composite materials, and MAX phase materials within the Il Trovatore project. In agreement with identified short-term candidate materials include Cr-coated zirconium-based a bibliographic survey within the R2CA project provided by EDF detailed various chromium coating processes and advances in coating technologies used by fuel vendors like Framatome, Westinghouse Electric Corporation, and counterparts. Material property reviews encompassed thermal and mechanical properties, corrosion resistance, and radiation effects under normal and high-temperature conditions.

With the help of these literature surveys, 3 independent numerical studies were proposed:

1. **Material properties for various ATF materials were upgraded in TRANSURANUS.** Collaborative efforts lead by JRC implemented and assessed $U_3Si_2$ fuel and FeCrAl cladding material properties, including uncertainty analysis using the Monte Carlo approach. [1]
2. FRAPTRAN code sensitivity studies, provided by Tractebel, on large break LOCA scenarios further assessed the performance of Cr-coated Zry cladding materials, necessitating code modifications to reproduce reduced burst strain accurately, but firstly showed delay burst and strain reduction in Cr-coated cladding materials compared to standard Zry-4. [3]
3. IRSN first and EDF then using the DRACCAR code under LOCA conditions showed promising delayed burst and strain reduction in Cr-coated cladding materials compared to standard Zry-4, [4] and [5].

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# 2 Bibliographic survey on e-ATF concepts

Enhanced Accident Tolerant Fuels (e-ATF) represent a significant innovation in the nuclear energy sector, aimed at enhancing the safety and resilience of nuclear reactors during extreme conditions, including accidents. This section reports the bibliographic surveys performed by JRC and EDF with the objective to identify e-ATF physical properties to feed fuel codes by providing a common base of material properties. A first detailed review of international project addressing e-ATF was provided by JRC [1] and a focus on outcomes of alumina-forming alloys like FeCrAl, ODS steels, SiC-SiC composite materials, and MAX phase materials within the Il Trovatore project [6,7]. In agreement with identified short-term candidate materials (including Cr-coated zirconium), a bibliographic survey was provided by EDF [2] within the R2CA project detailing various chromium coating processes and advances in coating technologies used by fuel vendors like Framatome, Westinghouse Electric Corporation, and counterparts. Material property reviews encompassed thermal and mechanical properties, corrosion resistance, and radiation effects under normal and high-temperature conditions.

## 2.1 Overview of international project dedicated to e-ATF materials

The real enthusiasm concerning e-ATF technologies have led to many projects and work done in the frame of international organisations that are listed in Table 1 & 2.

Table 1: List of project activities related to eATF organised by the IAEA.

| Project Acronym | Time frame | Scope | Reference |
|---|---|---|---|
| ACTOF | 2014-2019 | Fuel performance code benchmarking (normal operation, severe accidents)<br>Round robin exercise for ATF cladding | IAEA TECDOC 1921 |
| ATF-TS | 1-2024 | Single rod and bundle tests on ATF performance under normal, DBA and DEC conditions<br>Code benchmarking<br>Development of LOCA evaluation methodology for ATF<br>Development of open-source ATF database | https://nucleus.iaea.org/sites/connect/NFEpublic/Pages/ATF-TS.aspx<br>In progress |

Table 2: List of project activities related to eATF organised by the OECD-NEA.

| Project Acronym | Time frame | Scope | Reference |
|---|---|---|---|
| EGATFL | 2014-2018 | State-of-the-art report | NEA No 7317, 2018 (https://oe.cd/2mL)<br>https://www.oecd-nea.org/science/egatfl/ |
| TOPATF | 2018-2020 | Technical Opinion Paper on Safety Criteria for Accident-Tolerant Fuels | NEA No. 7576, 2022 https://www.oecd-nea.org/jcms/pl_71304/csni-technical-opinion-paper-no-19-applicability-of-nuclear-fuel-safety-criteria-to-accident-tolerant-fuel-designs |
| HRP | 2015-2020 | In-pile experiments with single rods with doped fuel and/or different ATF cladding materials | https://ife.no/en/tag/halden-reactor-project/<br>(HWR-1274) |

| | | | |
|---|---|---|---|
| SCIP IV | 2019-2024 | Out-of-pile fuel fragmentation tests (LOCA conditions) with additive fuels as ATF materials | https://www.studsvik.com/scip-project/scip-iv/ |
| INCA (proposal) | 2021-2024 | Compare in-pile creep of ATF cladding materials (coated Zry) with conventional Zry for VVER-1000 reactors. | https://www.oecd-nea.org/download/science/workshops/neaframework/ |
| QUENCH-ATF (proposal) | 2021-2024 | Integral out-of-pile LOCA bundle test to compare FeCrAl and optZIRLO. Complement to previous QUENCH19 test. | https://quench.forschung.kit.edu/ https://publikationen.bibliothek.kit.edu/1000089029 |

## 2.2 E-ATF Materials and IL-Trovatore outcomes

Within the framework of the European Il Trovatore project [6,7], extensive research and collaborative efforts have focused on identifying, processing, and validating new cladding materials that can overcome conditions better than standard nuclear fuels and claddings.

