# Technical Meeting on Advanced Technology Fuels:Progress on their Design, Manufacturing, Experimentation, Irradiation, and Case Studies for their Industrialization, Safety Evaluation, and Future Prospects

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# Book of Abstracts

# Contents

**4 / 2**

## FUEL SAFETY EVALUATION METHODOLOGY FOR ATFS UNDER LOSS-OF-COOLANT ACCIDENTS

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In response to the Fukushima Daiichi accident, the development of advanced technology and accident tolerance fuels (ATFs) aims to enhance accident tolerance. The International Atomic Energy Agency (IAEA) launched a Coordinated Research Project (CRP) on Testing and Simulations for ATFs (ATF-TS) in 2020, which consists in four key Work Tasks (WTs) [1]. The work task 3 aims at development and application of a best estimate plus uncertainty fuel safety evaluation methodology (FSEM) for ATFs during Loss-of-Coolant Accidents (LOCAs) [2]. The methodology leverages validated fuel rod codes through Halden integral LOCA tests to simulate the behaviour of ATFs under LOCA conditions in typical nuclear power plants (NPPs), using thermal hydraulic boundary conditions generated by a system thermal hydraulic code [3]. The outcomes of the WT3 were documented in volume 3 of the IAEA TECDOC [4]. This paper presents the selected cases, participants, fuel rod codes, ATF concepts, the simulated cases, specified assumptions and methods, results and discussions, as well as the perspectives.

**2 / 3**

## LEAD TEST RODS PROGRAM WITH CR-COATED CLADDING AT DOEL-4

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In order to obtain the first-hand information and valuable in-pile experience with evolutionary accident tolerant fuel (ATF) products, ELECTRABEL (ENGIE) and European Fuel Group (EFG) has reached an agreement in 2019 to load 4 RFA-2 OPT XLR fuel assemblies with EnCore® Cr-coated cladding lead test rods (LTRs) at Doel Nuclear Power Plant Unit 4 (Doel-4). Under this agreement, ENUSA and Westinghouse (EFG) are responsible for fuel design and safety evaluation, manufacturing, transportation and on-site inspection; TRACTEBEL (ENGIE) is responsible for core design and safety evaluation, as well as independent review and support to licensing; ELECTRABEL is the Licensee of the plant Doel 4, and is responsible for fuel loading and operation [1]-[4]. The licensing has been performed by Bel V, the technical support organization (TSO) to the Belgian Safety Authority (FANC) [5].

The Doel 4 LTR program was reviewed and approved by Bel V, in February 2020. The LTRs at Doel-4 were loaded at Doel-4 in July 2020, and have since completed its third cycle of operation in fall 2024. Pool side inspection has been performed after the first cycle [6] and second cycle [7], and the inspection results have been presented to Bel V to demonstrate the good performance and reloadability of the LTRs.

This paper presents the LTR program (§2), fuel design and safety evaluation (§3), core design and safety evaluation (§4), pool side visual inspection results (§5).

**4 / 4**

## DEVELOPMENT OF A LOCA FUEL SAFETY EVALUATION METHODOLOGY FOR CR-COATED CLADDING ATF - TRACTEBEL Contribution to IAEA CRP ATF-TS (2021-2024)

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Following the successful Coordinated Research Project (CRP) on Fuel Modelling in Accident Conditions (FUMAC) [1][2], the International Atomic Energy Agency (IAEA) initiated a new CRP ATF-TS in 2021 to support the development and evaluation of Advanced Technology and Accident Tolerant Fuels (ATFs) [2].

TRACTEBEL's contribution to the CRP ATF-TS focused on developing a methodology to assess the performance of Cr-coated cladding under Loss-of-Coolant Accident (LOCA) conditions. This involved improving the FRAPTRAN-1.5 fuel performance code and applying it to simulate experimental and reactor scenarios.

**3 / 6**

## THE IPEN/CNEN (BRAZIL) CONTRIBUTION TO ATF-TS CRP

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Introduction

Since the Fukushima Daiichi nuclear accident, the vulnerability of current fuel designs under severe accident conditions has been widely recognized. As a result, alternative and advanced fuel designs that offer greater resistance to fuel failure and hydrogen production have been actively explored. In this context, at the request of Member States, the IAEA has organized several Technical Meetings[1] and Coordinated Research Projects (CRPs)[2,3] to support the design and development of Accident Tolerant Fuel (ATF) technologies.
The IPEN/CNEN participated in the two most recent CRPs, FUMAC[2] and ACTOF[3] by performing fuel performance simulations for different ATF fuel designs under both normal and accident conditions. These simulations were conducted using the FRAPCON/FRAPTRAN [4,5] and TRANSURANUS[6] codes. The objective of this work is to present an IPEN/CNEN contribution to ATF-TS CRP.
Initially, the proposed activities involved the assessment and modification of the TRANSURANUS code to account for FeCrAl cladding, enabling the simulation of separate-effect burst experiments and the development of a LOCA evaluation methodology.

This methodology was based on the IFA-650 [7] experimental series and considered various LOCA accident scenarios, including SBLOCA, LBLOCA, and DEC-A. Moreover, was included experimental work to be performed at IPEN/CNEN laboratories, specifically oxidation and burst experiments for cladding based on iron alloy. However, due to the global COVID-19 outbreak, these experimental activities could not be carried out as planned.

ATF-TS CRP Scope

As recommendation of the previous CRP´s dedicated to ATF[1,4], the CRP ATF-TS was created and starts at 2021 and still in progress with 22 member states participation aiming to support their efforts to design and development an accident tolerant for light water reactors in order to increase the safety and reliability.
This CRP consists of four different groups and objectives:
 Work task 1: is dedicated to conducting single rod and bundle experimental tests using potential candidates for Accident Tolerant Fuels. These tests aim to evaluate the performance and behavior of ATF materials under various conditions, helping to identify promising candidates for further development, as well as obtain material properties data useful for fuel performance codes.

 Work task 2: is focused to benchmark fuel performance codes by considering experiments conducted in Work task 1, as well as existing tests and data on advanced fuel, including cladding materials. By comparing simulation results with experimental data, this work task aims to improve the accuracy and reliability of fuel performance predictions.

 Work task 3: aims to evaluate Loss-Of-Coolant Accident (LOCA) methodologies using a best estimate plus uncertainty approach for nuclear power applications in order to enhance safety assessments for reactor operation and design.

 Work task 4: is dedicated to compiling a comprehensive database of available material data, properties, and correlations for ATF materials. Moreover, aims to create a valuable open-source resource for researchers and engineers. By centralizing relevant information will facilitates access to critical data needed for ATF development and implementation.

The aims of this work is to present activities related to the work task 2, which intend to contribute to benchmark the fuel performance code TRANSURANUS [6] through an experiment similar to PUZRY experiment [7] performed at Hungarian AEKI –Center for Energy Research for FeCrAl alloy cladding under framework of ATF-TS CRP. Moreover, validates existing data and properties of FeCrAl alloy from open literature [8].

ACKNOWLEDGEMENTS
This work has been carried out under the ATF-TS CRP framework and the experiment data were organized by the International Atomic Energy Agency (IAEA) in cooperation with the Atomic Energy Research Institute of the Hungarian Academy of Sciences.
The authors acknowledge and are grateful to European Commission, Joint Research Centre, Directorate for Nuclear Safety and Security (Karlsruhe, Germany) for the TRANSURANUS code license, which made it possible to perform the ATF-TS CRP activities.

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**3 / 7**

## MODELLING OF ATF MICRO-CELL UO2 FUEL BEHAVIOUR UNDER IRRADIATION Impact of thermal conductivity on expected thermal-mechanical behavior under irradiation

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Microcell CERMET (CERamic METal) is an ATF (Accident Tolerant and Advanced Technology Fuels) where granules of UO2 are enveloped in metallic channel resulting in a nuclear fuel characterized by enhanced thermal conductivity. This reduces the energy stored in the fuel thus increasing the grace period in case of an accident. Few experimental data are today available on this type of fuels. Modelling and simulation can therefore support the understanding of the expected performance under irradiation and help the optimization of this promising ATF. The actual scope of the work presented here is to develop a computational scheme dedicated to simulate the behaviour under irradiation of CERMET fuels, in nominal, incidental and accidental conditions. This paper presents the first step in the process: the effective thermal conductivity of a micro-cell CERMET is modelled by homogenization approach and then verified/validated against available experimental data and full-field calculations. Maxwell model is found to be well adapted in modelling the effective radial thermal conductivity of CERMET fuel, even though anisotropy is neglected. The model is then implemented in the ALCYONE code of the PLEIADES platform where the expected thermal-mechanical behaviour of the CERMET is studied in nominal conditions. The CERMET shows lower centreline temperatures and a lower tendency to crack. Gap closure is delayed. Future work is dedicated to modelling the mechanical behaviour of the metal additive and of the resulting CERMET fuel under irradiation

**2 / 8**

## Performance analysis of HRP IFA790 experiment regarding microcell pellet irradiation behavior

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The Korea Atomic Energy Research Institute is developing micro-cell $UO_2$ pellets as a sintering material for accident-resistant nuclear fuel. Micro-cell sintered materials are composed of cell wall material formed by creating multiple membranes within the $UO_2$. The metal microcell sintered body improves the thermal conductivity of the $UO_2$ sintered body by connecting materials with high thermal conductivity, which reduces the nuclear fuel temperature and fission gas release, thereby improving the operating margin. The irradiation test in the frame of Halden Reactor Project was conducted on the IFA790 rig to verify the in-reactor performance of micro-cell pellet. This test is part of the ongoing International Thorium Consortium, which is led by Thor Energy of

Norway. The test began in December 2015 and concluded in February 2018, with an actual test duration of around 360 days and approximately 800 days of testing including downtime. The average burnup during the test was 16.2 MWd/kgU. Real-time measurement of the fuel centreline temperature in the reactor confirmed that the micro cell pellet improved the thermal conductivity in the reactor. This study describes the results of the IFA 790 test and analyses these results using the FRAPCON code.

**2 / 9**

# INTEGRAL TESTING OF ATF FUEL UNDER HIGH TEMPERATURE ACCIDENT CONDITIONS IN THE CODEX FACILITY

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INTRODUCTION: The main objective of accident-tolerant fuel (ATF) development is to withstand high-temperature accident conditions in nuclear reactors. An important part of ATF fuel qualification involves experimental testing of fuel components in both in-pile and out-of-pile test facilities. The CODEX experimental program included two tests with ATF fuel materials: the CODEX-ATF test, conducted in a steam atmosphere, and the CODEX-ATF-AIT test, which simulated an air ingress scenario.
1. THE CODEX FACILITY
The CODEX facility had hexagonal bundles with 7 electrically heated fuel rods in both experiments. The test section was similar to the previous CODEX experiments [1][2][3]. The height of test section was 1000 mm, but the length of the rods was only 650 mm (CODEX-ATF) and 910 mm (CODEX-ATF-AIT), so the upper part of the shroud was used for thermocouples, without fuel rods. The fuel rods were filled with zirconia pellets and heated by tungsten wires. The rods were individually pressurised. In the CODEX-ATF test the seven-rod bundle consisted of three uncoated and four Cr coated optZIRLOTM cladding tubes and uncoated spacer grids and shroud were used. In case of CODEX-ATF-AIT tests all rods, spacer grids and the shroud were coated by chromium. The Zr shroud was surrounded by several thermal insulation layers, electrical heaters and a steel tube (Fig. 1). The Cr coating of components was carried out at the Czech Technical University in Prague. The main parameters of the test section are summarised in Table 1.

FIG. 1. Horizontal cross section of the test section with thermal insulations
TABLE 1. MAIN CHARACTERISTICS OF CODEX TEST SECTIONS WITH ATF FUEL
CODEX-ATF CODEX-ATF-AIT
Number of fuel rods 7 7
Cladding of fuel rods No. 2., 4. and 6. optZIRLOTM Cr coated optZIRLOTM
Cladding of fuel rods No. 1., 3., 5. and 7. Cr coated optZIRLOTM
Length of fuel rods 650 mm 910 mm
External diameter of fuel rods 9.1 mm 9.1 mm
Pellet material inside of the rods ZrO2 ZrO2
Pellet material in the bottom of the rods Al2O3 Al2O3
Spacer grid material Zr1%Nb Cr coated Zr1%Nb
Number of spacer grids 2 3
Shroud material Zr2.5%Nb Cr coated Zr2.5%Nb

Shroud thickness 2 mm 2 mm
Length of shroud 1000 mm 1000 mm

The online gas composition measurement at the test section outlet was conducted using a quadrupole mass spectrometer. Gas sampling at the outlet was achieved by inserting a sampling tube into the off-gas pipe. The measurement system recorded system and rod internal pressures, flow rates, outlet gas composition, values of input and output power, coolant inlet and outlet temperatures and rod temperatures. Several high temperature thermocouples were built into the surface of fuel rods, shroud and insulation layers at different elevations.
2. THE CODEX-ATF EXPERIMENT
The CODEX-ATF simulated a high temperature nuclear power plant severe accident terminated by water quench. The test focused on the observation of fuel failure and degradation mechanisms.
In the preparatory phase the facility was heated up to 600 ℃ in 0.2 g/s steam and 0.2 g/s argon flow rates using both external heaters and fuel rod heaters. The heat-up phase continued with the same flow rates and with 1000 W heating power on the rods and 800 W power of external heaters. The cladding burst took place at ≈900 ℃ on most of the rods. The temperature increase was very smooth. During the quench phase, room temperature water was injected to the bottom of the test section, when the cladding temperature in the top of the bundle was above 1600 ℃. In the upper part of the fuel rods 1400 ℃ was reached (Fig. 2.).

FIG. 2. Cladding temperatures in the CODEX-ATF test FIG. 3. Cladding temperatures in the CODEX-ATF-AIT test

The total hydrogen production during the experiment was about 3 g, which indicated significant oxidation of the Zr components. Intense Zr-Cr eutectic formation took place at these temperatures on the external surface of Cr coated cladding tubes. The post-test examination showed large deformation and failure of both coated and uncoated cladding tubes (Fig. 4.).

FIG. 4. Bundle cross section (left), damaged uncoated (centre) and Cr-coated (right) cladding tubes removed from the top of CODEX-ATF bundle

FIG. 5. Oxide and nitride formation on the cladding of CODEX-ATF-AIT bundle

1. THE CODEX-ATF-AIT EXPERIMENT The main objective of the CODEX-ATF-AIT test was to check if accident tolerant fuel (ATF) cladding with Cr coating would have a protective role in case of NPP severe accidents with air ingress. The simulated scenario was a reactor accident with corium melting through the bottom head of the pressure vessel and penetration of air+steam mixture from the reactor cavity into the reactor vessel. The experiment was focused on covering several phenomena of fuel behaviour during accidents (burst, oxidation, nitriding, eutectic formation). Slow cool-down was selected to provide information on the state of the bundle before quench. In the preparatory phase the facility was heated up to 600 ℃ in steam –argon flow using both external heaters and fuel rod heaters. The pre-oxidation phase continued with the same flow rates and with stepwise increased heating power on the rods. During the heat-up phase the rods were pressurised and cladding burst took place between 750-800 ℃ on most of the rods. The opening allowed the coolant to enter the rods and start chemical reactions on both sides of the cladding. In the intermediate cool-down phase the temperatures were reduced below 750 ℃. In the steam-air phase the temperature increase was very smooth compared to the reference experiment CODEX-AIT-3 [2]. The reason of the slower temperature increase was the protective effect of the Cr coating. The duration of air ingress phase in the reference experiment was 1 hour with 1600 ℃ maximum cladding temperature, and it was 1.5 hour in the CODEX-ATF-AIT with 1545 ℃ maximum cladding temperature (Fig. 3.). The temperature profile significantly changed during the air ingress phase similar to the reference experiment: the maximum temperature moved from the upper section of the bundle to lower elevations due to the intense chemical interactions in the less oxidised and/or melted lower part. The maximum shroud temperature was similarly around 1470 ℃ at 300, 500 and 700 mm elevations.

The outlet gas composition showed that during the air ingress phase steam and oxygen starvation conditions were established. The partial consumption of nitrogen indicated the formation of nitrides as well. The total of 1.4 g hydrogen was produced during the pre-oxidation phase and 3.3 g of hydrogen during the air ingress phase. Slow cool-down of the bundle was performed in argon flow in order to avoid interactions that might take place during water quench. At the lower half of the bundle the main degradation mechanism of rods was the Cr-Zr eutectic melt formation and its fast oxidation/nitriding in the last phase of the experiment. From 600 mm upwards, all the pressurised rods were ballooned and the Cr coating became cracked on their surfaces (Fig.5.).

SUMMARY

Two integral tests with ATF cladding materials were conducted in the CODEX facility under steam and steam-air atmospheres (Table 2). The high-temperature behavior of the cladding materials in both tests demonstrated reduced hydrogen production, a lower heat-up rate, and longer coping times, highlighting the advantages of ATF materials. Cr-Zr eutectic formation was observed in both tests, and its role in cladding failure was identified. The experimental data are available in an electronic database for code development and validation purposes.

TABLE 2. MAIN CHARACTERISTICS OF CODEX TEST SECTIONS WITH ATF FUEL

| | CODEX-ATF | CODEX-ATF-AIT |
|---|---|---|
| Max. temperature | 1655 ℃ | 1545 ℃ |
| Steam atmosphere | Yes | Yes |
| Steam+air atmosphere | No | Yes |
| Oxidation | Yes | Yes |
| Nitriding | No | Yes |
| Cr-Zr eutectic | Yes | Yes |
| Quench | Yes | No |

ACKNOWLEDGEMENTS

The CODEX-ATF test was conducted within the framework of the IAEA ATF-TS project, while the CODEX-ATF-AIT test was carried out as part of the EU OFFERR project. Both tests were supported by the Paks Nuclear Power Plant. The main parameters of the scenarios were selected based on pre-test calculations performed by Pál Kostka and Gábor Lajtha (NUBIKI), Kirill Dolganov (IBRAE), Thorsten Hollands (GRS) and Líviusz Lovász (GRS).

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**3 / 11**

## IAEA ATF BENCHMARK OF FUEL ROD CODES FOR SIMULATION OF SELECTED SEPARATE EFFECT TESTS

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Since 2021, IAEA has been organizing a Coordinated Research Project (CRP) on Testing and Simulations of Accident Tolerant and Advanced Technology Fuels (ATF-TS). A work task (WT2) is dedicated to benchmark the computer codes used for ATF behaviour simulation. The WT includes ATF specific modelling developments in fuel performance codes and simulation benchmark exercises using these codes to predict ATF fuel rod behaviour in normal operating conditions, power ramps and accident conditions. Participation in this task is a prerequisite for the WT3 of the CRP (LOCA safety evaluation methodology), as it is a sort of code validation phase for investigating different ATF solutions. The WT is divided into two sub-tasks: single rod effects tests (WT2.1) which is the object of the

paper, and bundle tests (WT2.2).
In the single rod sub-task, the following fuel concepts and experimental cases are considered:
- Chromia-doped UO2 pellets, using Halden tests IFA677.1 and IFA716.1 for steady state conditions, and SCIP II power ramps for transient conditions,
- FeCrAl and chromium coated zirconium cladding, using separate effects tests performed in the experimental WT of the CRP (burst tests),
- Cr-coated Zircaloy-4 single rod test performed in the DEGREE facility.

**2 / 13**

## ZIRCONIUM ALLOY CLADDING WITH CHROMIUM COATING: DIFFUSION-REACTION MECHANISMS AND STRUCTURAL DEGRADATION UNDER EXTREME TEMPERATURES

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Zirconium alloys have long served as fuel cladding in light water reactors due to their excellent mechanical properties and low neutron absorption cross-section. However, during loss-of-coolant accidents (LOCAs), they react violently with high-temperature steam. Chromium-coated zirconium alloys are a leading candidate for accident-tolerant fuels (ATFs), yet their protective capability fails when temperatures exceed the Cr-Zr eutectic point (~1332℃). At such temperatures, Cr rapidly diffuses into the Zr matrix, forming a ZrCr2 layer (several micrometers thick), which thins the coating, initiates cracks, and embrittles the substrate.
Simultaneously, the Cr2O3 oxide layer accelerates spallation due to liquid-phase eutectic breakdown. The synergistic effect of Cr diffusion and steam oxidation at high temperatures results in a "crocodile-skin" oxidation morphology on the cladding surface, with oxidation rates surpassing those of uncoated zirconium alloys.
This study establishes a Zr-Cr diffusion mechanism model to quantitatively reveal Cr concentration profiles, ZrCr2 layer growth kinetics, and defect formation mechanisms, providing theoretical support for failure prediction and emergency strategies in extreme conditions.
Experimental Methods & Key Findings :
(a) 1250℃: No ZrCr2 layer formed within 10min; after 60min, a 3.29μm layer emerged, with Cr coating thickness significantly reduced (~40μm diffusion depth). This indicates that low temperatures require prolonged diffusion to accumulate critical concentrations.
(b) 1350℃: A 0.64 μm ZrCr2 layer formed in 30s, increasing to 1.69μm at 90s. Cr coating thickness remained stable, demonstrating better structural integrity under high-temperature, short-duration conditions.
(c) 1400℃ ~1482℃: ZrCr2 layer thickness saturated with prolonged holding (growth rate declined after 150s). Cracks and voids appeared at the coating-substrate interface due to diffusion-induced volume expansion and stress accumulation.
(d) 1482℃: The ZrCr2 layer was continuously consumed, with extended holding failing to thicken it or even causing its disappearance. This reveals a critical diffusion temperature and the dissolution mechanism of Cr atoms into the Zr matrix post-coating depletion.
(e) Driving Forces & Failure Mechanisms: ZrCr2 growth is driven by concentration gradients, thermodynamic negative free energy, and thermal stress fields. Kirkendall voids, lattice distortion, and grain boundary defects are the primary causes of coating failure.

**1 / 14**

## ATF CLADDING MATERIALS AND MANUFACTURING WITHIN IAEA ATF-TS CRP

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ATF CLADDING MATERIALS AND MANUFACTURING WITHIN IAEA ATF-TS CRP

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INTRODUCTION
This paper describes the materials and manufacturing processes of ATF concepts experimentally tested within the ATF-TS coordinated research project (CRP) from 2020 until 2025. Only cladding materials were tested and no fuels were provided to any participant. There are two possible development options for ATF claddings [1]:
- Improve currently used Zr-based alloys (e.g., deposition of protective coatings)
- Replace Zr-based alloys with alternative advanced materials

The CRP participants focused mainly on the improvement of traditional Zr-based materials using thin coatings on the outer surface, namely Cr-based coatings on commercial Zr-based alloys such as Optimized ZIRLOTM (Opt.ZIRLO), Zircaloy-4, and Zr-1%Nb. In addition, other innovative materials such as FeCrAl alloys and SiC/YAG composites were fabricated and provided for testing. Reference uncoated Zr-based materials were provided for parallel testing to provide a baseline for comparison with ATF materials when available. An overview of ATF materials tested is presented in Table 1.
The ATF and reference cladding materials were generously provided by the participants for testing. Fabrication processes and their optimizations were performed internally by material providers and were outside the scope of the CRP. It should also be noted that materials were provided in different geometries (plates and tubes; variations in outer diameter and length, etc.) and quantities. Some institutes provided only 1-2 cm long segments, others were able to provide tubes up to 1.5 meter long. In addition to the coated Zr-based alloys, two types of FeCrAl cladding were also tested. The first was provided by KIT and was a reference material for the QUENCH-19 test, and the second was FeCrAl provided by NPIC with different composition.
TABLE 1. SUMMARY OF ATF CLADDING MATERIALS TESTED WITHIN THE ATF-TS CRP AND THEIR PROVIDERS
Institute Country Coated Zr-based Cladding Alternative materials Notes
Czech Technical University in Prague Czech Republic Cr and CrN-Cr coated Opt. ZIRLOTM Tubes, large quantities, up to 1.5 meter long
Karlsruhe Institute of Technology Germany Cr-coated ZIRLO, M5 and Zircaloy-4 FeCrAl B136Y3 – 13Cr-6.2Al-0.03Y (wt.%) Tubes, large quantities, up to 0.5 meter long
Institute of Nuclear Chemistry and Technology Poland SiC/YAG coated Zr1Nb Plates, optimization of deposition process not completed
Canadian Nuclear Laboratories Canada TiAl-coated Zircaloy-4 CANDU reactors as a focus, tubes, large quantities
Nuclear Power Institute of China China Cr-coated Zircaloy-4 FeCrAl 13Cr-4.5Al-2Mo-1Nb-0.4Ta-0.05Y (wt%)
China Nuclear Power Engineering China Cr-coated Zircaloy-4
Belarusian State University Belarus Cr-coated and plasma-treated E110 Tubes, small quantities, optimization of fabrication process not completed
Atomic Energy Organization of Iran Iran Cr and CrN/Cr coated Zr1Nb Tubes, large quantities, coatings optimized during the project
KEPCO Nuclear Fuels Korea Cr coated HANA-6 Tubes, large quantities, coatings optimized

CHROMIUM COATED ZIRCONIUM-BASED ALLOY
Pure Cr coatings with the highest level of technology readiness were the most popular among ma-

terials providers. Six institutes provided coated Zr claddings with pure Cr coatings –CTU, KIT, CNPE, NPIC, AEOI, and KNF. The coating thickness and deposition parameters vary significantly, resulting in different microstructures and performance of the tested coatings. In addition, different Zr-substrates (commercial Zr-based cladding materials) were used by different institutes.
The Czech Technical University in Prague (CTU) developed three types of chromium coatings for Opt. ZIRLO™ and Zircaloy-4 cladding materials using the Hauzer Flexicoat 850 and 1500 machines. These coatings include a thick Cr coating (17-20 μm), a thin Cr coating (~7 μm) and a combination of CrN (approximately 6 μm) and an outer Cr coating (~12 μm). The deposition process involves cleaning, plasma cleaning, and physical vapor deposition (PVD) with specific parameters to ensure high-quality coatings. The resulting microstructure of these coatings is shown in Figure 1 [2].
Karlsruhe Institute of Technology (KIT) applied Cr coatings to Zircaloy-4, M5 and ZIRLO cladding tubes using magnetron sputtering. The process included ion etching with argon ions and coating at a maximum temperature of 270℃. Two coating thicknesses were used: 7.5 μm and 10.5 μm. Quality control was performed using destructive testing and adhesion tests according to German standards.
China Nuclear Power Engineering Co., Ltd. (CNPE) provided two types of Cr-coated cladding: a Zr-Sn-Nb alloy coated with magnetron sputtering and a Zry-4 alloy coated with arc ion plating. The coating microstructure analyses showed smooth surfaces and uniform chromium distribution for the magnetron sputtered samples, whereas the arc ion plated samples showed small droplets and thinner coatings.
The China Nuclear Power Institute (NPIC) used multi-arc ion plating to coat Zircaloy-4 cladding tubes. The process involved mechanical polishing, ultrasonic cleaning, and deposition with specific parameters to reduce droplet formation and improve the density of the coating. The resulting coatings exhibited fiber or columnar crystal morphology.
The Atomic Energy Organization of Iran (AEOI) prepared several batches of Cr-coated Zr-1%Nb samples using cathodic arc evaporation. The process included ultrasonic cleaning, argon ion sputtering, and deposition with specific parameters. The coatings showed good adhesion and uniform thickness, with a clear boundary between the Cr coating and the substrate. The deposition chamber used for coating preparation and resulting coating microstructure are shown in Figure 2.
And KEPCO Nuclear Fuels (KNF) used Arc Ion Plating (AIP) to coat the HANA-6 cladding tubes. The process involved plasma etching, Cr ion deposition, and post-coating inspections to ensure quality. The coated cladding materials were provided to various laboratories for detailed testing.

FIG. 1. Cr coatings produced by CTU on Opt. ZIRLO –SEM/EBSD analysis of the microstructure

(A)
(B) (C)
FIG. 2. (A) Arc-PVD chamber used by AEOI to deposit Cr coatings; (B) SEM-BSE image of the as-deposited Cr coating; (C) Fracture surface of Cr-coated Zr-1%Nb cladding tube

ZIRCONIUM-BASED ALLOYS WITH ALTERNATIVE COATINGS
Pure Cr is the most advanced solution for coated Zr-based claddings, but there are still several drawbacks such as Cr-Zr eutectic reaction at about 1330 ° C, Cr-enhanced embrittlement of Zr, low ductility of Cr, etc. Therefore, alternative coating designs and materials have been proposed and are being optimized. Four alternative coated cladding materials were fabricated within ATF-TS by CTU, AEOI, CNL, and INCT. The alternative coatings are CrN/Cr, TiAl, and SiC/YAG.
CTU developed CrN/Cr coated Opt. ZIRLO to enhance the corrosion resistance and reduce the interdiffusion of Zr-Cr. The CrN interlayer was chosen on the basis of its ability to reduce the kinetics of eutectic formation and improve high-temperature steam oxidation resistance. The deposition process involved reactive magnetron sputtering for the CrN layer, followed by a Cr top layer. The resulting microstructure showed a submicrometer Cr layer to improve adhesion, with a grain size and texture similar to that of pure Cr coatings.
AEOI prepared Zr-1%Nb tubular samples with CrN and multilayer Cr/CrN coatings. The coating process involved physical vapor deposition, with pre-treatment including ultrasonic cleaning and argon ion sputtering. The coatings were characterized by SEM and GI-XRD, revealing the presence of Cr, CrN, and Cr2N phases. The Cr2N phase formed due to nitrogen gas flow manipulation during the coating process. The coating parameters for both multilayer Cr/CrN and single layer CrN are shown in Table 2.
TAB.2. COATING PARAMETERS USED FOR CRN AND MULTILAYER COATING BY AEOI
Parameter Coating thickness (μm) Number of targets Bias voltage (V) Substrate temperature (℃) Ar gas pressure (sccm) Periodic injection of Nitrogen gas (min) Coating duration (h)

CrN Coating 7–10 2 100 300 ± 2 15 ~55.0* ~7.0
multilayer Coating 7–10 2 100 300 ± 2 15 ~4.0 ~8.5
* To control shedding and decrease residual stress of CrN coating

CNL provided Ti/Al coated Zircaloy-4 tubes with different outer diameters relevant to CANDU reactors for high-temperature oxidation and burst tests. The coating was applied by magnetron sputtering deposition, with a target thickness of 10 µm and a composition of 45-50 at.% Ti to 55-50 at.% Al. The resulting coatings showed good adhesion and uniform thickness, with some variations due to the deposition rate. The resulting microstructure observed by SEM and XRD pattern are shown in Figure 3.
(A)
(B)
FIG. 3. (A) TiAl cross-section with gaps between coating and the substrate for the 13.08 mm OD Tube due to Surface Roughness; (B) XRD Pattern from a 5.3 mm OD Zr-4 Tube, Coated with Ti/Al.

The Institute of Nuclear Chemistry and Technology (INCT) developed SiC/YAG composite coatings for Zr-1%Nb substrates using suspension plasma spraying (SPS). The coatings aimed to improve corrosion resistance and high-temperature stability. The SiC grains were coated with YAG using a sol-gel method, and the resulting powders were sprayed onto the substrates. The coatings were characterized by SEM and EDS, revealing a fine-grained structure with good adhesion. However, the initial deposits showed some delamination, indicating the need for further optimization. The SEM observations of the outer surface and cross section are shown in Figure 4.
(a) (b) (c) (d)
FIG. 4. Plasma sprayed coatings on Zr-1%Nb plate, spray distance 80 mm, SEM (SE) image, a) argon, b) argon-hydrogen and cross-section of the coatings sprayed by argon-hydrogen plasma SEM (BSE) image with magnification: c) 1 000, d) 4 000

FECRAL
KIT provided nuclear-grade FeCrAl alloys, specifically B136Y3 (Figure 5), originally provided by Oak Ridge National Laboratory (ORNL) and previously used for the QUENCH-19 bundle test [3]. These alloys were fabricated in tubular shapes with an outer diameter of 9.52 mm and a wall thickness of 381 µm. NPIC on the other hand developed a FeCrAl alloy with a composition of 13Cr-4.5Al-2Mo-1Nb-0.4Ta-0.05Y (wt%). This alloy was designed to improve the adhesion of the oxide protective layer and strengthen the Fe-bcc matrix. The alloy underwent homogenization, solution treatment, hot extrusion, and cold pilger rolling to achieve the final cladding tube dimensions of 9.5 mm outer diameter and 0.38 mm wall thickness. The processing flow included vacuum induction melting, homogenization, red feeding forging, extrusion, warm rolling, cold rolling, and recrystallization annealing.
Nuclear-grade FeCrAl alloys B136Y3 were provided by ORNL in tubular shapes. The tubes had an outer diameter (OD) of 9.52 ±0.04 mm; the wall thickness was 381 ±9 µm. Fig. 2.24 shows an optical micrograph after etching with a solution of glycerin of 45 ml, HNO3 of 15 ml and HCl of 30 ml. The grains are fairly uniformly distributed with a grain size of 80 ~ 130 µm. The chemical composition (Febal Cr13 Al6.2 Y0.03 C0.01) was provided by ORNL.

FIG. 5. Microstructure of the FeCrAl B136Y received

NPIC provided FeCrAl with chemical composition - 13Cr-4.5Al-2Mo-1Nb-0.4Ta-0.05Y (wt%). The addition of Y, Mo, Nb, and Ta aims to improve the adhesion of the oxide protective layer and strengthen the Fe-bcc matrix. The FeCrAl alloy was subjected to homogenization and solution treatment, followed by hot extrusion and cold pilger rolling with a thickness reduction of about 40% for four times to obtain the cladding tube. The FeCrAl ingot billet prepared by HIP (hot isostatic pressed) was then subjected to homogenization and solution treatment. Cold pilgered cladding tube was obtained by hot extrusion and pilger rolling with a thickness reduction of ~40% for four times and followed by annealing at 800℃ for 30 min to reduce stress and defects and prevent premature cracking during rolling. The final geometry of the cold pilgered thin-wall tube are 9.5 mm in outer diameter and 0.38 mm in wall thickness.
On the basis of high-temperature compression experiments on FeCrAl alloy, the influence of deformation parameters (deformation temperature, strain rate) on high-temperature deformation behaviours was studied. The Arrhenius type model was used to establish the constitutive equation and thermal processing diagram of the FeCrAl alloy, and the influence of thermal deformation parameters on the effect of the material microstructure. By studying the recrystallization behavior, texture evolution, Laves phase precipitation, and its interaction with grain boundaries during the processing and heat treatment of FeCrAl alloy cladding pipes, the quality of FeCrAl pipe processing and microstructure was well controlled. In order to improve the service performance of the pipe,

the introduction of kink strips effectively improved the strengthening and toughening effect of the alloy, the red forging process was used to optimize the hot processing technology of the tube blank, and the warm and cold rolling process was used to optimize the pipe rolling technology, successfully overcoming the problem of containing Laves phases. In order to solve the problems of high hardness and difficult processing of FeCrAl alloy, a method combining a warm/cold deformation process is used to achieve fine control of the structure and properties of FeCrAl cladding material and realize batch production processing of full-size defect-free FeCrAl alloy thin-walled cladding tubes.
The processing flow is as follows:
Vacuum induction melting → homogenization → red feeding forging → extrusion → warm rolling (three passes) → cold rolling (seven passes) → recrystallization annealing → final product.
(a) (b)
FIG. 6. Microstructure of FeCrAl (a) after pilger rolling and (b) after annealing
SUMMARY
This paper provides a detailed description of the ATF fuel cladding materials manufactured and tested within the ATF-TS CRP. The primary materials include chromium-coated zirconium-based alloys, chromium-nitrogen-/chromium-coated zirconium-based alloys, and FeCrAl alloys.
Chromium coatings were applied to zirconium-based alloys mainly to improve their oxidation resistance and mechanical properties. Various deposition techniques, such as magnetron sputtering and arc ion plating, are used to create thick and thin Cr coatings, as well as CrN/Cr multilayer coatings. These coatings are characterized by their uniform thickness, good adhesion, and enhanced high-temperature performance.
CrN/Cr coatings are developed to reduce the interdiffusion of Zr-Cr and improve corrosion resistance. The deposition process involves reactive magnetron sputtering for the CrN layer, followed by a Cr top layer. These coatings exhibit a fine-grained structure and good adhesion, making them suitable for high-temperature steam oxidation testing.
FeCrAl alloys were designed to enhance the high-temperature oxidation resistance and mechanical properties of the nuclear fuel cladding. These alloys undergo processes such as homogenization, hot extrusion, and cold pile rolling to achieve the desired microstructure and mechanical properties. The addition of elements such as Y, Mo, Nb, and Ta improves the adhesion of the oxide protective layer and strengthens the Fe-bcc matrix.
Based on the experience of ATF-TS fabrication, the following recommendations can be made:
- Optimization of Coating Processes: More research should focus on optimizing the deposition parameters for chromium and CrN/Cr coatings to enhance their performance under reactor conditions. This includes refining the thickness, composition, and microstructure of the coatings to achieve better adhesion and corrosion resistance.
- Development of New Materials: Exploring new materials and composite coatings can provide additional benefits in terms of corrosion resistance and high-temperature stability. The integration of these materials into existing cladding technologies should be investigated to improve overall performance.
- Testing and Validation: Extensive testing and validation of the coated cladding materials under simulated reactor conditions are essential. This includes high-temperature oxidation tests, mechanical property evaluations, and long-term durability assessments to ensure coatings can withstand the harsh environments of nuclear reactors.
- Collaboration and Knowledge Sharing: Continued collaboration between research institutions, universities, and industry partners is crucial to advance the development of advanced coatings for nuclear fuel cladding. The sharing of knowledge and best practices can accelerate the adoption of these technologies and improve their effectiveness.
By addressing these recommendations and focusing on future research directions, the development of advanced coatings for the cladding of nuclear fuel can significantly enhance the safety and efficiency of nuclear reactors, contributing to the sustainable and reliable generation of nuclear energy.
ACKNOWLEDGEMENTS
The activities presented in the paper were conducted within the framework of the IAEA ATF-TS project, and we would like to acknowledge the contribution of many coworkers, students, and advisors who help to fabricate, optimize, and share the ATF and reference cladding materials.
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**2 / 15**