### 2.2.1 Processing and Evaluations Domains

The Il Trovatore project has systematically approached the development of ATF materials through a series of focused domains:

1. **Processing & Joining of ATF Claddings:** This domain explores innovative methods for fabricating and assembling ATF cladding materials. Techniques such as advanced welding and surface treatment processes are investigated to ensure that the new cladding materials can be integrated seamlessly into existing fuel rod designs without compromising integrity or performance.

2. **Evaluation & Pre-screening:** Before ATF materials can be considered for in-service validation, they undergo rigorous evaluation and pre-screening to assess their physical, chemical, and mechanical properties. This phase involves a combination of experimental testing and computational modeling to identify materials that exhibit superior performance in terms of corrosion resistance, thermal conductivity, and mechanical strength.

3. **In-service Validation:** The most promising ATF materials proceed to in-service validation, where they are subjected to real-world reactor conditions. This critical phase aims to verify the materials' performance under normal operation, accident scenarios, and extended periods, providing empirical data to support their adoption in nuclear reactors.

### 2.2.2 Optimizing and Validating Selected ATF Claddings Concepts

Project partners have mainly focused on 3 promising ATF technologies:

1. **Silicon Carbide (SiC) composites:** offer exceptional resistance to high-temperature steam oxidation, a critical factor during reactor accidents. Their superior strength and corrosion resistance, coupled with low neutron absorption properties, make SiC-SiC composites a promising alternative for cladding materials.

2. **Coatings (MAX-Phase & Oxide Coatings):** Protective coatings on traditional Zirconium-based claddings can significantly enhance their accident tolerance. MAX-phase coatings, consisting of ternary carbides or nitrides, and oxide coatings, such as alumina, provide improved corrosion resistance and thermal stability, protecting the cladding from rapid degradation in accident conditions.

3. **ODS (Oxide Dispersion Strengthened) FeCrAl:** ODS FeCrAl alloys are engineered for high-temperature strength and oxidation resistance. The dispersion of fine oxide particles within the alloy matrix enhances its mechanical properties and resistance to creep and corrosion, particularly in the presence of high-temperature steam and radiation.

Outcomes from the Il Trovatore project yielded significant insights into the performance of these ATF materials. Through collaborative research and testing across various European institutions, the project demonstrated the potential of these materials to substantially improve the safety margins of nuclear fuel rods. Key findings include the superior performance of SiC-SiC composites under steam oxidation conditions, the effectiveness of MAX-phase and oxide coatings in protecting Zirconium-based claddings, and the enhanced durability of ODS FeCrAl in representative environments as for example illustrated in Figure 1.

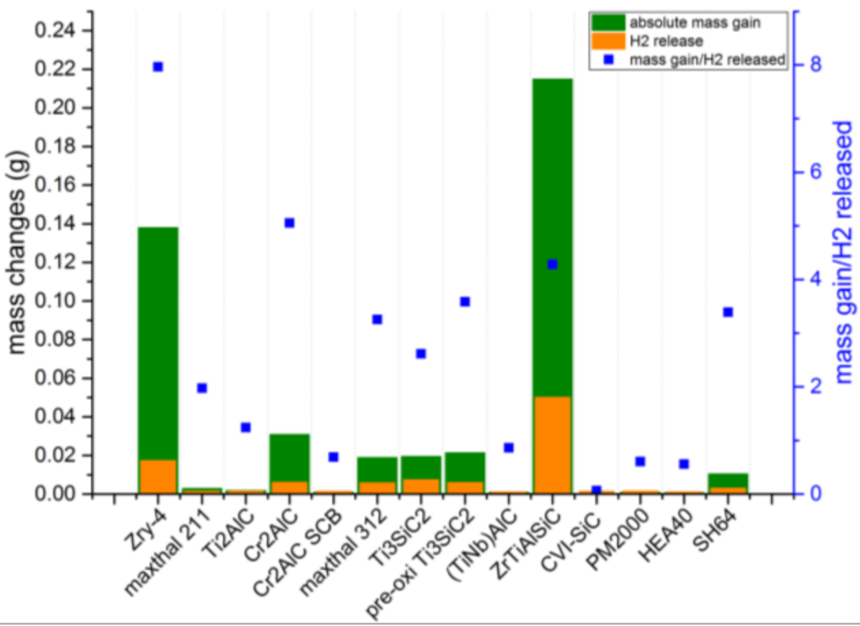


Figure 1: Static autoclave and loop tests under PWR conditions at IIT & SCK [7].