# SEPARATE EFFECT TESTS WITH ATF CLADDING MATERIALS WITHIN IAEA ATF-TS CRP

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SEPARATE EFFECT TESTS WITH ATF CLADDING MATERIALS WITHIN IAEA ATF-TS CRP

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INTRODUCTION
This paper describes the separate effect tests performed on ATF cladding materials within the ATF-TS coordinated research project (CRP) [1]. Work Task 1 (WT1) of the ATF-TS CRP focused on experimental testing of ATF claddings with high technology readiness levels under both accident and normal operating. This programme included subtasks on ATF fabrication, bundle tests at QUENCH, DEGREE, and CODEX facilities, and separate effect tests (SETs). The plan for the WT1 SETs started with a selection of candidate materials. First, CRP members and partners proposed a range of different materials that have different levels of technological readiness and, as such, are available in different quantities and geometries. The limitations given by the geometry and quantity of testing materials were taken into account when designing the testing campaign. There were also test priorities defined by material providers that had to be taken into account, as well as political constraints resulting from the wide participation of member states [2, 3].
In the second step, CRP members proposed to conduct testing campaigns using their facilities. The original idea was to define a fixed set of testing parameters and perform round-robin exercises (RRT) with the available ATF materials. However, it was soon recognized that each institute has testing capabilities (e.g., temperature range, manpower, geometry limitations) which would substantially reduce the scope and number of participants in the WT. Therefore, a broader experimental scope was defined and tuned to allow wide participation but still produced valuable results. The objective was to coordinate sample providers and experimentalists to produce valuable data - i.e. not to repeat what was done in the ACTOF CRP or other projects/complementary testing, if feasible/RRTs, if feasible/support of future benchmarks (fuel performance ballooning/burst, DEGREE tests, QUENCH tests).). The constraints and strategy of WT1 are illustrated in Figure 1.
FIG. 1. Strategy and limitations in the WT1 SET experimental
The following priorities for the SETs were identified:
• Testing to support the following bundle tests, their interpretation, pre- and post-test calculations:
o QUENCH-19 with FeCrAl B136Y
o DEGREE DBA and DEC tests with Cr-based coated Zircaloy-4 (PWR geometry)
o CODEX-ATF with Cr-coated Opt. ZIRLO (VVER geometry)
• Derivation of correlations for fuel performance codes:
o LOCA-relevant ballooning/burst properties –creep and burst criteria (stress- and strain-based)
o Models for high-temperature oxidation (DBA and BDBA range) and corrosion kinetics (normal steady-state operation)
o Mechanical and Thermal Properties
• Generation of validation cases that can be used to validate models in single rod codes
o Separate effect test with Cr-coated Zr and FeCrAl
• Compilation of data for the IAEA Fuels and Materials database
o Derivation/fitting of experimental data produced by several laboratories.
o Sharing ATF properties and models

EXPERIMENTAL TESTING
In total, 14 institutes and laboratories participated in the experimental investigations within the ATF-TS CRP. All tests can be categorized into seven subtasks, as shown in Figure 2. The scope was first proposed by the members and later adapted based on the available materials and priorities presented in Chapter 1. It should also be noted that there was a different availability of recourses at

various laboratories, and as a result, there is a big difference between the final scope. Some members performed hundreds of tests, and some were able to handle only a few.

FIG. 2. Classification of experiments performed within ATF-TS WT1

Table 1 shows the list of experimental measurements in participating laboratories. Most of the tests have been completed until summer 2024 and then documented in the TECDOC that is now in the publication process. Testing continued also in 2025 in different laboratories and will be presented in the IAEA Technical Meeting or different conferences. In total, more than 160 meters of reference and ATF tubes and several m2 of reference and ATF plates were shared and tested.[4, 5]

TABLE 1. LIST OF INSTITUTES/LABORATORIES INVOLVED IN THE EXPERIMENTAL TESTING AND THEIR FOCUS WITHIN THE ATF-TS CRP
Institute Country Experiments performed
Czech Technical University in Prague/UJP Praha Czech Republic Ballooning/burst tests, HT steam oxidation, long-term VVER corrosion
JRC Karlsruhe European Commission Thermophysical properties
JRC Petten European Commission Mechanical tests in the LWR environment
CRIEPI Japan Single rod and bundle tests
Karlsruhe Institute of Technology Germany Single rod and bundle tests
Institute of Nuclear Chemistry and Technology Poland Neutron activation analysis, long-term corrosion, HT annealing
China Nuclear Power Engineering China Long-term corrosion PWR, HT oxidation, ballooning / burst
Atomic Energy Organization of Iran Iran HT oxidation
China Nuclear Power Technology Research Institute China Axial Tensile Tests, Internal Burst Tests, Internal Climb Tests, Autoclave Corrosion Tests, HT Steam Oxidation
EK, Centre for Energy Research Hungary Ballooning/burst tests, Mandrel tests, HT steam oxidation, bundle test
Universidad Politécnica de Madrid Spain Cathodic charge, Hydriding, Ring compression tests, HT creep
Korea Atomic Energy Research Institute Korea Microstructural Investigations
Seoul National University Korea LOCA tests, DSC, and post-quench ductility evaluation

HIGH TEMPERATURE CREEP, BALLOONING AND BURST TESTS
Due to different limitations, the tests are not standard RRTs, but rather complementary tests of several cladding candidates. However, RRTs were possible in several instances, the laboratories and their test matrices are summarized in table 2.
TAB.2. CONTRIBUTORS TO THE BALLOONING/BURST TASK
Institute Country Experiments performed
CTU/UJP Praha Czech Republic Isobaric tests with temperature ramp (0.7K/sec and 7K/sec) + isothermal isobaric creep tests with coated Zr
EK, Centre for Energy Research Hungary Constant temperature –pressure ramps (100 kPa/s) with coated Zr and FeCeAl
CNPRI China Pressure ramp tests at 350ºC with reference and coated Zr
UPM Spain Pressure ramp tests at 600 ° C with reference and coated Zr
CNPE China Pressure ramp tests at 800-900 ° C with reference Zr coated and coated
SNU Korea Single rod ramp tests simulating first phase of LOCA

The conclusions of this test series can be summarized as [6, 7]:
- Coated Zr cladding
o Limited benefit of coatings in time-to-burst, generally within the uncertainty range
o Ballooning and opening size coated Zr provide some benefits, but
the extra margin depends on the specific test conditions and methodology.
o New creep correlation and burst criteria derived (both stress- and strain-based for coated cladding) and provided to WT2 for model implementation and validation. Correction coefficients proposed for fuel performance models.
o In most cases, the reference Zr materials behave similarly to the coated Zr given ±20% uncertainty.
- FeCrAl
o A limited volume of material was available, but a very different performance was discovered for FeCrAl.
o Fuel performance codes were unable to predict this behavior because the models were historically

developed and tuned for Zr-based materials.
Dozens of experimental datasets were measured and prepared for the IAEA Fuels and Materials database. Further analysis is foreseen, as well as usage of the data for code validation, and the data are open to the wide fuel performance community.
HIGH-TEMPERATURE OXIDATION TESTS
A reduced high-temperature steam oxidation rate is a key advantage of ATFcladdings. This section presents an overview of the experimental results on the oxidation rates of various coated zirconium alloys, considering the types of base zirconium alloys, coating materials and methods, and coating thicknesses. The post-LOCA ductility of the Cr-coated cladding tubes tested is evaluated using Ring Compression Tests to assess the potential impact of Cr-coated zirconium alloys on the limits of the emergency core cooling system. Additionally, the formation of a Zr-Cr eutectic mixture, a critical safety challenge associated with Zr-Cr systems, and its influence on the structural integrity of Cr-coated cladding, was examined by tests conducted above 1330℃. The institutes contributing to this task are summarized in table 3.
TAB.2. CONTRIBUTORS TO THE BALLOONING/BURST TASK
Institute Country Experiments performed
CNPRI Czech Republic HT double-sided steam oxidation up to 1400 ℃

EK, Center for Energy Research Hungary Small scale steam oxidation test
CTU/UJP China Pressure ramp tests at 350 ° C with reference and coated Zr
INCT Spain Ar annealing test
CNPE China Small scale steam oxidation test
SNU Korea LOCA testing; PQD evaluation
KIT Germany Single rod tests
CRIEPI Japan Single rod tests

The highlights of this test series can be summarized as follows.
- Coated Zr cladding
o Eutectic Cr/Zr –another degradation mechanism (e.g. similar to secondary hydriding or oxygen embrittlement), not hard limit (melting point). Effect of coating thickness observed.
o The ECR criteria for coated Zr do not have physical meaning and should be replaced. But it is conservative and can be used by industry if needed.
o CrN/Cr –benefits in BDBA conditions confirmed by CRIEPI, KIT. However, open questions remain, but it seems like a feasible way to further improve pure Cr coatings.
- FeCrAl
o Ramp rate sensitive oxidation kinetics, improved model derived.
o Transition of oxidation kinetics around 1375 ° C leading to rapid material degradation.
AUTOCLAVE CORROSION TESTS
Long-term corrosion tests were conducted on zirconium alloys with various coatings, focusing on their performance in simulated nuclear reactor environments. The research was carried out by multiple institutions, including UJP, CNPRI, INCT, and UPM.
The long-term corrosion behaviour of ATFs is a critical aspect of the safety and efficiency of nuclear reactors. The primary objective was to evaluate the performance of these coatings in simulated environments (PWR, WWER) over extended periods. Experiments are also included under extended conditions to accelerate corrosion processes (steam 400 ℃). The tests were conducted in autoclaves, simulating the primary water chemistry of pressurized water reactors (PWRs) and WWER reactors.

FIG. 3. AUTOCLAVES, SAMPLE HOLDERS, AND SAMPLES TESTED WITHIN THIS ATF-TS

The results indicate that chromium-based coatings significantly improve the corrosion resistance of zirconium alloys. However, the long-term performance of these coatings is influenced by factors such as coating defects and the deposition process.
SUMMARY
WT 1 of the ATF-TS CRP focused on the experimental testing of various ATF cladding materials. This work task is fundamental for all other work tasks within ATF-TS CRP, namely WT2 –ATF fuel performance modelling and code validation; WT3 –LOCA methodology and uncertainty quantification with ATF materials; WT4 - open-source database similar to ATF MATPRO.
Eight institutes from IAEA member countries fabricated and shared different ATF cladding concepts, and in addition, three types of reference commercial Zr-based alloys were tested in the same campaigns to provide a baseline for performance comparison. 14 institutes proposed and completed experimental tests with the ATF and reference cladding materials provided. The experimental tests can be divided into ballooning/burst tests (7 institutes); HT oxidation tests or annealing (10 institutes);

long-term autoclave corrosion tests (5 institutes); mechanical tests (4 institutes); other tests (4 institutes) and bundle tests (3 institutes). Despite significant challenges, namely the COVID pandemic, complicated political relationships between various members, and logistical/administrative complications with cladding materials, WT1 was able to generate several hundreds of unique datasets that have been uploaded to the IAEA Fuel Experimental Data Repository and opened to all interested partners from all over the world. In addition, WT1 produced new correlations and models for fuel performance codes (creep, strain/stress failure criteria, mechanical and corrosion models, and thermal properties) that were shared with WT2. Based on these new models, benchmark cases were defined to validate these new models in fuel rod or integral codes. The results of extensive experimental tests were also used to derive multiplication factors and uncertainty ranges for the LOCA methodology in WT3, and lastly, the results are available for WT4 to be processed and implemented in the new MATPRO-like ATF database.
In summary, the experimental programme within ATF-TS was the most extensive and successful CRP in the IAEA's history in the fuel area. Some of the activities are still ongoing, and many new partnerships have been created. More details on individual experimental contributions from WT1 participants are presented in separate papers within the IAEA Technical Meeting. The complete results and their comparison will be presented in a dedicated TECDOC to be published at the beginning of 2026 by the IAEA.
ACKNOWLEDGEMENTS
The activities presented in the paper were conducted within the framework of the IAEA ATF-TS project, and I would like to acknowledge the contribution of many coworkers, students, and advisors who help to fabricate, optimize and share the materials, conduct the experiments, and evaluate the results.
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**3 / 16**

## MANDREL TEST SIMULATION USING THE TRANSURANUS CODE: INSIGHTS FROM AN OFFERR PROJECT

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The paper presents the work carried out within the framework of the OFFERR European User Facility Network, aimed at evaluating the mechanical behaviour of unirradiated Cr-coated and uncoated E110-like cladding under mandrel loading conditions. Experiments were conducted at 25 and 300℃,

and the cladding response was preliminary simulated using the TRANSURANUS fuel performance code. Two boundary condition strategies were investigated: (i) imposing the mandrel-cladding contact pressure obtained from 3D finite element analyses, and (ii) prescribing the cladding inner wall deformation. The results highlight the need for improved material models and demonstrate the limitations of 1D approaches in capturing complex mechanical responses under non-axisymmetric loading.

**4 / 17**

## Towards RIA simulations of ATF fuel in large and small modular water cooled reactors by means of TRANSURANUS

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In the frame of various Euratom and other international projects, (enhanced) accident tolerant fuel properties (ATF) have been implemented in the TRANSURANUS fuel rod performance code of the JRC. The impact of ATF has been addressed in different studies by means of the TRANSURANUS code. In the first study the focused was on steady-state performance in a case of the coordinated research project FUMEX-II organized by the IAEA. In a second study, this was extended by means of a design accident base study in a small modular reactor of the Nuscale-type in the frame of the McSAFER project utilizing fresh fuel. We used the Serpent2 code for Monte Carlo neutronics, SUBCHANFLOW code for thermal hydraulics, and the TRANSURANUS code to model the thermal mechanics of all the individual fuel rods. We successfully simulated two rod ejection scenarios and analysed the effect of some ATF. These results are briefly outlined here and complemented with a more detailed stand-alone TRANSURANUS simulation of a commercial irradiated fuel rod submitted to a RIA from the FUMEX-III coordinated research project of the IAEA. The RIA analysis of this FK1 rod was carried out both with the standard materials as well as the same ATF from the RIA simulations in the McSAFER paper.

**4 / 19**

## Advancements in hydrogen-based transient cladding strain limits to support accident tolerant fuel

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ADVANCEMENTS IN HYDROGEN-BASED
TRANSIENT CLADDING STRAIN LIMITS
TO SUPPORT ACCIDENT TOLERANT FUEL

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INTRODUCTION: As the nuclear industry strives to support accident tolerant fuel (ATF), high burnup operation, extended cycle lengths, power uprates, and other major programs for pressurized water reactor (PWR) plants, an unintended consequence is a reduction in fuel performance margin, particularly for transient cladding strain (TCS). Historically, the U.S. Nuclear Regulatory Commission (NRC) has imposed a 1% uniform strain limit on zirconium-based cladding during overpower transients. However, this fixed limit does not account for material-specific behaviour, especially under varying hydrogen content and irradiation conditions. To address this, Westinghouse, in collaboration with the Pressurized Water Reactor Owners Group (PWROG), developed a hydrogen-based, performance-driven strain limit model for existing low recrystallized zirconium-based cladding. However, newer ATF cladding materials have been developed since the initial hydrogen-based strain limit was approved. Some of these materials, such as highly recrystallized zirconium-based cladding, behave differently from existing cladding alloys and cannot utilize the hydrogen strain limit in its current form. An improved limit, once again being developed by Westinghouse in collaboration with the PWROG, aims to recover margin for Westinghouse zirconium-based cladding, including ATF products. This advancement will enable the use of advanced fuel products and technologies for high energy fuel (HEF) applications and increased fuel duty.

1 BACKGROUND

During steady-state operation, stress and strain on the cladding are due to the natural contact pressure between the fuel pellet and the zirconium-based cladding. Fuel rods are designed with an initial gap between the pellet and cladding to accommodate this contact pressure.