## 2.3 Bibliographic survey on Cr-coated zirconium-based cladding materials

EDF and most of utilities worldwide consider Cr-coated zirconium-based cladding materials as the most promising short-term technology. Zirconium-based materials are widely used as cladding in nuclear reactors due to their low neutron absorption cross-section, excellent corrosion resistance, and favorable mechanical properties. However, their susceptibility to corrosion under specific conditions such as LOCA condition has driven research into enhancing their properties. Chromium coatings have emerged as a promising solution to augment corrosion resistance and overall durability. The focused bibliographic survey provided by EDF [2] explores the advancements in chromium-coated Zr-based materials, focusing on their material properties.

### 2.3.1 Chromium Coating Techniques

The application of chromium coatings aims to improve the mechanical and physical properties of Zr-based materials. Various techniques such as Physical Vapor Deposition (PVD), Chemical Vapor Deposition (CVD), Cold Spray (CS), and 3D Laser-Melt Coating have been developed, each offering unique advantages. Recent advancements include nanostructured coatings for enhanced mechanical properties, multilayer coatings for synergistic benefits, and plasma-based techniques for improved adhesion.

### 2.3.2 Properties of Chromium-Coated Materials

Physical and Thermal Properties: The incorporation of chromium coatings influences several physical and thermal properties crucial for nuclear reactor applications. Key attributes include density, specific heat, thermal conductivity, and fusion temperature of both chromium metal and chromium oxide. These properties play a significant role in the thermal management and stability of nuclear fuels and have been reported in the bibliographic survey provided by EDF.

Mechanical Properties: Chromium-coated materials exhibit enhanced mechanical strength and durability, which are essential for maintaining structural integrity under reactor operating conditions. Studies have shown that coatings do not significantly impact tensile properties at room temperature, and the interface strength between chromium and Zr-based materials is sufficient for operational demands.

Corrosion Resistance: A critical advantage of chromium-coated materials is their improved resistance to corrosion, especially under high-temperature and high-pressure conditions typical of nuclear reactors. Experimental studies confirm that chromium coatings slow down steam oxidation kinetics, providing additional protection during loss-of-coolant accidents (LOCA) and other extreme conditions.

### 2.3.3 Radiation Effects and Durability

The response of chromium-coated materials to radiation is a critical area of study, given the high-radiation environment of nuclear reactors. Initial research indicates promising stability under irradiation, with minimal impact on the coating's mechanical properties and corrosion resistance. However, further investigations are necessary to fully understand the long-term effects of neutron irradiation on coated materials.

### 2.3.4 Future Research Directions

Future research will likely focus on refining coating technologies, developing advanced computational models for material behavior prediction, and customizing coatings for specific reactor designs. Moreover, experimental studies on irradiated samples are crucial for validating the performance improvements and ensuring the safety and reliability of chromium-coated materials in real-world reactor conditions. Collaborative international efforts and standardization of testing protocols will play a vital role in advancing the application of chromium-coated materials in nuclear energy technologies.

### 2.3.5 Conclusion

Chromium-coated Zr-based materials represent a significant advancement in the quest for more accident-tolerant nuclear fuels. By enhancing mechanical properties, corrosion resistance, and stability under irradiation, these materials offer the potential for safer and more efficient nuclear reactor operations. Ongoing research and development efforts are critical for overcoming current challenges and realizing the full potential of chromium coatings in enhancing the safety and sustainability of nuclear power.

# 3 E-ATF evaluation in fuel performance codes

## 3.1 Extension of the TRANSURANUS Fuel Performance Code and e-ATF evaluation

This work has been performed by JRC [1] .

### 3.1.1 Introduction to TRANSURANUS Code Enhancement:

The TRANSURANUS code is a computational tool used for simulating the behavior of nuclear fuel under various conditions. The Joint Research Centre (JRC) has embarked on a project to extend this code to include various materials properties and models for enhanced Accident Tolerant Fuels (ATF). This initiative is driven by the need to accurately predict the performance of ATF materials in reactor environments, thereby ensuring their safety and efficacy.

The first step in extending the TRANSURANUS code involves the integration of physical, chemical, and mechanical properties of new ATF materials based on information from the open literature. These materials include, but are not limited to, Uranium Silicide (U3Si2) fuel, Iron-Chromium-Aluminium (FeCrAl) cladding, and Silicon Carbide (SiC) composites. The inclusion of these materials requires a comprehensive dataset that describes their behavior under a range of operational and accident conditions.

The process of integrating these properties into the TRANSURANUS code involves several tasks:

- Compilation of Material Data: Gathering extensive experimental and theoretical data on the thermal, mechanical, and radiation response properties of ATF materials.
- Model Development: Developing or refining models that accurately represent the behavior of these materials under normal operating conditions, as well as under accident scenarios such as loss-of-coolant accidents (LOCA) and reactivity-initiated accidents (RIA).
- Code Implementation: Incorporating these models into the TRANSURANUS code, ensuring that they interact correctly with existing models and algorithms to provide reliable simulations of fuel rod performance.

### 3.1.2 Uncertainty and Sensitivity Studies with TU/Python

With the new materials properties and models incorporated, the JRC conducted uncertainty and sensitivity studies using TU/Python, a graphical user interface designed for the TRANSURANUS code [8]. These studies are essential for assessing the reliability and accuracy of the code's predictions in the context of ATF materials.

The objectives of these studies include:

- Identifying Key Parameters: Determining which material properties and model parameters have the most significant impact on the performance and safety margins of ATF-equipped fuel rods.

- Quantifying Uncertainties: Assessing the uncertainty in code predictions due to variability in material properties and modeling assumptions. This involves generating probability distributions for input parameters and analyzing their influence on output variables.

- Sensitivity Analysis: Employing statistical methods to quantify the sensitivity of simulation results to changes in input parameters. This analysis helps prioritize research and development efforts by identifying the properties that most critically affect fuel performance.

### 3.1.3 Statistical Modelling with TU and TU/ Python

Statistical modelling in the context of nuclear fuel performance, particularly with the TRANSURANUS (TU) code, allow understanding the behavior of nuclear fuel under a variety of conditions. This approach leverages mathematical and computational techniques to predict the outcomes of complex physical phenomena within nuclear reactors, incorporating the inherent uncertainties in material properties, operational conditions, and model parameters.

Secondly, a central component of the statistical modelling approach is Monte Carlo input sampling, a method used to address the uncertainties in the input parameters of the fuel performance simulations. This technique involves generating many input parameter sets randomly, according to specified probability distributions for each parameter. These distributions reflect the range and likelihood of each parameter's potential values, based on material properties.

Monte Carlo sampling allows for the exploration of the input parameter space extensively, ensuring that the simulations capture a wide array of possible scenarios. This approach is particularly valuable in assessing the reliability and safety margins of nuclear fuels, as it accounts for the variability and uncertainty inherent in fuel behavior predictions. In general, three types of parameters are considered: Material properties, Fabrication parameters, Irradiation parameters. Material parameters considered in the (TUPython) study [8] are given in the following figure.

| **Fuel material property** | **Uncertainty range** |
|---|---|
| Thermal conductivity | ±5% |
| Thermal expansion coefficient | ±5% |
| Creep rate | 0.5-10 (multiplication factor) |
| Density | ±2% |
| Elastic modulus | ±30% |
| Swelling | ±20% |
| Heat capacity at constant pressure | ±5% |

| Clad material property | Uncertainty range |
|---|---|
| Thermal conductivity | ±20% |
| Thermal expansion coefficient | ±30% |
| Creep rate | ±50% |
| Poisson ratio | ±10% |
| Elastic modulus | ±5% |
| Heat capacity at constant pressure | ±5% |
| Yield stress | ±30% |

Figure 2: Material parameters considered in TU/Python study

To facilitate the execution of statistical modelling and analysis, the TRANSURANUS code is complemented by TUPython, a graphical user interface designed to enhance user interaction with the code, especially for statistical analyses. TUPython serves key functions in the statistical modelling process:

- Simplifying Configuration: TUPython provides a user-friendly environment for setting up simulation parameters, including the definition of probability distributions for Monte Carlo input sampling.
- Automating Simulations: It automates the process of running multiple simulations with varied input sets generated through Monte Carlo sampling, significantly reducing the manual effort required and minimizing the potential for errors.
- Analyzing and Visualizing Results: After simulations are completed, TUPython aids in processing the output data, allowing users to perform statistical analyses such as sensitivity analysis and uncertainty quantification. The tool offers visualization capabilities, such as plotting histograms, scatter plots, and sensitivity charts, which are crucial for interpreting the results and drawing meaningful conclusions.

Parameters of interest can be evaluated, JRC provided in this study [8] analyses on many of these parameters:

- Impact of ATF on central temperatures
- Impact of ATF on integral fission gas release
- Impact of ATF on rod pressure and gap size
- Effect of material property uncertainties (TUPython) on central fuel temperature
- Effect of material property uncertainties (TUPython) on FGR and clad hoop stress
- Correlations between cladding outer radius and various cladding properties
- Evolution of Pearson’s coefficients of sensitivity for the outer cladding radius

### 3.1.4 Conclusion

The extension of the TRANSURANUS fuel performance code to include enhanced ATF materials is a significant step towards the deployment of safer and more efficient nuclear fuels. By integrating comprehensive material properties and models into the code and conducting thorough uncertainty and sensitivity analyses using TU/Python, the JRC aims to provide a robust framework for evaluating the performance of ATF in nuclear reactors. This work achieved these following proposes:

- Successful implementation of new ATF properties (U3Si2, FeCrAl, Hastelloy) reflect excellent structure of the TRANSURANUS platform.
- Preliminary simulations confirm reduced ATF fuel temperatures and clad creep →improve the fuel rod behaviour under normal operation conditions.
- Need to extend models/material properties (e.g. FG model and TUBRNP version for Cr-doped fuel, high temperature properties for Cr-coated cladding, etc.).
- New TUPython tool also extend the UA/SA analysis to include e.g. Pearson coefficients.