During steady-state operation, the pellet-cladding gap closes over time due to the solid swelling of the fuel pellets and the compressive pressure differential causing the cladding to creep down. This contact pressure between the pellet and cladding results in stress/strain, which is typically benign and insufficient to cause fuel failure. However, transient conditions such as overpower events can rapidly increase power, causing thermal expansion of the fuel pellet at a faster rate than the cladding material. This heightened pressure can lead to strain-induced deformation of the cladding, potentially causing fuel failure and releasing fission byproducts into the reactor coolant. The strain applied to the cladding during overpower accidents may be exacerbated by ATF and low enriched uranium plus (LEU+) products. As an example, increased fuel density provides a neutronics benefit because it allows for more 235U in the core; however, a denser pellet may also have increased swelling and subsequently higher cladding strain. Changes in the cladding alloy may improve resistance to corrosion at high burnup, but it can come at the expense of material strength. Development of advanced fuel products requires a balance between numerous, often competing, factors.

Historically, Westinghouse fuel has adhered to a constant 1% cladding strain limit during overpower transient events, as prescribed by the U.S. NRC Standard Review Plan (SRP) Section 4.2 [1]. This limit does not explicitly account for various factors impacting material strength, such as cladding properties, manufacturing processes, irradiation, temperature, and hydrogen uptake. Westinghouse, in collaboration with PWROG, has developed an alternative hydrogen-based strain limit [2]. A hydrogen-based strain limit derived from measured test data provides a more accurate representation of zirconium-based cladding behaviour and improves strain margin.

1.1. Hydrogen and Material Strength

Measurements and mechanical testing have shown that the yield strength (YS) and ultimate tensile strength (UTS) of zirconium-based cladding has a strong dependence on hydrogen content.

During reactor operation, zirconium cladding reacts with the coolant to produce zirconium oxide ($ZrO_2$) and free hydrogen ($H_2$). A fraction of the hydrogen is absorbed into the cladding, while the rest disperses into the coolant. Hydrogen pickup in the cladding is cumulative, and more is absorbed the longer the fuel rod remains in operation. If enough hydrogen is absorbed, it can exceed the terminal solid solubility (TSS) limit. Excess hydrogen above the TSS is precipitated as ZrH platelets. These zirconium hydrides accumulate near the outer cladding wall due to the temperature gradient across the cladding. Hydrides form crack propagation pathways which cause ductility degradation in the cladding and a corresponding decrease in UTS.

1.2. Hydrogen-Based Strain Limit

Westinghouse developed a uniform plastic elongation (UEPlastic) strain limit for zirconium-based fuel cladding based on measured hydrogen and strain data [2]. In this context, UEPlastic corresponds to the plastic strain component of the UTS. Unlike the uniform 1% limit, the UEPlastic strain limit explicitly accounts for the effects of hydrogen embrittlement while continuing to ensure the integrity of the fuel during overpower transient events.

Comparing between the NRC SRP 1% strain limit [1] and the Westinghouse hydrogen-based UEPlastic strain limit. Until the hydrogen reaches the saturation limit, the UEPlastic strain limit is held steady at a value greater than 1%. After the saturation limit, when excess hydrogen is present in the cladding in the form of circumferential hydrides, the strain limit steadily decreases at an exponential decay.

This hydrogen-based strain limit increases margin relative to the 1% limit as long as significant hydrogen is not absorbed into the fuel cladding during operation.

2 DISCUSSION

The hydrogen strain limit from [2] was only developed and licensed using the cladding alloys available at the time. Since then, Westinghouse has developed new ATF products which are capable of operating for longer residency times and burnup up to 75 GWD/MTU due to their increased resistance to corrosion (both oxidation and hydrogen pickup). Improved corrosion resistance increases the benefit of a hydrogen-based limit by reducing the free hydrogen accumulated in the cladding during operation. However, advanced fuel products do not always exhibit the same behaviours as legacy cladding alloys.

For example, newer alloys with higher recrystallized annealing (RXA) are more ductile than older alloys with less RXA. Preliminary tests show that high RXA alloys have almost no UEPlastic strain even though they have significantly higher total elongation relative to legacy fuel. Thus, a UEPlastic strain limit like that developed in [2] is overly conservative for high RXA cladding.

2.1. Extension of a Hydrogen-Based Transient Cladding Strain Limit to ATF Products

Westinghouse and the PWROG are investigating use of the total (elastic plus plastic) uniform elongation (UETotal) instead of UEPlastic as a method for determining the strain limit for ATF products. UETotal increases the allowable strain relative to UEPlastic which the cladding can undergo during an overpower accident without exceeding the UTS. This is a significant improvement in strain margin, particularly for high RXA cladding where the UEPlastic is near zero. Allowing for both plastic and elastic strain, as opposed to just the plastic component, is consistent with the requirements of the NRC SRP [1]. UETotal has also been presented as a more appropriate measure for calculating the cladding strain limit than UEPlastic at a recent American Society for Testing and Materials (ASTM) symposium on zirconium.

Westinghouse is currently partnering with Oak Ridge National Laboratory and the Studsvik testing facility in Nyköping, Sweden to perform additional testing, some first-of-a-kind, on ATF products. The tests are design to validate that UETotal is a viable design limit for cladding strain during an overpower event and will serve as a supplement to the existing measured database. Once the new data is available, Westinghouse and the PWROG intend to submit a supplement to the original hydrogen-based strain topical report [2] which would extend use of a UETotal strain limit for ATF products.

3 CONCLUSION

ATF features are critical to enabling longer residency times, burnup operation up to 75 GWD/MTU, LEU+ fuel, power uprates, and other major programs. Transient cladding strain is currently one of the most limiting fuel performance design criteria for PWR plants with ATF products because they cannot take advantage of the current hydrogen-based strain limit. Application of a new UETotal limit, however, allows utilities to take advantage of ATF products without sacrificing design margin for transient cladding strain. Providing strain margin relief to ATF products is an important step in supporting the future operation of PWR plants.

ACKNOWLEDGEMENTS
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**4 / 20**

## Constraints Involved in Safety Review of Accident Tolerant Fuels - Pakistan's Perspective

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In the era of rapidly advancing technologies, the capacity of regulatory bodies to conduct thorough and timely safety reviews has become increasingly critical. Approaches to evaluate novel nuclear fuel technologies vary across regulatory bodies; however, a shared objective remains to assess the safety benefits of advanced fuels, particularly Accident Tolerant Fuels (ATFs) and to shape future review strategies accordingly. Despite the existing presence of conventional nuclear fuels within the industry, the accelerating pace of innovation in advanced fuel development demands enhanced review expertise. Engaging with this emerging domain further demands to ensure the country-specific regulatory oversight. ATFs represent a significant advancement in the nuclear fuel technology. Through advanced materials and innovative design improvements in both fuel cladding and fuel pellets, ATFs offer superior performance characteristics. The development of ATFs introduces meaningful enhancements to nuclear safety by strengthening the safety margins, such as the core overheating. These technological upgrades make ATFs a promising choice for future reactors, seeking improved accident response capabilities.
As the global nuclear industry evolves toward the use of higher fuel burnup rates and increased uranium enrichment levels, the adoption of ATFs will require substantial regulatory assessment. For regulatory bodies, this means investing in deeper technical understanding, refining analytical methods, and establishing clear pathways for evaluation to ensure that ATFs meet both national and international safety standards. Currently, ATF is not in use within Pakistan's nuclear infrastructure. However, it seems that the ATF will be a promising candidate for next-generation nuclear fuels, likely to become globally viable in upcoming years. Therefore, PNRA is committed to preparing for future regulatory demands by staying updated on the latest technical progress and safety considerations. As many nations continue to make progress in this emerging field of ATFs, the Pakistan Nuclear Regulatory Authority (PNRA) is also actively working to understand and assess its potential.
In the light of growing international interest in ATFs, PNRA has initiated a capacity building project to explore the advancements in this emerging field and to proactively identify associated intricacies involved in the safety review of ATFs. To support this capacity building initiative at PNRA, a dedicated team was formulated in 2022 and the said project is in-progress.
A core component of our capacity building project involves conducting a gap analysis to assess our existing regulatory capabilities and to determine its relevance to future applications for advanced nuclear fuels. The findings aim to result in informed decision making for safe and responsible deployment of ATFs, and for the targeted regulatory enhancements to strengthen safety of future reactor operations.
Knowledge gained through this technical meeting will have a critical role in enhancing the ongoing capacity building project at PNRA and to learn from the experiences of the other countries. Taking advantage of the participation in this forum, we aim to further perform a gap analysis between our regulatory infrastructure and the global regulatory framework.

**2 / 22**

## Investigation of the temperature-dependent failure processes in a PVD Cr-coated ZIRLO nuclear fuel cladding material using real-time X-ray micro-tomography imaging

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Zirconium alloys (Zircaloys) are widely used as fuel cladding in LWRs but suffer rapid oxidation in high-temperature steam, compromising structural integrity. Cr coatings, especially those applied via physical vapour deposition (PVD) method, offer improved oxidation resistance and mechanical stability. According to open literature, the 3D real-time failure process of such material under mechanical loading at extreme temperatures remains poorly understood.
This study investigates the deformation and fracture of one PVD Cr-coated Optimized ZIRLO® cladding material under C-ring compression loading using real-time synchrotron X-ray computed tomography (XCT) at temperatures up to 950 ℃ in argon atmosphere, and post-tested SEM and EBSD analyses were performed to support XCT observations and assess temperature-dependent damage mechanisms.

**2 / 23**

## The study on the interdiffusion performance of Cr-Zr under high temperature in Cr-coated cladding

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Abstract: The Cr-coated cladding has excellent corrosion resistance and good bonding performance with zirconium alloy matrix, and has become the most promising material selection for ATF cladding in engineering applications. Due to the fact that the Cr-coated cladding is obtained by preparing a layer of Cr coating on the outer surface of the traditional zirconium alloy cladding, the intermediate phase formed by the diffusion of Cr-Zr interface at high temperature may cause the coating to crack or peel off, and the diffusion behavior of Cr-Zr at high temperature will also consume the thickness of the Cr coating, thereby affecting the service performance of the Cr-coated cladding. It is necessary to study the interdiffusion behavior of Cr-Zr at high temperature in order to understand the high-temperature performance changes of Cr-coated cladding and promote its engineering applications.
To study the diffusion behavior of Cr-Zr at high temperature and obtain the structural changes, diffusion layer thickness changes, and Cr coating consumption rate during Cr-Zr interdiffusion, This report conducted annealing tests on Cr-coated cladding at different temperature.The experimental results indicate that as the annealing temperature increases, the thickness of the Cr-Zr intermediate diffusion layer will increase, and larger thickness of intermediate diffusion layer will cause internal cracking and lead to coating failure.This report is based on the residual Cr coating thickness and intermediate layer thickness of samples annealed at different temperatures, and obtains the diffusion activation energy and factor of Cr coating, which can provide support for the subsequent calculation of Cr coating consumption rate.

**5 / 25**

## BWR perspective of industrial applications of ATF (high burnups, FFRD, time @ temperature)

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In an age that advanced nuclear reactors are at the forefront of the research community, we should not forget that future of the nuclear industry heavily depends on the safe, reliable, economic operation of the current LWR fleet. With the advances in modeling and simulation techniques, methodologies, tools, and with the significant operating experience, LWR fleet capacity factor and reliability have improved significantly over the last few decades. That said, changing economic priorities, development, and advancement of competing technologies, the LWR industry has been continuously investigating new initiatives using innovative solutions to further improve fleet's economics as well as overall safety of the plants. However, these new opportunities often present technical and regulatory challenges. Experts from LWR industry will discuss these challenges as well as the new opportunities identified while addressing these challenges via application of innovative solutions, such as for the fuel fragmentation relocation and dispersion (FFRD), dose consequences, and time at temperature (t@T) concepts.

**2 / 27**

## Integral Irradiations and Post-Irradiation Examinations of ATF Concepts Performed at Idaho National Laboratory

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INTRODUCTION: In recent years, Idaho National Laboratory (INL) has performed both irradiation and post-irradiation examination (PIE) tests to support development and deployment of advanced technology fuel (ATF) concepts. The paper overviews test irradiations under pressurized water reactor (PWR) conditions at both commercial and test reactors. PIE was performed on rodlets fabricated by Framatome and irradiated in the ATF-2 water loop at INL's advanced test reactor (ATR), as well as on lead test rods fabricated by Westinghouse and irradiated at the Byron generating station. Selected results from both are presented here.

**2 / 28**

## CORROSION KINETICS OF AUSTENITIC STAINLESS STEEL UNDER WATER NORMAL OPERATION PWR CONDITION

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In the context of accident tolerant fuel (ATF) development, long-term corrosion tests were conducted on a niobium-stabilized austenitic stainless steel (Nb-ASS) to assess its oxidation resistance under normal water operation conditions in a pressurized water reactor (PWR) environment. Due to the limited availability of corrosion data for this alloy in such conditions, two independent experimental

campaigns were carried out. The specimens were exposed to ultrapure, deoxygenated water at 360 ℃ and 20 MPa for approximately 84 days, with sampling intervals of 21 days, enabling the modeling of weight gain kinetics.
Under these conditions, the oxidation behavior of stainless steels is typically characterized by the formation of a dual-layer oxide scale, composed of an outer magnetite layer and an inner chromium-rich spinel (chromite) layer. After X-ray diffraction (XRD) analysis confirmed the presence of both phases on the tested samples, a bilayer model consisting of equal proportions of magnetite and chromite was adopted to estimate the oxide thickness.
This study presents a methodology for determining oxide layer thickness from mass gain data and compares the results with published values for Zr-, Fe-, and Ni-base alloys. Additionally, oxidation kinetics were analyzed assuming a solid-state diffusion mechanism governed by a parabolic rate law. The calculated kinetic parameters further support the high oxidation resistance of the Nb-ASS alloy compared to conventional Zr-base alloys. Its behavior also closely resembles that of widely used austenitic stainless steels, such as AISI 304, reinforcing its potential as a promising ATF cladding candidat.

**5 / 29**

## Current status of development and qualification of advanced technology fuel (ATF) in Russia

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Global trends in the development of nuclear power plants are related to the increased electricity generation, improved efficiency, and the environmental attractiveness of nuclear power generation, and are aimed at:
- ensuring the reliability and safety of nuclear power plants, including the development and introduction of new fuel modifications;
- improving the economic efficiency of the fuel cycle at all stages, including the transition to a closed nuclear fuel cycle and minimising radioactive waste for disposal;
- increasing the duration of fuel cycles (up to 24 months).
The fuel company TVEL JSC, part of the Rosatom State Corporation, develops and manufactures nuclear fuel and its components for power and research reactors, Atomflot reactors, and Floating thermal nuclear power plant. Historically, TVEL has produced nuclear fuel for reactors developed in the Soviet Union and Russia. Currently, TVEL also produces and supplies fuel and fuel components for power and research reactors developed in other countries. More than 70 power reactors and 30 research reactors in thirteen countries operate on TVEL fuel. Thus, TVEL is one of the leading global suppliers of nuclear fuel.

**1 / 35**

## Development Status of FeCrAl-ODS Cladded Accident Tolerant Fuel for BWRs

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The FeCrAl-oxide dispersion strengthened (ODS) alloy is a promising candidate alloy for the accident tolerant fuel (ATF) cladding of light water reactors (LWRs) and has been recently developed in Japan. This paper reports on the progress of the development of accident tolerant FeCrAl-ODS fuel claddings for boiling water reactors (BWRs) in Japan.
Both experimental and analytical studies were conducted to evaluate the influence of implementation of the FeCrAl-ODS fuel cladding to the current BWRs.
In the experimental study, key material properties of FeCrAl-ODS fuel cladding have been obtained and accumulated to support the evaluations in the analytical study. Effect of neutron irradiation on mechanical properties were also evaluated by irradiation tests at a test reactor.
In the analytical study, in order to evaluate the influence of implementation of the FeCrAl-ODS cladding to the current BWRs, the core characteristics and the fuel behavior were evaluated in the analysis study at the normal operating condition. The merits of applying FeCrAl-ODS cladding to BWRs were preliminary evaluated by the MAAP 5.05b code with and without accident management.
Although research and development efforts are steadily advancing toward practical implementation, further research and development is still needed. At the end of the presentation, to proceed the practical implementation of the FeCrAl-ODS fuel cladding, the challenges and perspectives found in the program will be outlined.

Part of this study is the result of "Development of Technical Basis for Introducing Advanced Fuels Contributing to Safety Improvement of Current Light Water Reactors" program carried out by JAEA under the project on technical development for improving nuclear safety, supported by Ministry of Economy, Trade and Industry (METI) of Japan.

**3 / 37**

# STATUS OF FUEL AND THERMAL HYDRAULIC COUPLED ANALYSIS IN KOREA

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The behavior of fuel and thermal-hydraulics is strongly interdependent during transient conditions. In particular, during a Loss of Coolant Accident (LOCA), the cladding temperature and internal rod pressure increase significantly due to the loss of effective heat removal by the coolant. This can lead to severe cladding deformation. If the cladding undergoes large deformation or rupture, it may obstruct the coolant flow, thereby degrading the reactor's coolability and potentially leading to a severe accident. Accordingly, it is critical to evaluate and quantify the consequences of design basis accidents such as LOCA with respect to established safety criteria. To address the complex interactions between fuel and thermal-hydraulic behavior in such scenarios, various multiphysics coupling studies have been conducted in Korea. Notably, integrated analysis systems such as MARS-KS/FRAPTRAN, CUPID/FRAPTRAN, CUPID/MERCURY, and MARS-KS/CUPID have been developed to simulate these coupled phenomena. This study provides an overview of the current status of multiphysics-coupled analysis research in Korea. Coupled analysis approaches have proven to be valuable tools for improving the understanding of fuel behavior under accident conditions. Since fuel behavior is highly sensitive to the prevailing thermal-hydraulic environment—and conversely, thermal-hydraulic behavior is influenced by fuel deformation (e.g., through changes in flow channel geometry and local heat generation)—a coupled analysis framework enables more accurate prediction by providing realistic boundary conditions for both domains.