## 3.2 Evaluation of the LOCA performance of the evolutionary Accident Tolerant Fuel (eATF) with Chromium coated Zr cladding with FRAPTRAN

This section presents the Tractebel contribution [3]), including an assessment of the performance of the Chromium coated Zirconium cladding during LOCA with FRAPTRAN. The section includes the following aspects:

- Review the applicability of FRAPTRAN fuel rod code models, and if necessary, propose necessary modifications of the models for simulation of the behaviour of the Chromium coated Zirconium cladding during LOCA (3.2.1).
- Perform the sensitivity analysis with FRAPTRAN fuel rod code to assess the performance of the Chromium coated Zirconium cladding during LOCA (3.2.2).

### 3.2.1 The applicability of FRAPTRAN fuel rod code for simulation of the expected Cr-coated Zirconium cladding LOCA behaviour

The expected fuel behaviour during LOCAs for ATF cladding is similar to the conventional Zirconium alloy (Zr) cladding, but with delayed ballooning and burst and reduced PCT, increased resistance to cladding post-quench ductility loss (rupture) and increased coping time under design basis accidents or design extension conditions.

The updated FRAPTRAN-TE-1.5 code has been used to simulate the Halden LOCA tests (IFA-650.9 and IFA-650.10). It has been shown that:

- The updated FRAPTRAN-TE-1.5 can well simulate the LOCA fuel thermal behaviours (PCT, rod internal pressure, burst time) as observed in the Halden LOCA tests IFA-650.9 and IFA-650.10, using adequate thermal hydraulic boundary conditions and the axial relocation model.
- The updated FRAPTRAN-TE-1.5 does not well simulate the LOCA fuel mechanical deformation as observed in the Halden LOCA tests IFA-650.9 and IFA-650.10, using the default FRACAS-I model and BALON2 ballooning model.
- The finite element analysis (FEA) model with burst criterion based on the cladding effective plastic strain (with BALON2 ballooning model off) predicts better the LOCA fuel mechanical deformation as observed in the Halden LOCA tests IFA-650.9 and IFA-650.10.

In order to test the feasibility of FRAPTRAN for simulation of expected LOCA fuel behaviors with coated cladding ATF and to identify possible further model improvements, some parametric sensitivity studies are performed using different model options and assumptions on the FRAPTRAN-TE-1.5 simulation of the Halden LOCA tests (IFA- 650.9 and IFA-650.10) [29].

The parametric sensitivity study on base irradiation and transient high temperature oxidation is performed to simulate the expected effect of coated cladding with the current version of FRAPTRAN-TE-1.5. It has been shown that the reduced initial corrosion and transient high temperature oxidation, as long as allow FRAPTRAN-TE-1.5 to simulate the expected behaviours with coated cladding:

- to delay the time of burst.
- to reduce the peak cladding temperature.
- at a lower extent, to reduce the cladding axial elongation.

The parametric sensitivity study on the mechanical model options and burst criteria also allows simulating a delayed burst time but does not allow simulating the expected reduced cladding radial deformation.

The above results confirm the feasibility using model options and multipliers on the current models for Zr-4 corrosion and high temperature oxidation to simulate, in a simplified manner, the expected coated cladding behaviours except for the reduced burst strain during LOCA.

This section will focus on the sensitivity study on the mechanical model and burst criteria. The previous burst tests with Cr-coated cladding show a distinct reduction in circumferential strain and rupture opening size for the Cr-coated claddings compared to the uncoated claddings, and higher burst temperatures or burst times are also found for the Cr-coated claddings. The two main reasons for the improved performance are the lower mechanical degradation of Zircaloy due to the protection of Cr coating from oxidation and the much higher relative strength and creep resistance of Cr vs. Zircaloy at high temperatures in range where Zircaloy burst is typically observed (700–1000 ◦C). However, whether the performance of Cr-coated cladding is bounded by the existing Zircaloy burst criteria remains unknown, and more separate and integral tests on Cr-coated cladding performance during LOCAs are required to further enrich the database for determining the best-estimated criterion of Cr-coated cladding burst.

The cladding burst criteria for Zircaloy cladding burst under LOCA conditions can be categorized in three types:

- stress-based criteria,
- strain-based criteria,
- plasticity instability criterion.

In FRAPTRAN, the default option for ballooning and burst model is the FRACAS-I thin wall deformation model (mechan=2) and the BALON2 burst model (noball=0) BALON model, which is based on both an empirical hoop stress limit and an empirical strain limit.

After the cladding deformation has been calculated by FRACAS-I, a check is made to determine whether or not the cladding ballooning model should be used. The check consists of comparing the calculated cladding effective plastic strain with the cladding instability strain given by MATPRO.