**1 / 38**

## FRAMATOME PROGRAMS TO SUPPORT US PWR OPTIMIZATION

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**4 / 39**

## SUMMARY OF NEA WORKING GROUP FOR FUEL SAFETY'S TECHNICAL OPINION PAPER ON ATF

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See attachment

**3 / 40**

## IAEA ATF-TS benchmark for simulation of bundle tests

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IAEA ATF-TS Benchmark FOR simulation of bundle tests

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INTRODUCTION: As part of the IAEA ATF-TS project, not only numerous single rod tests were carried out with ATF materials, but also two bundle tests with Cr coated claddings made of Zr alloys:

the DEGREE-B3 bundle test at CRIEPI [1] and CODEX-ATF test at HUN-REN EK [2]. The advantage of bundle tests lies in the creation of prototypical adiabatic conditions and the possibility of studying the mutual influence of fuel rods. In addition, such integral tests are a good basis for verification and validation of computer codes. Therefore, it was decided to conduct a benchmark within the framework of this IAEA project using experimental data obtained both during tests and in post-test studies. In addition to the two bundle tests using chromium-coated zirconium claddings, it was proposed to also use the results of the QUENCH-19 bundle test with FeCrAl claddings previously conducted at FZK. Conducting benchmark for the QUENCH-19 test was initiated within the framework of the previous IAEA ACTOF project [3], but then only two research organizations managed to take part in this project. Now the range of organizations involved has been significantly expanded.
BENCHMARK ON THE QUENCH-19 BUNDLE TEST PERFORMED WITH FECRAL CLADDINGS
The QUENCH-19 bundle experiment with 24 B136Y cladding tubes and 4 Kanthal AF spacer grids as well as 7 KANTHAL APM corner rods and KANTHAL APM shroud was conducted at KIT on 29th August 2018 [4]. This was performed in cooperation with the Oakridge National Laboratory (ORNL). The test objective was the comparison of FeCrAl(Y) and ZIRLO claddings under similar electrical power and gas flow conditions. The experiment was performed in four stages. The electrical power supply was the same as in the reference test QUENCH-15 (ZIRLO) during the first two stages (pre-oxidation and transient). The third stage with constant electrical power was performed to extend the temperature increase period. The test was terminated at peak cladding temperature of about 1460 ℃ by water flooding similar to QUENCH-15. The total hydrogen production was 9.2 g (47.6 g for QUENCH-15).
Seven organizations provided results for exercises on the modelling of the QUENCH-19 bundle test (Table 1).
TABLE 1. ORGANIZATIONS AND CODES PARTICIPATED IN THE QUENCH-19 BENCHMARK
Participant CNEA
Argentina CTU
Czech Republic GRS
Germany IBRAE
Russia KIT/INR Germany NINE
Italy UPM/NFQ
Spain
Code DIONISIO MELCOR ATHLET-CD SOCRAT ASTEC MELCOR MELCOR
For almost all codes, the rod bundle was described by three concentric rings as shown in Fig. 1: an inner ring (ROD1) containing four central rods, a second ring containing eight intermediate rods (ROD2), and a third ring containing twelve peripheral rods (ROD3). When modeling with the MELCOR code, NINE and CTU applied a division into two groups of fuel rods: internal and external rods. Only one central rod was modelled with the DIONISIO code. The corner zirconium rods used for the bundle instrumentation were taken into account by their effect on reducing the flow area of the assembly. In addition, their outer surface area was taken into account when calculating the hydrogen release due to their oxidation. Also, when calculating the hydrogen release, the influence of the inner surface of the shroud was taken into account.

FIG. 1. Composition of the QUENCH-19 bundle
According to the benchmark conditions, each code had to calculate - based on specified boundary conditions and experimental data on the temporary change in electrical power supplied to the bundle - the temperature history at each of the seventeen elevations of the assembly. In addition, the most important parameter for comparing the efficiency of codes should have been the calculated value of hydrogen release.
QUENCH-19: Comparison of temperature predictions
Based on the readings of the thermocouples of the central and intermediate rods, an axial distribution of temperatures in the inner bundle part was obtained 300 s before the start of the reflood, namely at the time of 8800 s (in a later period, a number of thermocouples failed). Comparison of these experimental data with the results of calculations shows a good prediction of the position of the maximum temperature at the bundle elevation of 850 mm by most codes (Fig. 2). Below this level, the data from the four codes practically coincide with the experimental data. Above 850 mm, the data of the two codes coincide with the measured values. The other two codes give overpredicted temperature values.
Comparison of calculated temperatures with experimental ones at the bundle elevation of 950 mm throughout the experiment shows overestimated values for all codes - satisfactory for the first (Fig. 3) and second (Fig.4) groups of rods and significantly overestimated for the shroud (Fig. 5). The latter

circumstance may be due to insufficient consideration of the steam-water mixture entering through leaks into the space between the shroud and the cooling jacket surrounding it [4].

FIG 2. Axial temperature profiles for QUENCH 19 FIG. 3. Temperature progress for internal rods of the QUENCH-19 bundle

FIG. 4. Temperature progress for external rods FIG. 5. Temperature progress for shroud
QUENCH-19: Comparison of hydrogen predictions
When metal M is oxidized in steam, hydrogen is released, the release rate of which is determined by the degree of oxidation:
xM + yH2O = MxOy + yH2 (1)
The enhanced oxidation resistance of FeCrAl alloys at high temperatures relies on the formation of a slowly growing and highly protective $Al_2O_3$ scale [5]. The formation of a protective alumina scale is determined by the competition between the oxidation rate governed by diffusion of O and Al through the oxide layer and the diffusion of aluminium in the substrate to the interface. Alumina performs its protective role at temperatures below approximately 1650 K. At higher temperatures, accelerated diffusion processes lead to increased Fe oxidation, leading to a catastrophic increase in the oxidation rate. Based on the results of oxidation experiments performed at MIT with the B136Y3 samples (FeCrAl alloy used for the QUENCH-19 claddings) [6], the following correlations for the parabolic rate constant of the sample mass gain have been proposed to use for all codes:
K_MIT [g^2/(〖cm〗^4 s)]={⊠(9.62×10^(-12), &T≤1473 K@A_B exp((-E_B)/RT), &1473 <T<1648 K@A_Fe exp((-E_Fe)/RT), &T≥1648 K (melting point of FeO))⊠ (2)
where the activation energies E_B=594354 J/mol and E_Fe=352513 J/mol, the pre-exponential constants A_B=3×109 g2/cm4s and A_Fe=2.4×106 g2/cm4s.
It should be noted that more detailed experiments carried out later at KIT [7] showed more precise results with the following kinetics for this alloy (derived from data published in [7]):
K_KIT [g^2/( 〖cm〗 ^4 s)]={⊠(A_L exp((-E_L)/RT), &873 <T<1173 K (transient alumina)@4.69·10^(-14), 1173 ≤T≤1273 K@A_H exp((-E_H)/RT), &1273 <T<1648 K (α-alumina)@A_Fe exp((-E_Fe)/RT), &T≥1648 K (melting point of FeO))⊠ (3)
where the activation energies ⊠L=184729 J/mol, ⊠H=287748 J/mol and ⊠⊠⊠=352513 J/mol, the pre-exponential constants ⊠L=5375.6*10-8, AH=6*10-2 and ⊠⊠⊠=2.4×106 g2/cm4s.
A comparison of the two oxidation correlations presented in Fig. 6 shows that the correlation obtained from the MIT data is more conservative and thus gives a more conservative estimate for the hydrogen release.

FIG. 6. Comparison of two parabolic rate constants for mass gain during oxidation of the B136Y3 alloy
For the oxidation of KANTHAL alloys (used for shroud and corner rods), it is proposed to use the following correlation established in the temperature range 1323 < T < 1749 K for the KANTHAL APMT alloy [8]:
〖K_A=A〗 _A exp((-E_A)/RT) (4)
where the activation energies E_A=344000 J/mol, the pre-exponential constants A_A=7.84 g2/cm4s.
All benchmark participants used the proposed oxidation correlations with minor individual adjustments for better matching when transitioning between different temperature intervals. Comparative results for the simulated integral hydrogen release are presented in Fig. 7.

FIG. 7. Simulation results for hydrogen release during the QUENCH-19 bundle test
A very good prediction for the total mass of released hydrogen was given by two codes, which is primarily due to the fairly accurate calculation of bundle temperatures by these codes. The deviation in the prediction of total hydrogen release by other codes is due to either increased calculated temperatures (overprediction of hydrogen) or individual modification of the oxidation correlation for iron (underprediction of hydrogen).
Benchmark on the DEGREE-B3 bundle test performed with Cr coated Zry-4 claddings
The DEGREE-B3 bundle experiment with nine Zircaloy-4 cladding tubes with 235 mm length (provided by KIT/Karlsruhe and PVD coated to 20 µm Cr layer by CTU/Prague) was conducted at CRIEPI on 11th April 2023. The inductive heated test bundle was oxidised in a flow of steam/Ar gas mixture under transient conditions up the peak cladding temperature of 1350 °C and then cooled in Ar. Before testing, all nine rods were pressurised with He to 6 MPa and showed symmetrical ballooning and burst during the test, with the middle at the hottest bundle elevation of 135 mm.
Four organizations provided results for exercises on the modelling of the DEGREE-B3 bundle test (Table 2).
TABLE 2. ORGANIZATIONS AND CODES PARTICIPATED IN THE DEGREE-B3 BENCHMARK

Participant KIT/INR
Germany CRIEPI
Japan IBRAE
Russia
Code ASTEC FRAPTRAN SOCRAT
DEGREE-B3: Comparison of temperature and burst predictions
According to experimental data, the induction heating power during the preparatory stage was 2 kW, then during the bundle heating stage it increased to 23 kW for 430 s, after which the induction heating was turned off. Tungsten rods installed in the center of each of the nine fuel rod simulators were used as susceptors. However, the alternating magnetic field also excited eddy currents in the cladding tubes, i.e. some of the heat was also generated in the claddings, and not just in the center of the fuel elements. Different considerations of this fact by different codes may cause differences in predictions of thermohydraulic effects.
Since the thermocouples in the bundle were installed at the 125 and 180 mm elevations (below and above the burst positions), temperature simulations were performed for these bundle elevations. The corresponding calculated data are presented in Figs. 8 and 9. The temperature escalation predicted by the SOCRAT code at 125 mm should be associated with the diffusion of chromium into the zirconium matrix, leading to the disappearance of the protective chromium layer and accelerated oxidation of zirconium at this temperature [5, 9].
A detailed account of the dependence of the mechanical properties of the cladding on temperature in the SOCRAT code made it possible to quite accurately predict the burst temperature of the central cladding and the corresponding pressure decrease inside this rod (Fig. 10). The value of this parameter (about 840 K) corresponds to the burst temperatures observed for uncoated Zircaloy-4 claddings [10].

FIG 8. Clad temperatures of central rod at 125 mm FIG. 9. Clad temperatures of central rod at 180 mm

FIG. 10. Pressure progress inside the central rod of the DEGREE-B3 bundle
DEGREE-B3: Comparison of hydrogen predictions
The oxidation by steam of the chromium results in the formation of a well-adherent and protective Cr2O3 layer and a certain amount of hydrogen is released:
2Cr+3H2O=Cr2O3+3H2 (5)
The growth kinetics of these oxides can be described by parabolic correlations according to the following correlations:
Cr2O3 thickness (derived from the data in [9]) δ [m]=2.63· 〖10〗 ^(-3)·e^(-119747/(R·T))·√t (6)
Cr2O3 mass gain Δm [kg/m^2 ]= 〖δ·ρ〗 _Cr2O3·(3M_O)/M_Cr2O3 =4.327·e^(-119747/(R·T))·√t (7)
where density of chromia ρCr2O3=5210 kg/m³, molar mass of oxygen MO=16, chromia MCr2O3=152.
These correlations are valid up to a temperature of 1332 ℃, after which chromium diffusing into the zirconium matrix forms the Cr/Zr eutectic melt. Taking into account correlation (7), codes FRAPTRAN and ASTEC obtained the hydrogen release shown in Fig. 11. Since FRAPTRAN is a single-rod code, the common hydrogen release from the nine-fuel bundle was calculated by multiplying by 9. Calculations with the SOCRAT code showed an excess of the threshold temperature of 1332 ℃, so after reaching this value, the standard Cathcart-Pawel correlation was used for the oxidation of Zircaloy-4 (Fig. 12). From the two presented figures it is clear that taking into account only the oxidation of chromium leads to an underestimation of hydrogen, while inclusion of the kinetics of zirconium oxidation too early gives an overestimated result.

Fig 11. Hydrogen release without Zr oxidation Fig. 12. Hydrogen release with Zr oxidation
Benchmark on the CODEX-ATF bundle test performed with Cr coated ZIRLO claddings
The CODEX-ATF bundle experiment with seven electrically heated rods, having opt. ZIRLO cladding tubes with 650 mm length, 9.1 mm outer diameter and 0.58 mm wall thickness, was conducted at HUN-REN EK/Budapest on 11th August 2023. The bundle composition included one centre rod, six peripherical rods, two Zr1%Nb grids with the pitch of 12.75 mm. Four claddings were PVD coated to 20 µm Cr layer by CTU/Prague, three other cladding tubes were not coated. The bundle was surrounded by Zr2.5%Nb hexagonal shroud. During the test, the bundle was pre-oxidized in Ar/steam atmosphere (each gas 0.2 g/s). The test was terminated by water quench with the water injected from the bundle bottom with the flow rate of 10 g/s.
Four organizations provided results for exercises on the pre- and post-test modelling of the CODEX-ATF bundle test (Table 3).
TABLE 3. ORGANIZATIONS AND CODES PARTICIPATED IN THE CODEX-ATF PRE-TEST CAL-

CULATIONS
Participant CNEA
Argentina GRS
Germany IBRAE
Russia NUBIKI
Hungary
Code DIONISIO ATHLET-CD SOCRAT ASTEC
pre-test + + +
post-test + + +
CODEX-ATF: Comparison of CODEX-ATF temperature predictions
According to the pre-test specification, the pre-oxidation should be performed at 1000 W bundle power and 800 W power of the shroud heater. The accelerated last transient stage should last 200 s with the bundle power increased to 2000 W. However, commissioning tests carried out after pre-test calculations showed that an increased temperature growth can occur without increasing the bundle power. Therefore, it was decided to carry out the pre-oxidation and accelerated transition stages at a bundle power of 1000 W with an increased duration of the entire experiment. Of course, this led to a deviation of the temperature history from the calculated values. Comparison of calculated temperatures of the central rod with experimental ones at the hottest bundle elevation of 550 mm throughout the experiment shows underestimated values for all codes (Fig. 13).
The post-test calculations showed much more correct results. However, all codes did not reproduce the temperature escalation before quench.

pre-test calculations for central rod ATHLET-CD post-test

SOCRAT post-test DIONISIO post-test
FIG. 13. Temperature progress for the central CODEX-ATF rod at the hottest bundle elevation
CODEX-ATF: Comparison of hydrogen predictions
Because of the seven rods, only four cladding tubes were coated with chromium, the oxidation correlations presented in Chapter 3.2 were applied only to them. In the SOCRAT calculations, the hydrogen produced by the oxidation of coated claddings is due not only to the oxidation of the Cr, but also to the oxidation of the underlying Zr. For the cladding of the three remaining rods and the inner surface of the shroud, the standard Cathcart-Pawel correlation was used or in SOCRAT case, a mechanistic model for Zry oxidation was used. As a result, predictions of the hydrogen release rate were obtained, presented in Fig. 14. Due to differences in temperature predictions, there is a noticeable scattering in the hydrogen release rate prediction for the whole bundle even before the temperature escalation begins. The noticeable jump in the hydrogen release rate prediction by the SOCRAT code at t≈14250 s is associated with the switch from the chromium oxidation model to the zirconium oxidation model upon reaching the Cr/Zr eutectic point (1332 ℃).

Whole bundle (coated and uncoated claddings, shroud) Only coated claddings
FIG. 14. Hydrogen production rates predicted for the CODEX-ATF
The corresponding integral hydrogen releases are presented in Table 4 and Fig. 15. While ATHLET-CD overestimates the hydrogen release by a factor of two (due to overpredicted temperatures for not coated claddings), the DIONISIO code underestimates the integral hydrogen release by a factor of three (temperatures were underestimated). The SOCRAT code showed the result closest to the measured values (more accurate temperature prediction and consideration of oxidation of zirconium substrates in coated tubes). A comparison of the calculated data on hydrogen release by the oxidation of zirconium and chromium shows that the predominant amount of hydrogen is associated with the oxidation of bundle parts made of Zr alloy not protected by a Cr coating.
TABLE 4. INTEGRAL HYDROGEN RELEASE (IN GRAMS)
Experiment ATHLET/GRS DIONISIO/CNEA SOCRAT/IBRAE
2.91 Zr oxidation: 4.96 Zr oxidation: 1.02 Zr oxidation: 1.57
Cr oxidation: 0.33 Cr oxidation: 0.02 Oxidation of coated claddings: 0.91
total: 5.29 total: 1.04 total: 2.48

FIG. 15. Integral hydrogen release predicted for the CODEX-ATF
Conclusions
Benchmarks for simulating bundle experiments with ATF cladding materials, organized within the framework of the IAEA ATF-TS project, showed a good possibility of adapting codes for new materials. While the thermal-hydraulic parameters of the experiments were calculated using algorithms already built into the codes, the oxidation modules were modified to take into account the correlations of FeCrAl and Cr oxidation. The oxidation of FeCrAl included the entire operating temperature

range, while the behaviour of the chromium coating was described for temperatures below the point of formation of the Cr/Zr eutectic melt. Further research is needed to take into account processes above this eutectic point.

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**2 / 41**

# ZIRCONIUM ALLOY CLADDING WITH CHROMIUM COATING: DIFFUSION-REACTION MECHANISMS AND STRUCTURAL DEGRADATION UNDER EXTREME TEMPERATURES

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This study establishes a Zr-Cr diffusion mechanism model to quantitatively reveal Cr concentration profiles, ZrCr2 layer growth kinetics, and defect formation mechanisms based on an experimental investigation, providing theoretical support for failure prediction and emergency strategies in extreme conditions.

**2 / 42**

# TESTING-SIMULATION OF PELLET-CLADDING MECHANICAL INTERACTION WITH ATF CLADDINGS

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INTRODUCTION: Chromium-based coated Accident Tolerant Fuel (ATF) cladding, currently the only feasible surface treatment for pressurized water reactors, mitigates the limitations of conventional Zirconium (Zr) claddings by enhancing performance, enabling higher burnup, and reducing exothermic metal-water reactions and cladding embrittlement from condition I to Beyond Design Basis Condition (BDBC), with the incorporation of a nitride layer specifically aimed to limit chromium diffusion, enhancing the overall stability of the coating. As coating performance depends on the integrity of the coating, its behaviour must be assessed through the whole plant operation spectrum. In this regard, to date, most studies have focused on BDBC, leaving the assessment of conditions I and II largely theoretical, potentially overlooking pre-accident failures. Pellet Cladding Mechanical Interaction (PCMI), driven by pellet swelling, is an unavoidable phenomenon in conditions I and II that generates radial contact pressure and cladding expansion, potentially leading to coating cracking. To evaluate coating response under such conditions, the mandrel ductility test was employed, as it simulates pellet expansion through radial displacement of a segmented tool. The SNETP-OFFERR project on the Simulation of PCMI with ATF claddings was a joint effort involving a user team, composed of several European nuclear research institutions, coordinated by the Università di Pisa, and the infrastructure team, the HUN-REN Centre for Energy Research. The technical focus of this collaboration was the development and application of redesigned segmented mandrel ductility tests, followed by post-test analysis of a range of unirradiated, coated and uncoated claddings. The primary objective of the project was to investigate the mechanical behaviour, deformation capability, and failure modes of the coating in samples subjected to radial displacement under postulated PCMI conditions, while in parallel, a novel Finite Element simulation was built and followed by an innovative code coupling to assess the mechanical sub-models of fuel performance codes to simulate coating behaviour. The mandrel test was followed by post-test analyses, including Scanning Electron Microscopy (SEM), Light Optical Microscopy (LOM), metallography, and high-temperature oxidation of coated samples. SEM and LOM were conducted after the mandrel test to investigate the potential presence of cracks, as their formation was uncertain. Subsequently, high-temperature oxidation tests were performed to assess the impact of possible pre-existing cracks on coating behaviour with oxidation parameters of 1000 ℃ for 1800 seconds in steam atmosphere, simulating extended LOss of Coolant Accident (LOCA) condition. Metallography aimed to evaluate microstructural changes at the cladding–coating interface, identify delamination or crack propagation, and assess diffusion behaviour, particularly in nitride-containing coatings, following mechanical strain and oxidation.
1. MANDREL DUCTILITY TESTING
To evaluate coating behaviour under postulated PCMI conditions, 25 unirradiated cladding samples were tested, including Cr, CrN, and Cr/CrN multilayer coatings applied with via various physical vapour deposition techniques on Zr1%Nb and Optimized ZIRLO (Opt-ZIRLO) alloys [1]. Uncoated Zr1%Nb and E110 served as references, with the Zr1%Nb and Opt-ZIRLO samples supplied by AEOI and CTU respectively in the IAEA CRP ATF-TS. The mandrel test setup involves a cladding sample that is radially expanded by the mandrel segments due to the vertical insertion of a central spike. This test simulates pellet expansion, a key contributor to fuel behaviour during power ramps and extended burnup. The current mandrel test has undergone significant modifications compared to earlier mandrel tests [2], more closely resembling PCMI conditions through the addition of a fillet and a modified contact surface. The test parameters included an average total radial strain of

5%, with quasi-static crosshead displacements of 0.0175/0.175 mm/min and temperatures of approximately 20 ℃ and 300 ℃. Strain was conservatively exceeded the design limits of stress/strain/fatigue, while temperature conditions reflected realistic conditions. Additionally, a 3D animation was created to improve understanding of the testing process and aid code developers with clearer visuals (Fig. 1).

FIG. 1. 3D Animation of the Mandrel Ductility Test setup [1]
2. EXPERIMENTAL
The test included monotonic loading, reaching target deformation in 11 minutes after mandrel contact, and quasi-cyclic loading applied to a Cr-coated Opt-ZIRLO sample with 11 incremental steps to evaluate relaxation and fatigue, followed by unloading. Strict control of frictional conditions was essential, as even minor variations in friction coefficients significantly influenced load measurements and local cladding deformation, despite consistent strain outcomes, as confirmed through controlled blind tests. Preliminary results showed all samples reached 5% total average deformation without cracking of the Zr-alloy, confirming Zr cladding substrate ductility and load-bearing. Both coated and uncoated Opt-ZIRLO required higher force than Zr1%Nb at both temperatures, indicating greater strength of Opt-ZIRLO, though this difference was lower at higher temperature. The coatings were affected only by localized phenomena like surface cracking, with minimal impact on the mechanical behaviour of the claddings. Comparison between cyclic and monotonic loading highlights the effect of fuel conditioning on cladding behaviour once hard contact is established (Fig. 2).

FIG. 2. Force–average displacement and force–time curves of Opt-ZIRLO samples tested at 300 ℃ [1]
3. POST-TEST SEM
The first phase of post-test characterization involved SEM to visualize the crack patterns and surface features induced during deformation (Fig. 3). The examination revealed consistent axial cracking across all coated samples, which was unrelated to the fabrication lines; however, the morphology and density of the cracks varied significantly with the Zr-alloy, temperature, and coating composition. For pure Cr-coated Opt-ZIRLO, at most strained locations, sparse ≈1 μm wide axial cracks were observed, while Zr1%Nb showed wider cracks (3–5 μm) spaced ≈150 μm apart. Crack width also increased with temperature mainly in Zr1%Nb, remaining nearly unchanged in Opt-ZIRLO, indicating greater resilience of Opt-ZIRLO and higher plasticity in Zr1%Nb, with both localized with sharp tips, indicative of the brittle fracture behaviour of the Cr layer in both alloys and temperatures. While in CrN and Cr/CrN multilayer coatings on Zr1%Nb, a different cracking pattern, with dense often V-shaped crack network (2-4 crack), with 1-5 μm width occurring every ≈100 μm was seen. These cracks followed distinctive lightning bolt-like paths, frequently branching and intersecting tangentially, with localized chipping at intersections, indicative of a pronounced brittle failure mechanism. Nevertheless, even with significant surface degradation, generalized delamination of the coatings was not observed, and the layers remained strongly adhered to the substrate.

FIG. 3. Crack Width and Morphology of coating mandrel tested at 300 ℃: a) Cr-coated Opt-ZIRLO b) Cr-coated Zr1%Nb c) CrN-coated Zr1%Nb d) Cr/CrN-coated Zr1%Nb [1]
4. POST-TEST OXIDATION AND METALLOGRAPHY
To assess the effect of pre-existing cracks and reveal microcracking, six coated samples, Cr-coated Opt-ZIRLO and Cr/CrN-coated Zr1%Nb, were exposed to oxidation in steam at 1000 ℃ for 1800 seconds with Ar carrier gas, simulating extended LOCA after postulated-PCMI. Accurate assessment of coating oxidation behaviour requires prior consideration of mechanical behaviour. The specific mass gain normalized to surface data from oxidation tests revealed that Opt-ZIRLO-based samples showed higher mass gain than Zr1%Nb samples.

FIG. 4. Metallography after oxidation of a) Cr-coated Zr1%Nb and b) Cr-coated Opt-ZIRLO mandrel tested at 300 ℃, and c) Cr-coated Zr1%Nb and d) Cr-coated Opt-ZIRLO mandrel tested at 20 ℃ [1]
The preliminary studies by LOM analysis reveals formation of green $Cr_2O_3$ layer on the coating surface and at cracked and exposed areas, which is not beneficial. Due to stability of Zirconia, the steam will preferentially oxidize the underlying Zr-alloy, while also leading to the formation of an oxygen-stabilized region, promoting embrittlement. Over time, the diffusion of Zr through Cr grain boundaries toward the outer surface causes reduction of $Cr_2O_3$, accelerating Zr oxidation, and potentially forming voids in the Cr layer. This effect will be accelerated when the volumetric expansion during oxidation induces tensile bending moment on the brittle interface, promoting further crack propagation beneath the coating with subsequent increase in oxidation. These results showed the behaviour of pre-cracked coating layers (Fig. 4) –which is largely unexplored in the literature compared to the extensive studies on intact coatings –showing that once the coating is cracked, its protective

effectiveness may be significantly compromised under accidental conditions. To further investigate the oxidation effect on coating composition and morphology, samples were embedded in epoxy and prepared for metallographic analysis. LOM showed that coating cracks expose the Zr-alloy to localized oxidation, promoting crack growth in the oxygen-stabilized region, unlike uniform oxidation in conventional alloys, this may accelerate localized cracking and lower fracture toughness. The oxidation of the inner side reduces the thickness load-bearing prior beta phase. LOM analysis of CrN and Cr/CrN multilayer samples tested at different temperatures revealed that nitrogen interdiffusion in the coating and on the coating-substrate interface, leading to the formation of a brittle golden ZrN phase in the coating, at crack sites and interface regions (Fig. 5). Although ZrN may initially enhances corrosion resistance by reducing the intermixing of Cr and Zr, it then transform into porous $ZrO_2$ at high-temperature in steam atmosphere, accelerating oxidation. This effect was more evident in CrN-based coatings, highlighting a trade-off between short-term protection and long-term stability in nitrogen-containing coatings

FIG. 5. Metallography after oxidation of CrN-coated Zr1%Nb mandrel tested a) at 20 ℃ and b) at 300 ℃, and Cr/CrN-coated Zr1%Nb mandrel tested c) at 20 ℃ and d) at 300 ℃ [1]
5. SUMMARY
The OFFERR ATF mandrel ductility testing project successfully assessed the mechanical behaviour of several ATF cladding candidates under representative postulated-PCMI conditions and established a novel database on coating crack formation. Cr-based coatings showed strong adhesion and good oxidation resistance, but remained prone to brittle cracking, which may hinder the cladding's ability to fully meet ATF performance requirements. Post-test evaluations revealed the coating's complex mechanical behaviour under varying conditions, underscoring the need for further experimental studies on its tensile mechanical properties. Selecting the best-performing coating composition and technology based on oxidation resistance and cracking is challenging, as each cladding type is designed for different reactor environments and conditions. However, the inherent brittleness of Cr and the interdiffusion of both Cr and nitrogen remain critical factors to consider in performance evaluation. Moreover, although all coating layers exhibited cracking, no spalling or large-scale delamination occurred. The coatings remained adherent even under extended LOCA oxidation conditions, indicating preserved mechanical integrity despite localized damage. However, the protective effectiveness of cracked coatings remains questionable and may diminish over time. These results highlight the importance of optimized design in terms of thickness, application techniques, and substrate compatibility. Improved microstructural control and impurity management, along with advanced predictive modelling through finite element methods and code coupling with fuel performance models, are essential next steps. Future efforts should incorporate irradiation effects such as neutron-induced hardening, creep, and fission gas behaviour to fully assess ATF cladding performance under in-reactor conditions, as current findings are based solely on unirradiated intact coatings.
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**2 / 43**

## SEPARATE EFFECTS TESTS CONDUCTED WITH ATF FUEL CLADDING MATERIALS

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INTRODUCTION: In the framework of the IAEA Coordinated Research Project Testing and Simulation for Advanced Technology and Accident Tolerant Fuels (ATF-TS), a comprehensive experimental program was conducted to investigate the performance of advanced cladding materials. The goal was to characterize their behaviour under various conditions and to compare the newly obtained data with that of traditional cladding alloys used in water-cooled reactors.
1. OVERVIEW
Comparative tests were carried out using cladding tubes supplied by multiple ATF-TS project partners. The tested cladding types included traditional reference materials—such as Zircaloy-4, ZIRLO™, Optimized ZIRLO™, and Zr1%Nb alloys—as well as advanced materials featuring coatings or alternative compositions. These included claddings with external surface coatings of chromium (Cr), chromium nitride (CrN), multilayer CrN/Cr, titanium-aluminium (TiAl), and a high-performance FeCrAl (B136Y3) alloy.
The HUN-REN Centre for Energy Research was responsible for performing a substantial number of separate effect tests and one integral test using an electrically heated bundle (CODEX-ATF). Data from the CODEX test were stored in a separate database. This report summarizes the main characteristics and results from the following separate effect tests: Ballooning and burst tests, Oxidation in high-temperature steam, Tensile tests using ring samples, Ring compression tests, Mandrel tests and Scanning electron microscopy (SEM) examinations.
2. MATERIALS AND COMPLETED TESTS
Thirteen different cladding types were used in the HUN-REN EK test program, provided by several ATF-TS partners: Karlsruhe Institute of Technology (KIT), Atomic Energy Organization of Iran (AEOI), Canadian Nuclear Laboratories (CNL), Czech Technical University in Prague (CTU), Belarusian State University (BSU). Although parallel tests were planned for all materials, the available tube lengths varied between suppliers. Consequently, the number of tests performed for certain cladding types was limited. Detailed test matrices, the separate reports and a full list of completed measurements are provided in Appendix I of the TECDOC.
3. DATABASE DIRECTORY STRUCTURE
The experimental database created for the ATF-TS program builds on the structure of the earlier E110 cladding database developed by the predecessor of HUN-REN EK. The previous database is publicly available in the OECD NEA databank under the code NEA-1799 IFPE/AEKI-EDB-E110. The new database for the ATF-TS tests mirrors this directory structure to facilitate future integration [1]. Each test series is documented in a PDF report, and raw data are provided in both Excel spreadsheets and ASCII files. Each individual measurement has a dedicated spreadsheet and corresponding ASCII file, ensuring consistency and traceability across formats.
4. BURST TESTS
Several types of potential ATF candidate cladding materials were subjected to ballooning and burst tests at elevated temperatures using various internal pressurization rates. In total, 119 burst tests were conducted. While the majority of tests took place in an inert atmosphere, a subset was carried out in a steam-rich oxidizing environment following pre-oxidation. The primary outcomes included the measured burst pressures, non-destructive evaluations of the deformed geometry, and visual inspections of the coatings to assess damage incurred during ballooning and rupture.
The results indicated that, under isothermal burst test conditions, there was no statistically significant difference in burst pressure between coated and uncoated reference claddings. Similarly, the analysis of maximum strain at the burst site revealed no substantial variation between the two groups; however, this was largely attributed to high data scatter, which resulted in broad confidence intervals. Notably, all tested coatings remained adherent to the underlying cladding with no signs of spalling. Nevertheless, the observed cracking patterns varied significantly depending on the coating type.
5. OXIDATION EXPERIMENTS
The oxidation resistance of various cladding materials was evaluated using samples with different surface treatments, including Cr, CrN, CrN/Cr, and TiAl coatings, as well as uncoated zirconium alloys and the FeCrAl alloy. Oxidation was carried out in a high-temperature tube furnace equipped with a quartz tube. The setup consisted of a steam generator, a three-zone resistance furnace with precise temperature control, and a condenser. The cladding tubes were cut into 8 mm long segments, suitable for subsequent ring compression testing. Each segment was oxidized on both sides while placed in a quartz boat. The tests were conducted under isothermal conditions at two temperatures 1000 ℃ for 1 hour and 1200 ℃ for 30 minutes. In total, 92 oxidation tests were conducted. The mass gain and Equivalent Cladding Reacted (ECR) were calculated to quantify the extent of oxidation. For coated samples, it was necessary to determine the specific mass gain on the coated surface

separately, as total surface-based calculations did not yield accurate results.
The majority of coated samples exhibited better oxidation resistance than uncoated ones, with two notable exceptions: the titanium-aluminium coated and plasma-treated chromium-coated zirconium alloys. These coatings were not effective in preventing oxidation beneath the surface. In contrast, the Cr-coated Optimized ZIRLO™ and the FeCrAl alloy demonstrated excellent resistance to steam oxidation at both test temperatures.
6. RING TENSILE TESTS
Tensile strength was assessed using ring segments from both coated and uncoated zirconium alloy claddings in as-received and oxidized conditions. The tubes were supplied by KIT, AEOI, CNL, and CTU. In total, 60 ring samples were tested. Each cladding tube was sectioned into 2 mm rings. For each material, six rings were tested: two in as-received condition, two after oxidation at 1000 ℃ for 3600 seconds, and two after oxidation at 1200 ℃ for 1800 seconds. Geometrical measurements were taken at three locations per ring. Tensile tests were conducted using an INSTRON 1195 machine at room temperature with a crosshead speed of 0.5 mm/min. Load-displacement data were recorded every 0.4 seconds until the samples fractured.
Coatings had little influence on the tensile strength of as-received zirconium alloy samples. However, the oxidation treatment led to significant changes. Zirconium-based samples became brittle after oxidation, due to the two-sided exposure during steam treatment, which led to heavy internal oxidation. For samples oxidized at 1000 ℃, ultimate tensile strength (UTS) ranged from 3.5 to 188 MPa. After oxidation at 1200 ℃, UTS values dropped to between 0 and 24 MPa. In contrast, FeCrAl samples retained much of their mechanical strength even after oxidation. Their UTS values were over 600 MPa in as-received state, ~500 MPa after oxidation at 1000 ℃, ~440 MPa after oxidation at 1200 ℃. These results suggest that thermal exposure played a more significant role than oxidation in altering the mechanical properties of FeCrAl. Notably, all oxidized FeCrAl samples remained ductile.
7. RING COMPRESSION TESTS
Ring compression tests were performed on both coated and uncoated zirconium alloy claddings, as well as on FeCrAl samples, following steam oxidation. A total of 62 samples were tested. The cladding tubes were sectioned into 8 mm long rings and oxidized in high-temperature steam. For each cladding type, three samples were tested after oxidation at 1000 ℃ for 3600 seconds, and three others after oxidation at 1200 ℃ for 1800 seconds. Compression tests were conducted using an INSTRON 1195 universal test machine at room temperature (~20 ℃), with a crosshead speed of 0.5 mm/min.
The results showed that most uncoated zirconium alloys became brittle after oxidation at 1000 ℃. An exception was the Zircaloy-4 provided by KIT, which exhibited a ductile plateau in its load-displacement curve. All other uncoated alloys fractured quickly following elastic deformation. Differences in maximum load values were strongly influenced by cladding wall thickness. For instance, Zircaloy-4 samples from CNL had thinner walls, which explained their lower maximum loads.
Among the tested materials, Cr-coated and uncoated Optimized ZIRLO™ samples demonstrated distinct mechanical behaviours. The uncoated sample exhibited brittle failure after reaching a maximum load of 321 N, whereas the Cr-coated sample maintained ductile deformation and reached a maximum load of 435 N. Thus, the coating imparted a measure of ductility under the given oxidation conditions.
FeCrAl samples remained fully ductile, even after oxidation at 1200 ℃ for 1800 seconds. However, under these extreme oxidation conditions, Cr coatings failed to prevent embrittlement in either Optimized ZIRLO™ or Zr1%Nb claddings. The titanium-aluminum (TiAl) coating also did not improve high-temperature oxidation resistance or mechanical performance.
8. MANDREL TESTS
Drawing on available literature, a mandrel test setup was designed and constructed to study the mechanical interaction between fuel pellets and cladding under PCMI-like radial loading conditions. Mandrel ductility tests were carried out to evaluate the ductility of both coated and uncoated ring specimens sectioned from fuel cladding tubes. All tests were conducted at room temperature. Similar to previous mechanical tests, the coatings on zirconium alloy claddings did not significantly alter the deformation behaviour of as-received samples. The results showed that the presence of coatings had a negligible impact on the overall strength of the cladding materials. For the FeCrAl cladding, the measured maximum force was lower than that of the zirconium-based samples, primarily due to its reduced wall thickness. Additionally, the maximum diameter increase observed in FeCrAl was significantly smaller compared to the other reference materials. All tested materials met the target deformation without failure, though the coatings developed surface cracks, which were further analysed in the SEM investigations.
9. SEM EXAMINATIONS
Morphological analysis of the cladding samples was conducted using two scanning electron microscopes. Secondary electron (SEI) and backscattered electron (BEI) images were taken at 5 keV. Cross-sectional and fracture surfaces were also examined. A total of 23 cladding samples in both as-received and oxidized states were analysed. Investigations focused on the outer surfaces, cross-sections, and fracture surfaces after mechanical testing. The SEM analyses corroborated mechanical