Instability strain = f (average temperature at rupture, effects of irradiation, cold work, and temperature variation)

If the cladding effective plastic strain is greater than the cladding instability strain (0.05), the ballooning model, BALON2, is used to calculate the localized, nonuniform straining of the cladding. Once the instability strain is reached in one node, no further strain is calculated by FRACAS-I for any nodes.

The BALON2 model divides the ballooning node into 12 radial and 12 axial sub-nodes. For the node that has reached the instability strain, the radial average hoop, axial, and radial strains at the axial sub-node with the maximum hoop strain calculated by BALON2 is used as the hoop, axial, and radial strains for the ballooning node. BALON2 calculates the extent and shape of the localized large cladding deformation that occurs between the time that the cladding effective strain exceeds the instability strain and the time of cladding rupture.

The cladding is assumed to consist of a network of membrane elements subjected to a pressure difference between the inside surface and the outside surface.,. The equations for the model are derived from the thin shell membrane equilibrium equation and geometric constraints. In addition, the model calculates the temperature rise of the cladding due to heat transfer across the fuel-cladding gap. The fuel surface is assumed to have a nonuniform temperature. The model accounts for the extra cooling the cladding receives as it bulges outwardly.

The stress-based criteria are assumed that the cladding fails when the cladding true hoop stress exceeds the burst stress. The hoop stress for symmetrical deformation of a thin-wall tube is given by:

$$\sigma Hoop = p/Rs \quad (1)$$

where p is the differential pressure, R is the instantaneous mean tube radius and s is the instantaneous tube wall thickness. It is assumed that the cross-sectional area of the tube wall is conserved during deformation.

An empirical hoop stress limit σB in FRAPTRAN is used in FRAPTRAN for Zircaloy-2, Zircaloy-4, ZIRLO™, Optimized ZIRLO™ and M5™.

### 3.2.2 Sensitivity analysis with FRAPTRAN fuel rod code to assess the performance of the Chromium coated Zirconium cladding during LOCA

The following sensitivity studies were performed using 9 parametric cases named P1 to P9 and compared to an initial set of data corresponding to a specific LOCA transient to simulate the LOCA performance of the evolutionary Accident Tolerant Fuel (e-ATF) with Chromium coated cladding:

Reducing the corrosion and high temperature oxidation (P1 vs. P0) leads to a simulation of:
- Slightly delayed burst (expected).
- Slightly increased cladding deformation (unexpected).
- Decreased cladding temperatures (expected) (expected).
- Decreased ECR (expected).
- Deactivating BALON2 while keeping the FRACAS-1 model (P2 vs. P1) leads to:
- Important delay of burst (expected).

The use of the FEA model (P3-9 vs. P0-2) leads to:
- Slightly increased burst time (expected).
- Increased cladding deformation (unexpected).
- Significant increased stress (expected).

The rupture model "NUREG-0630 fast ramp" (P3 vs. P4) leads to:

- Reduced cladding deformation (expected).
- Reduced cladding temperatures (expected).
- Reduced ECR (expected).

Reducing the maximum effective plastic strain value with FEA model (P5-8) leads to:

- Reduced burst time for low values corresponding to Cr-coated claddings (~0.25) (unexpected).
- Reduced cladding deformation (expected).
- Reduced peak cladding temperature (expected).

Reducing the annealing rate for the cladding material (P9 vs. P5) leads to:

- Important increase of burst time (expected).
- Slightly reduced burst strain (expected).
- Significant increased stress (expected).

### 3.2.3 Conclusion

Simulating Cr-coated cladding behaviours during a typical NPP LOCA with FRAPTRAN has been tested through the selection of appropriate mechanical models and material properties. This is done by adjusting the corrosion and high temperature oxidation models, the mechanical and rupture models. The existing FRAPTRAN model options allow to simulate the following behaviours of Cr-coated claddings with reduced corrosion and high temperature oxidation, BALOB2 off, FEA model with NUREG-0630 burst strain model or user-defined rupture strain, and reduced annealing rate:

- Delayed burst.
- Lower cladding temperatures.
- Decreased ECR.

However, some code modifications in FRAPTRAN are required in order to be able to accurately simulate the reduced burst strain of Cr-coated cladding:

- Burst criteria: some multipliers on the burst stresses (*sigburststress*) and strains (*sigburststrain*) and on the instability strain for BALON2 model (*siginstabstrain*) are available in FRAPTRAN but doesn't allow to affect the burst time. Specific burst stress or strain models should be developed based on the separate effect tests and implemented in FRAPTRAN.
- The reduced annealing rate (*sigcladanneal*) seems favourable for simulating slightly reduced strain at burst together with delayed burst but need to be validated against test data.
- The high temperature creep law of Cr-coated cladding should be developed and implemented.

## 3.3 Cr-coated zirconium-based cladding materials evaluation in LOCA conditions with the DRACCAR code

In this section, some insights from IRSN's DRACCAR simulations performed within the project are given.

### 3.3.1 Introduction

In the continuous pursuit of nuclear reactor safety enhancements, the French Institute for Radiological Protection and Nuclear Safety (IRSN) is progressively gathering the knowledge on Accident Tolerant Fuels (ATFs) throughout the development of software in the FUEL+ platform. In particular, the DRACCAR software's capability to assess the behavior of chromium-coated Zirconium-based cladding materials under Loss-Of-Coolant Accident (LOCA) scenarios were highlighted during the project. This section synthesizes the results gained from the QUENCH-L1 experiments and the subsequent simulations, underscoring the potential of ATF materials in elevating nuclear reactor safety margins.

### 3.3.2 DRACCAR modelling and ATF Simulation

The DRACCAR software, developed by IRSN, is a software coupling thermal hydraulics and thermomechanics at sub-channel scale and devoted to multi-rod configuration analysis. The tool is designed for simulating the behavior of fuel assemblies during accident conditions including ballooning, contact between rods, channel blockage and reflooding. In this context, the QUENCH-L1 experiments conducted at the Karlsruhe Institute of Technology (KIT) with Zry-4 claddings provided a valuable dataset for assessing the performance of ATF cladding candidates, particularly chromium-coated Zirconium-based materials.

Initial simulations employed ATF material properties, with a specific focus on the creep rate and burst stress criteria. Notably, the creep rate for the chromium-coated Zirconium alloy (Zry-4) was adjusted to half of the standard Zry-4 rate. As no specific data concerning burst were found in opened literature, the burst stress criterion was kept identical to the standard. This modification helps to understand the material's deformation under LOCA conditions.

### 3.3.3 Simulation Outcomes

The DRACCAR simulations revealed several key findings. Firstly, the adjustment of the creep rate significantly impacted the maximum strain experienced by the cladding, effectively reducing it to half of that observed in the standard simulation case. This simulation result suggests the potential of ATF materials, particularly chromium-coated Zry-4, to withstand the mechanical stresses encountered during a LOCA scenario more effectively than traditional materials.

Moreover, the simulations indicated that the burst for ATF-like materials occurred slightly later compared to the standard laws. This delay, consistent with the lower creep rate, suggests an enhanced tolerance to the extreme conditions, manifesting as higher burst pressure and temperature thresholds. However, it was noted that the location of the burst prediction remained unaffected by the material laws, indicating a need for further investigation into the hot spot distribution effects and their impact on cladding failure. This result needs to be confirmed as it was obtained by applying standard material burst criterion and not a dedicated model specific to Cr coated Zr alloy.

### 3.3.4 Prospects

The preliminary investigations conducted using the DRACCAR software provide a promising outlook for ATF cladding candidates. However, the research underscores the necessity for continued exploration into the specific parameters that govern the performance of these advanced materials. Future prospects include testing alternative burst criteria, examining the oxidation behavior of ATF materials, and incorporating comprehensive models that can accurately simulate the multifaceted interactions between the cladding, coolant, and fuel under accident conditions.

The advancements in ATF cladding materials, exemplified by the chromium-coated Zry-4 alloy, represent a significant milestone in the quest for safer nuclear reactors. IRSN's simulation efforts using the DRACCAR software have provided valuable insights into the potential of these materials to enhance reactor safety during LOCA scenarios. As the nuclear community moves forward, the findings from these simulations will undoubtedly play a critical role in guiding future research, development, and implementation strategies for Accident Tolerant Fuels. The ongoing collaboration between research institutions, industry partners, and regulatory bodies will be pivotal in realizing the full potential of ATF technologies, ensuring the sustainable and safe operation of nuclear reactors worldwide.

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## 3.4 Advancements in Accident Tolerant Fuel (ATF) Cladding: A Synthesis of EDF's Research and International Collaboration

### 3.4.1 Introduction

The Electricité de France (EDF) has been actively involved in the Research & Development (R&D) of Accident Tolerant Fuels (E-ATF) that promise to offer higher safety margins, especially during extreme conditions such as Loss of Coolant Accidents (LOCAs).

### 3.4.2 EDF's Vision for E-ATF Development

The primary goal of E-ATF development is to extend the available response time before the activation of ultimate safety systems, thereby preserving the core integrity for as long as possible in case of core drying up. E-ATF fuels address multiple issues including potential cost savings, increased safety margins in accidental situations (mainly LOCA) and ensuring fuel performance and reliability are at least equivalent to current standards. This initiative also serves to demonstrate the nuclear industry's dedication to continuous safety improvement through innovation.

### 3.4.3 Insights from Comparative Studies

EDF, alongside international partners, has conducted several comparative studies using the simulation tool DRACCAR to evaluate the performance of E-ATF against standard fuel under LOCA conditions [5]. These studies revealed that Chromium coatings on Zr alloys significantly reduce the creep rate and increase the burst temperature, enhancing mechanical strength.