and oxidation test results, illustrating the limits of coating effectiveness at elevated temperatures and identifying failure modes associated with each cladding/coating combination.
In selected cases, focused ion beam (FIB) analysis was used to ablate micro-trenches (~a few micrometres deep) into the surface to reveal subsurface structures. Samples were tilted at 52° during FIB preparation, and measured layer thicknesses were corrected by a factor of 1.27 to account for this tilt. Elemental analysis was performed using an Oxford X-MAX 20 energy-dispersive X-ray (EDX) spectrometer with a silicon drift detector. Measurements were conducted on selected areas of interest.
Cr-coated Optimized ZIRLO™ samples: The outer surfaces were dominated by plate-like Cr-oxide crystallites. At lower oxidation temperatures, these plates were numerous and randomly oriented, with sizes ranging from a few tenths of a micrometre to about 1 µm. At higher temperatures, the number of crystallites decreased while their size increased. Smoother surface regions contained detectable amounts of Cr and O alongside Zr.
Zr1%Nb samples coated with Cr, CrN, or Cr+CrN: These displayed heterogeneous surface structures with a combination of plate-like crystallites, smooth patches, cracks, cavities, and particle agglomerates. At 1200 ℃, the crystallites became more prominent in both size and contrast. Cr-oxides were confirmed as the primary phase. Some smoother regions and grains also revealed the presence of Zr and impurities.
TiAl-coated Zircaloy-4 samples: The TiAl layer failed to prevent oxidation of the underlying zirconium. Oxidation led to spalling of the coating in several locations. The spalled layers contained higher concentrations of Ti compared to the remaining cladding material, indicating separation of the Ti-rich coating during the exposure to steam.
10. SUMMARY
The separate effect tests conducted at the HUN-REN Centre for Energy Research under the IAEA ATF-TS program provided a comprehensive dataset on the behaviour of advanced cladding materials under thermal and mechanical stresses. Key conclusions from the test series include:
—Cr coatings significantly improved oxidation resistance of zirconium alloys at high temperatures, with Optimized ZIRLO™ and FeCrAl performing best. However, certain coatings such as TiAl and plasma-treated Cr were ineffective under the tested conditions.
—FeCrAl retained ductility and strength after oxidation, whereas Zr-based claddings became brittle, particularly after two-sided oxidation. Cr coatings improved ductility in ring compression tests, but could not prevent embrittlement at 1200 ℃.
—Surface cracks and morphological degradation were observed in coated claddings after testing. Crystallite growth and surface inhomogeneities were strongly temperature-dependent. SEM imaging confirmed that TiAl coating is unstable and prone to delamination during oxidation.
All test data are available in a structured digital format in the IAEA database. This supports future access for fuel modelling and safety analysis purposes. The findings contribute to the broader international effort to qualify accident-tolerant fuels for use in current and next-generation water-cooled nuclear reactors. Further studies, will be required to fully validate these materials for in-core application.
ACKNOWLEDGEMENTS
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**4 / 44**

## Risk-Informed Licensing Approaches to Advanced Fuel Technologies: A Regulatory Perspective

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In the United States, recent legislative actions and executive orders have spurred significant interest in utilizing advanced fuel technologies and analytical approaches to support power uprates and extended cycles at current operating boiling and pressurized water reactors. To facilitate efficient and timely licensing reviews while continuing to protect public health and safety, the U. S. Nuclear Regulatory Commission (NRC) has been exploring a variety of risk-informed and performance-based licensing pathways, some initiated by the NRC and some proposed by licensees/applicants. This presentation will focus on commercial light water power reactor applications, though the underlying licensing concepts may be more generally applicable. Near term challenges to be resolved include licensing of novel small modular reactor designs and enabling higher burnups and enrichments for current operating reactors to allow for longer cycles and higher power levels. Establishing clear and efficient licensing pathways to address these challenges without compromising safety will require significant evolution in regulatory frameworks, licensing processes, and technical bases by the NRC. While the loss of coolant accident is the primary focus of this presentation due to its unique role as the primary design basis accident that nuclear power plants must be designed and maintained to mitigate, some of the changes will also ultimately inform similar changes elsewhere in the NRC's licensing of nuclear power plants.

**3 / 45**

## FINITE ELEMENT SIMULATION OF MANDREL TEST WITH ATF CLADDING

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FINITE ELEMENT SIMULATION OF MANDREL TEST WITH ATF CLADDING
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INTRODUCTION: Accident Tolerant Fuel (ATF) claddings are designed to overcome the limitations of conventional Zirconium (Zr) alloys from Normal Operation Conditions (NOC) to Beyond Design Basis Conditions (BDBC) by reducing the metal–water reaction and hydrogen generation. As a short-term solution, coating stability at T>1200 ℃ in high steam pressure (BDBC), compatibility with the coolant, and minimal impact on neutron economy (NOC) is required, making thin (<50 μm) Chromium (Cr) coatings the only option for Pressurized Water Reactors (PWRs). However, to meet its intended function at each mode of operation, the coating's structural integrity must be verified, especially for NOC, as any failure during this phase would leave the core unprotected in accident scenarios. Physical Vapour Deposition (PVD) is a promising technique, as it offers uniform thickness, reduced residual stress, and eliminates defects such as wrinkles commonly seen in cold spray. The dominance of deposition kinetics over thermodynamics tailors a temperature-dependent, textured microstructure with finer grains than the bulk, while impurities primarily influence the fracture behaviour. To date, R&D has mainly focused on the mechanical integrity of unirradiated Cr-based coatings under accident conditions, while their behaviour under NOC, despite being the most frequent mode, remains largely theoretical, as coating integrity during NOC directly affects cladding performance under BDBC. As the Cr-layer is thin and it introduces minimal additional effects, conventional PWR performance metrics and phenomenon remain applicable. Under NOC,

Pellet Cladding Mechanical Interaction (PCMI) is an unavoidable phenomenon that can impair the protectiveness of Cr-based coatings due to the radial expansion of the cladding caused by the pellet expansion. Additionally, the urge for higher burnup levels, the adoption of load-following, along with pellet deformation (chipping-hourglassing), intensifies the PCMI. To investigate coating behavior under the postulated PCMI, the modified mandrel ductility test was conducted within the SNETP-OFFERR "SPCMI with ATF Cladding"project, jointly led by the Università di Pisa (user team) and the HUN-REN Centre for Energy Research (infrastructure team). The project was in two interconnected phases: mandrel ductility tests, including post-test analysis such as SEM, profilometry, etc, on over 25 unirradiated as fabricated cladding samples, including Optimized ZIRLO (Opt-ZIRLO) and Zr1%Nb coated with Cr, CrN, or multilayer Cr/CrN using various PVD-sputtering methods, provided by CTU and AEOI from the IAEA CRP ATF-TS, along with uncoated E110 reference samples. Simultaneously, numerical studies using a validated 3D Finite Element (FE) model of mandrel test with coated Zr1%Nb sample were performed, integrated with an innovative code-coupling procedure. Furthermore, fractography analysis of fractured surfaces, in conjunction with ATF-TS separate-effect tests, contributed to the identification of failure modes.
1. MANDREL DUCTILITY TEST
The current mandrel test is displacement controlled, it employs a rigid hexagonal pyramidal spike mounted on a tensile machine, engaging with six rigid tapered (γ=2°) segments (separated by small clearances), with filleted edges, in contact with tube over an active height close to a single fuel pellet (Fig. 1). The sample sits on the ledge of the segments, having an initial gap between the segment and the tube (≈250μm). The presence of two fillets on the active length of the segment generates localized strain zones, which makes the configuration analogous the PCMI. To clarify the coating behaviour, first the loading procedure, consisting of loading (monotonic or cyclic) and unloading shall be studied. At the initial stage (Contact Initialization), spike insertion at a speed of (uz) establishes contact on the tapered side of the segment, creating driving pressure, inducing friction-controlled sliding over the greased base plate, while the machine records minimal forces. Immediately after the gap closure, as contact has already been established, the elastic radial expansion of the tube (Deformation Response) occurs as the measured force rises rapidly, then the forces plateau above the yield strength followed by plastic radial expansion. The assumption of a rigid mandrel significantly simplifies the analysis, allowing the radial expansion of the tube at the segment–cladding contact to be ur=uz×tg(γ). Though idealized, it illustrates, at the contact regions, the cladding follows the segment, resulting in a 3D strain state, whereas regions with clearance experience a near plane-strain condition. At higher deformation the wall thinning and subsequent necking occurs, while cracking initiates in the high-strain contact zones. In the mandrel test, friction serves as both a critical mechanical factor and a significant source of uncertainty, as it affects both force transmission and local cladding deformation, potentially causing uneven segment sliding or even leading to segments binding and moving in pairs. Two strategies are implemented, first to minimize friction, as lower values are preferred for the material characterization test –including mitigation through standardized lubrication applied before each test –and to control friction variability with blind tests conducted on the reference material. The testing parameters set in the project consisted of two crosshead displacement rates, 0.0833 and 0.833 mm/s, quasi-static loading up to an average of 5% strain (conservative) to ensure that cladding kept its density and circularity, and they were conducted at room temperature and 300 ℃ (enclosed within a two-part furnace), with the latter being closer to realistic conditions, while the former was conservative.

`FIG. 1. Mandrel Test Setup` `FIG. 2. Force–Radial Displacement Curve`

1. PROFILOMETRY Profilometry is commonly used in post-test examinations to assess dimensional changes of the sample. In this study, Keyence VR-5200 machine (non-contact 3D surface profilometer) was used to measure surface topography. After setting a robust procedure to extract data, combined with mandrel test results, statistical analysis of axial and radial deformation revealed the influence of temperature, materials, and test uncertainties. As expected, the coated Zr-1%Nb samples exhibited lower force and higher plastic deformation and radial displacements compared to their Opt-ZIRLO counterparts at both temperatures. Moreover, results revealed an increase in the deformed area with respect to the active region of the segment, showing the creation of a tangent location. Additionally, profilometry identified additional, bending-induced radial deformation, which was previously unaccounted for as a potential source of experimental uncertainty. These results were then used to validate the 3D FE model, specifically developed to model the current mandrel test.

2. FINITE ELEMENT MODEL DESCRIPTION Accurate fuel performance predictions rely on vali-

dated codes, appropriately scaled mechanical models, and realistic material properties. Given the complexity of the test and its inherent uncertainties, including complex geometry, large strains, material and geometric nonlinearity, and uncertain frictional effects, 3D FE analysis was chosen to simulate cladding behaviour. The 3D nonlinear analysis was performed using HEXAGON Marc/Mentat 2022.2, a FE code suited for advanced structural and multi-physics problems. Modelling began by generating deformable meshed bodies and rigid geometries, starting from a 2D CAD file of the actual test geometry, converted into a detailed 3D refined meshed 8-noded elements to meet both contact and meshing criteria. The selection of the glued mesh contact for addition of the Cr layer to the model was supported by experimental observations, as fractography of the coating interface revealed strong adhesion, as it was meshed with the exact axial and circumferential density as the cladding to ensure a homogenous response under loading (Fig. 4). The model also features realistic contact interactions (Fig. 5), boundary conditions, and loading cases, providing good alignment to the actual test. As with any simulation, validation and consistency checks were performed with reference samples to ensure the adequacy and accuracy of the procedure. A sensitivity study was also conducted on meshing, using axial mesh ratios of 1:1 and 2:1 between the mandrel. The elastic bending of previously omitted components was added to the model to better estimate the initial stages of the test (Fig. 5).

FIG. 4. Contact Description of FE Model FIG. 5. Material Description of the Modified FE Model

3.1 MECHANICAL ANALYSIS BASED ON FE MODEL OF MANDREL TEST

Once the model was validated, a mechanical analysis of the coated cladding was performed based on the material properties implemented. The first step was to verify the rigidity of the segments and the spike, as the assumption of a 5% average strain relies on their rigidity. After that a stress-strain analysis was conducted through the wall thickness. At the first phase of the test, the maximum stress occurs at the inner wall of the cladding due to initial hard contact, where the tangential stress component is highest, followed by a complex increase in stress through the thickness over time. While the Cr-layer shows an offset in stress relative to the Zr-alloy at the Cr-Zr interface, which propagates with contact, it suggests that coating failure likely occurred before Zr yielding, as both models displayed a consistent trend (Fig. 6), despite differences in specific details. Therefore, in contact problems like PCMI, plasticity along the thickness varies with contact propagation, with Cr layer affecting the mechanical response locally. The strain analysis on the other hand, confirmed that plasticity is also likely to occur at the inner side, yet more in between the segments, where the tangential strain is maximized (Fig. 7). A comprehensive fractographic analysis of the samples from the current project, coupled with mandrel tests performed during the ATF-TS separate effect test [2] revealed an intergranular-brittle failure mode instead of transgranular-brittle, indicating earlier failure than predicted by bulk material properties. This may hint at the importance of impurities, such as oxygen and carbon, which can impede the movement of dislocations, and it may vary for each coating technology and Zr-alloy.

FIG. 6. Stress Profile Through Thickness FIG. 7. Strain Profile at the Mid-Surface of Deformed Zone

4.CODE COUPLING

With ATF cladding advancing to TRL 7 and green taxonomy favouring advanced fuel technologies, there is a growing need for Fuel Performance Codes (FPCs) to simulate the ATF parent rod under NOC. However, since the FPCs are validated for uncoated claddings, their mechanical sub-models must be reassessed to ensure an accurate prediction of coating behaviour, particularly due to their limits in capturing complex contact-related phenomena, such as PCMI. On the other hand, 3D codes can accurately capture complex contact mechanisms. Therefore, to leverage the advantages of both approaches, an innovative code coupling strategy was developed. The coupling approach was based on the premise that cladding deformation observed in the mandrel test can be replicated using an equivalent gas, as the contact pressure during NOC-PCMI can be approximated by an increase in internal rod pressure. To support this approach, azimuthally averaged contact pressures were derived from the validated mandrel FE model (ECRI-51-B2) at specified locations (Fig.8), including the mid-height region under quasi-plane strain, and fillet zones with pronounced strain concentrations. During the coupling process, challenges such as interface inconsistencies, feedback instabilities, and uncertainties in friction modelling were addressed by the strategy implemented, maintaining clear boundaries between the 3D structural code and 1.5-D FPC TRANSURANUS, reducing uncertainties, and preserving model integrity. To verify the governing equations established in analogy between the mandrel test and the equivalent gas model, a separate FE model was realized. The results showed that, at maximum internal pressure, corresponding to peak loading condition, the coated cladding deformation aligned with the target average of 5%, confirming the accuracy of the model (Fig. 9).

FIG. 8. Pressure History Obtained from Validated Mandrel FE Model
FIG. 9. Total Equivalent Plastic Deformation of Coated Sample at the Peak Pressure

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**2 / 46**

## BEHAVIOUR OF BUNDLES WITH CR AND CR/CRN COATED ZR ALLOY CLADDING UNDER SIMULATING DESIGN EXTENTION CONDITIONS AT THE DEGREE FACILITY

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Integral tests with bundles of Zircaloy_4 fuel cladding coated with 10 μm Cr, 20 μm Cr, and 26 μm Cr/CrN were carried out under simulating design extension conditions (DEC) using an induction heating furnace at the CRIEPI's DEGREE facility. The bundles were heated to 1350 ℃ or 1500 ℃ in a steam-Ar mixed gas atmosphere and then quenched in Ar gas. In addition to on-line measurements of cladding surface temperatures and fuel rod internal pressure (RIP), post-test analyses were performed to understand the ballooning/burst behaviour and the degradation behaviour of the coating layer itself and those at the coating/substrate interface.