Preliminary investigations show that adjusting the creep rate for Cr coated Zry-4 to half of the standard Zry-4 rate, while keeping the burst stress criterion identical, yielded promising results. Comparative analyses using different models and simulations consistently demonstrate that E-ATFs, particularly those utilizing Cr-coated Zr alloys, exhibit lower balloon diameters and higher burst temperatures.

### 3.4.4 Conclusion

The pursuit of E-ATF development represents a critical step towards bolstering the safety and reliability of nuclear reactors. EDF's efforts, coupled with international collaboration, underscore the global commitment to achieve higher safety standards through innovative fuel technology. As the nuclear community moves forward, the integration of new material properties, enhancement of existing models, and the continual assessment of E-ATF's performance will remain paramount. The promising outcomes of ongoing research affirm the potential of E-ATFs in maintaining reactor integrity under adverse conditions (mainly LOCA), ultimately contributing to the safer and more sustainable operation of nuclear power plants worldwide.

# 4 Summary and conclusions

The European project R2CA provided several overviews of the outcomes of the research on enhanced accident tolerant fuel (eATF) sometimes referred to as advanced technological fuels. The JRC took part in the review of alumina forming alloys, such as FeCrAl, ODS steels, SiC-SiC composite materials and MAX phase material analysed in the frame of the Il Trovatore project. The analysis covered the static autoclave tests under PWR conditions, showing excellent performance of currently used zirconium alloys and that chromia showed the lowest corrosion rate after zirconia. Furthermore, steam oxidation tests at KIT performed at 1200°C also revealed the low reaction rates of SiC and that FeCrAl showed a high resistance against steam oxidation.

One of the main near-term candidate materials that stand out are the Cr-coated zirconium-based cladding materials. In the frame of the R2CA project, a bibliographic survey was prepared by EDF. The review started with the various chromium coating processes currently applied such as physical or chemical vapour deposition, cold spray and 3D laser-melt coating. Most of these coating techniques are currently used by fuel vendors Framatome, Westinghouse Electric Corporation in the US (PVD and CS), and in South Korea (Laser Melt Coating). In addition, some of the advances in coating technologies were summarized. The review of material properties then covered thermal properties (thermal expansion, diffusivity, conductivity, and heat capacity at constant pressure), mechanical properties for both normal operating conditions as well as at high temperature for loss of coolant conditions, corrosion resistance and the effects caused by radiation.

Based on the review, a first assessment was made during the project of the Cr-coated cladding materials under LOCA conditions by means of the DRACCAR code, comparing standard fuel with Zry and the coated cladding material under LOCA conditions. More precisely, IRSN re-assessed the QUENCH L1 case from the QUENCH LOCA program of KIT, which was applied in the coordinated research project (CRP) of the IAEA FUel Modelling under Accident Conditions (FUMAC). For this purpose, the creep rate of Zry-4 was divided by a factor 2, the same burst stress criterion of standard Zry-4 was applied, and the same oxidation model was also considered. A maximum strain in the ATF-like material that is about half that of standard Zry-4 along was obtained with a delayed burst that was consistent with experimental observations, despite the large scatter in the experimental data in terms of circumferential strain and position of the burst. A similar assessment was made by EDF with DRACCAR based on the IFA650.10 case of Halden, which was also applied in the FUMAC benchmark of the IAEA. Adopting the same assumptions, a slight delay of the burst in Cr-coated Zry in comparison with the standard cladding was also shown.

The impact of Cr-coated Zry cladding materials during LOCA conditions was also assessed by means of a sensitivity study with the FRAPTRAN code on the basis of a large break LOCA scenario with a maximum peak cladding temperature of 1189°C and equivalent cladding reacted (ECR)of 6.76%. The simulations considered a reduced corrosion and high temperature oxidation, a finite element analysis model applying the NUREG 0630 burst strain model or a user defined rupture strain, and a reduced annealing rate. Nevertheless, some code modifications in both DRACCAR and FRAPTRAN codes are still required to be able to reproduce the reduced burst strain of Cr-coated claddings, including a high temperature creep law.

Finally, some material properties for various ATF materials were upgraded in TRANSURANUS by JRC. In collaboration with different partners the properties for U3Si2 fuel and FeCrAl cladding material properties have been implemented progressively and assessed on the basis of a case from the CRP FUMEX-II. In addition to estimating the impact of new material properties on the fuel performance, an uncertainty analysis for the ATF material properties was also performed by means of the built-in Monte Carlo approach in TRANSURANUS. Analysis of the outcome by means of the TUPython tool allowed to evaluate the impact of the uncertainties of the ATF properties, and to evaluate the relative impact by means of the Pearson coefficient.

A similar task was carried out in collaboration with the University of Cambridge for the material properties for Hastelloy-N cladding. In a next phase, also Cr-coated Zircaloy will be simulated by means of the TUmech tool that has been created in the frame of a collaboration with CIEMAT. Nevertheless, the publication of this work with Cr-coated cladding will be out of the timeframe of the R2CA project.

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