**2 / 47**

## AN OVERVIEW OF CEA LOCA SEPARATE EFFECTS TESTING APPLIED TO CHROMIUM-COATED ZIRCONIUM ALLOY CLADDINGS

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This presentation will focus on the behaviour of Cr-coated zirconium alloy claddings under LOCA conditions. Among the various coating concepts investigated by CEA over the past 15 years, particular attention has been paid to chromium coatings deposited via Physical Vapour Deposition –High

Power Impulse Magnetron Sputtering (PVD-HiPIMS), identified early on as a promising candidate for enhanced accident tolerance. The presentation will provide an overview of key findings based on a series of published studies by CEA, covering experimental methods and dedicated test facilities used to assess the performance of these coatings under both isothermal and ramped thermal loading, including ballooning, burst, and high-temperature steam oxidation.

**1 / 48**

## WESTINGHOUSE ENCORE® ACCIDENT TOLERANT FUEL AND HIGH ENERGY FUEL STATUS AND COMMERCIALIZATION

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INTRODUCTION: Following the Fukushima Daiichi powerplant accident in March 2011 which resulted from an offshore tsunami, the U.S. Department of Energy (DOE) launched a program to investigate enhancing the accident tolerance of light water reactors (LWRs). The program mandated a focus on the development of accident tolerant fuel (ATF) that when compared to traditional UO2 and zirconium-alloy fuel systems could tolerate a loss of active cooling in the reactor core for a considerably longer time period while maintaining or improving the fuel performance during normal operation, operational transients, or accidents [1]. Aside from the safety enhancements offered by ATF concepts, there is an economic driver for implementation as the new fuel concepts could increase fuel flexibility through power uprates and reaching higher burnups [2].
To support improved operating and safety margins while also enabling longer cycles, Westinghouse is commercializing a near-term high energy EnCore ATF product comprising chromium (Cr) coated zirconium-alloy fuel cladding with doped UO2 (Advanced DOped Pellet Technology –ADOPT™) fuel pellets with greater than 5% 235U by the late 2020s. Post-irradiation examination (PIE) results of commercially irradiated Cr-coated lead test rods (LTR) have demonstrated enhanced performance with minimal oxidation and crud adherence to high burnups near 75 GWd/kgU. A Spring 2025 lead test assembly insertion with these new products included ADOPT pellets with greater than 5% 235U fuel pellets. Licensing topical reports for high energy fuel (HEF) technologies and associated codes and methods are underway and to support future customer deliveries, manufacturing implementation of increased enrichment ADOPT fuel pellets and Cr coated fuel cladding are in-progress at multiple Westinghouse fabrication facilities.
1. ATF AND HIGH ENERGY FUEL TECHNOLOGIES AND BENEFITS
Increased electricity demand, the potential for zero-carbon new capacity installation and emission credits and other carbon-reduction initiatives are incentivizing nuclear power utilities to focus on increasing generation capability. Thus, utilities are striving for maximum electricity generation and reduced operating costs while enhancing safety. The Westinghouse HEF Program, which includes EnCore ATF technologies, combines near-term and future advancements of fuel cladding, pellets, and structural features to achieve these goals along with higher enriched (>5% 235U) fuel pellets [3], FIG. 1.

FIG. 1. The Westinghouse High Energy Fuel and Accident Tolerant Fuel Program.
Existing fuel advancements include AXIOM® zirconium-alloy cladding and PRIME™ structural features. AXIOM is a fully licensed, proven cladding which offers improved corrosion resistance, lower hydrogen pickup, and enhanced dimensional stability for higher duty use compared to other zirconium alloys. The PRIME skeleton offers improved corrosion resistance and lower grid growth through material optimizations, lowers assembly pressure drop, and provides more robust debris protection through bottom nozzle design enhancements [3].
Westinghouse EnCore Cr coated fuel cladding refers to a thin, dense, and adherent cold sprayed coating applied to the outer diameter of existing zirconium-alloy fuel cladding. The Cr coated cladding significantly reduces corrosion, hydrogen pickup, and cladding wear during normal operation compared to traditional zirconium-alloy fuel cladding materials. The coating provides enhanced safety

during transient or accident scenarios by significantly reducing high temperature oxidation and improving rupture performance during a loss-of-coolant accident (LOCA) [5][6]. EnCore Cr coated fuel rod cladding has been irradiated in multiple commercial pressurized water reactor (PWR) nuclear power plants, reaching high burnup near 75 GWd/kgU. Poolside examinations as well as destructive PIE in hot cell confirmed full coating adhesion and protection of the zirconium-alloy substrate [7][8]. Westinghouse has decades of boiling water reactor (BWR) operating experience with ADOPT doped UO2 fuel pellets and recent insertion in two PWR reactors supports product implementation in the PWR market. Added dopants include chromia (Cr2O3) and alumina (Al2O3) which result in a larger grain size compared to standard UO2 and pellets improve thermal stability, increase uranium density, and improve oxidation resistance for increased pellet safety and operational margins [3]. Future ATF fuel cladding and pellet concepts for further operational enhancements are still in the research stage and include a silicon carbide (SiC) fuel cladding that will further increase the peak cladding temperature while preventing cladding rupture, and uranium nitride pellets (UN) which significantly increases thermal conductivity and density compared to standard UO2 [9].

2. COMMERCIALIZATION AND LICENSING

While ADOPT pellet fabrication technology is well established, production line capability is being added to Westinghouse Columbia Fuel Fabrication Facility in Columbia, SC, USA. Additionally, to support needs for operational flexibility through longer cycles and high burnups, construction of a new deconversion facility to manufacture ADOPT and UO2 fuels with greater than 5% 235U is underway with projected operability in 2028.

Westinghouse began developing coated cladding materials in 2012 with development efforts increasing after the Fukushima Daiichi event. Since then, collaborations with universities, national laboratories, and external suppliers have led to the development and testing of Cr coatings resulting in an optimized, scaled Cr coated fuel cladding to support LTR insertions. Testing of early development coatings aided in confirmation of performance and refinement of coating specifications [10]. Transitioning from laboratory-scale to full-scale development resulted in the need to adapt existing manufacturing techniques to process and inspect new cladding materials. Lessons learned from development and LTR programs are critical to support future product commercialization. Westinghouse has designed and is manufacturing equipment to produce and inspect Cr coated zirconium-alloy fuel cladding with the manufacturing line supporting initial production demands and projected operability in 2026.

Aside from manufacturing commercialization of ATF HEF technologies, product licensing is also a critical part of new product development to support customer adoption. Significant progress has been made in licensing EnCore HEF technologies and Westinghouse codes and methods to support fuel management strategies that require exceeding the current 235U enrichment limit of 5% and burnups up to 75 GWd/kgU to enable high burnup and 24-month cycles. Several topical reports are already approved for use including ADOPT fuel pellets, reports for increased fuel pellet enrichment and incremental burnup extension, and the Westinghouse PARAGON2™ two-dimensional fuel energy transport code for commercial nuclear fuel modeling of 235U enrichments up to 10%. In addition, the Westinghouse Traveller fuel assembly shipping container package has been approved for 235U enrichments exceeding 5%. The topical report for EnCore Chromium Coated Cladding was submitted for review in 2025. Topical reports under development include extension for high burnups up to 75 GWd/kgU as well as fuel rod design and safety analysis methodology reports to support ATF and HEF technologies. The remaining topical reports are planned for submittal in 2025 and 2026.

3. CONCLUSIONS

In conclusion, tomorrow's needs include 24-month cycles with higher enrichment and higher burnup utilizing advanced fuel technologies to further improve fuel cycle costs as well as safety and operational margins. Westinghouse HEF Program combines both currently available and future advanced fuel technologies to support these critical nuclear industry drivers in a multi-phased approach for PWRs. Recent advancements include ADOPT fuel pellets which feature higher uranium density and safety benefits and Cr coated cladding which offers a reduced corrosion rate and hydrogen pickup, reduced high temperature oxidation, improved fretting resistance and improved LOCA rupture behavior. Lessons learned from early development and LTR programs have enabled product optimization for implementation into fuel fabrication facilities for further commercialization. Various concurrent licensing topical reports further support ATF technologies and extended fuel operability.

ACKNOWLEDGEMENTS

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**5 / 49**

## OVERALL PERFORMANCE OF CR-COATED ACCIDENT TOLERANT FUEL CLADDING IN LARGE PWRS AND IMPLICATIONS FOR SMR APPLICATIONS

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OVERALL PERFORMANCE OF CR-COATED ACCIDENT TOLERANT FUEL CLADDING IN LARGE PWRS AND IMPLICATIONS FOR SMR APPLICATIONS

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After more than a decade of research into the performance of Cr-coated Accident Tolerant Fuel (ATF) cladding, the nuclear industry is now moving toward commercialization, supported by an improved understanding of its overall behavior. In several respects—such as steady-state corrosion resistance [1], steam oxidation resistance [2,3], generally crack-resistant coating behavior at elevated temperatures relevant to normal operation, and slightly reduced inward cladding creep during steady-state

[4]—Cr-coated cladding demonstrates desirable performance. It also exhibits comparable fuel ballooning and burst behavior to uncoated zirconium alloy cladding [5,6], and only limited additional oxidation through cracked coatings near burst openings [6], illuminating the promise of the concept envisioned at its inception.
However, recent findings also highlight inherent limitations that may constrain the achievable operational and safety benefits. These include: (i) degradation of coating protectiveness due to Zr diffusion from the substrate, creating oxygen ingress paths through the Cr layer [3]; (ii) formation of a eutectic phase at around 1330 ℃, which degrades oxidation resistance in steam and causes a sharp increase in hydrogen generation rate, thereby limiting any meaningful increase in the peak cladding temperature limit [7,8]; and (iii) significant loss of post-LOCA ductility due to oxygen uptake and secondary hydriding near the burst opening, leaving the cladding vulnerable to traditional inner-wall oxidation mechanisms [9,10]. Furthermore, Cr-coating does not mitigate over-pressurization or Fuel Fragmentation, Relocation, and Dispersal (FFRD), which remain key safety concerns for high-burnup fuels in the 24-month cycles sought by the industry [5]. In addition, the potential burnup extensions are competed by modern zirconium alloys, whose excellent corrosion resistance is sufficient to support discharge burnup extension of large PWRs (~75 MWd/kgU).
These concerns highlight the importance of rethinking the synergy between advanced fuel materials and reactor design. In line with it, low power density Small Modular Reactors (SMRs) provide operating conditions in which the advantages of Cr-coated ATF can be most effectively realized. Operating at lower linear heat rates giving lower fuel temperature and offering more graceful accident scenarios with no risk of fuel burst, SMRs can fully capitalize on the superior corrosion resistance of Cr coatings. This could enable ultra-long cycles and burnup extension with LEU+ fuels, aligning with operational and economic incentives. We believe that such applications deserve increased attention and could define a future strategic direction for SMR and Cr-coated ATF deployment.
ACKNOWLEDGEMENTS
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## ROLE OF ATF IN ENHANCING SMR PERFORMANCE

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This paper overviews the impact of introducing Advanced Technology Fuels (ATFs) to SMRs (mainly water-cooled), in order to improve their performance while meeting their promised safety goals. Specifically, the application of coated zircaloy cladding, SiC cladding, helical-cruciform fuel, internally and externally cooled annular fuel, Thorium fuels, increasing the BWR and PWR lattice size will be discussed. Almost in all cases, the application of ATFs and/or increase in power density will require fuel enrichment of greater than 5%. Therefore, application of LEU+ and HALEU for SMRs will also be covered.

**1 / 52**

## Research Progress and Prospect of Cr Coated Cladding Using in CNNC

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Chromium(Cr)-coated zirconium cladding has become a leading candidate material for the accident-tolerant fuel (ATF) cladding in recent engineering activities after more than a decade of technological developing. Extensive research has been conducted worldwide, covering the preparation process, performance evaluation, and test verification. The commercial reactor irradiation of Cr-coated cladding fuel rods has been conducted, marking an important transition from technology research and development to engineering application. As one of the main player in nuclear fuel domain, CNNC has developed it's own Cr-coated zirconium cladding based on N36 zircalloy, and loaded some of this type of claddings into CF3 fuel assembly, achieving the in-pile irradiation of the ATF characteristic assembly (Cr-coated cladding,) in 2021, the irradiation in NPP completed in Nov. 2024. The poolside examinations have showed excellent appearance after unloading from the reactor core.
The whole scenario of Cr-coated N36 alloy zirconium cladding developing and the prospects of its using are discussed in this paper. The multi-arc ion plating and high-bonding anti-radiation coating preparation technology of Cr-coated cladding has been established, a large-scale full-size cladding tube coating arc ion plating equipment has been developed, and a small batch of high-quality Cr-coated cladding has been manufactured recently and are going to be loading next year.

Fig. 1. A small batch of full-size Cr-coated cladding
During the development of Cr-coated cladding, key out-of-pile performance tests and verifications were carried out, mainly including:
1) Long-term corrosion studies of coated tube samples in the high-temperature and high-pressure simulated power plant water (360℃/18.6MPa, B=200mg/kg, Li=1.24mg/kg) showed that the corrosion rate of the Cr coating was one order of magnitude lower than that of the zirconium alloy (Fig. 2);
2) The Cr coating would not peel off from the zirconium substrate under high-temperature oxidation conditions, demonstrating good adhesion performance;
3) After being held in a vacuum quartz tube at 700℃-1300℃ for 10 minutes and then rapidly cooled to 100℃(boiling water), the samples remained intact, with no cracking or peeling of the coating observed (Fig. 3);
4) Compared with uncoated tubes, the tensile and burst performance of Cr-coated cladding did not change significantly, but the creep and fatigue performance change was not negligible;
5) Cr-Zr interdiffusion and eutectic melting would occur in Cr-coated cladding at high temperatures (Fig. 4), especially the eutectic melting phenomenon requires further study.

Fig. 2. the corrosion of Cr-coated cladding in the simulated power plant water

Fig. 3. Microscopic surface morphology of coated tube specimens after thermal shock

Fig. 4. Thickness variation of the metal compound layer
In November 2021, the commercial reactor irradiation of the Cr-coated cladding characteristic assemblies was started. By November 2024, the irradiation was completed and the fuel assembly was unloaded. The poolside inspections indicate that the Cr-coated cladding had excellent corrosion resistance in the reactor (Fig. 5).

Fig. 5. Poolside inspection of Cr-coated fuel rods
In order to solid the basis for full scale deploying of coated cladding, the hot cell examinations and the loading of some leading fuel assemblies are in plan. At the same time, the using strategies and economical are researched. The analytical results are going to be presented in thia paper.

**2 / 54**

## High-Temperature Oxidation Behavior of Cr/CrN Multilayer Coatings on Zr–Nb Alloy for Accident Tolerant Fuels

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After Fukushima Daiichi nuclear accident, ATFs (Accident Tolerent Fuels) are being developed to prevent severve accident or to delay the progression of accidents in nuclear power plants. Some of the new designs are focoused on conventional Zr cladding coated with materials that improve cladding behaviour in high-temperature enviroments. This paper presents high-temperature experiments carried out in the temperature ranges of 900-1200 0C and different times in both steam and air enviroments on the Cr/CrN multilayer coated samples. The coatings with the average thickness of about 7µm and the individual layer of about 20 nm have been deposite on the Zr-1%Nb cladding by cathodic Arc-PVD technique. The phase structure and morphology of the coated and uncoated samples have been evaluated via grazing incident angle X-ray difraction (GI-XRD), optical microscopy (OM) and scanning electron microscopy (SEM) equiped with energy dispersive spectroscopy (EDS). The XRD analysis results reveal the presence of three phases: Cr, CrN, and Cr2N in multilayer coated sample. The results show that the thickness of the oxide layer was increased with the increase in oxidation time. In the case of oxidized uncoated samples, holes and micro-cracks were observed in the oxide layer, which can be caused by the increasing stresses due to the growth of the oxide layer. The cross sectional SEM morphologies and EDS results of the oxidized (Cr/CrN) multilayer coated samples indicated that the thickness of remained Cr interlayer of the oxidized sample in steam was more than the oxidized sample in air. Besides, with the increase in the oxidation temperature, the thickness of the oxide layer and the oxygen-rich layer beneath the oxide layer were increased. Finally, the finding demonstrate that the use of alternating Cr/CrN multilayer can be controlled the oxidation rate of the Zr-Nb alloy at high temperatures, resultes in the improve performance of the cladding tube in the in-reactor accident conditions.

**4 / 56**

## EATF safety benefit: EDF point of view.

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Enhanced accident tolerant fuel (EATF) were developed in the wake of Fukushima accident to provide some delays in case of severe accident. Today the highest TRL product is clearly coated cladding, more specifically chromium coated cladding.
EDF is currently investigating the benefits that would be provided by EATF in its own plants. Some

lead test rods are irradiated. They reached their first cycle last year. Their visual appearance is bright as already shown previously.
Based on these observations and using all the data available, EDF has started to assess the benefit of these EATF. The method is simply to compare a standard fuel with an EATF one at a given burnup by extrapolating what is already known or published.
At this point, the benefit of EATF begins to be revealed in various safety files. One key takeaway from all this work is that if hydrogen pick-up is drastically reduced, it would generate margin as hydrogen concentration is the root of many safety criteria today.
In a second step some thoughts about some specificities of EATF such as the loss of adhesion or the lack of coating at some elevation are discussed or even the scratch resulting from the fuel rod insertion into the fuel assembly.
Finally, the specificities of the French closed fuel cycle are adressed as they could bring additional difficulties to the licensing of EATF.

**1 / 57**

## Discussion on Advances in ATF

**2 / 58**

## Discussion

**2 / 59**

## IAEA Fuel DB

**3 / 60**

## Discussion

**4 / 61**

## IAEA PRESENTATION ON FUEL SAFETY ACTIVITY

**5 / 62**

## Discussion: IAEA future activity, including future CRP on ATF, and TM documentation strategy

**4 / 64**

## Discussion on ATF Safety Analysis and Licensing Practices. The recommendations for future activities on the topic, including CRP

**5 / 65**

## Summary of the meeting

The Co-Chairs of the sessions to provide summaries of the session discussion.
The Workshop Co- Chairs to present the outcomes of the discussions and recommendations for future IAEA activities in the ATF area

**5 / 66**

## Debriefing to Dir NEFW

**5 / 67**

## Closure of the meeting