Accident Tolerant Fuel: an update report 2020-2023

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Mr Peter Rudling, Chairman of the Board of ANT International

Contents

Executive summary

This updated version of the ANT International 2021 ATF report will provide the reader with a useful, quick but comprehensive overview of the latest ATF technical developments. It will give relevant information about ATF cladding and fuel, appropriate warnings and useful insights to nuclear fuel engineers and designers as well as to the fuel buyers.

To follow the industry emphasis on the optimization of the first barrier, which is directly in contact with the coolant water or the water vapor during accidental transients, the focus of the report is on ATF cladding tubes. Nevertheless, to improve fuel cycle economics, the nuclear industry is working also on high density, high thermal conductivity fuel concepts but at a slower pace because the industrial scaling of these fuel concepts is more problematic and are considered at longer term only. Nuclear fuel cladding envelops the fuel and is a major barrier against the release of fission products. Traditional zirconium-based fuel cladding materials have served the nuclear industry well due to their excellent high-temperature properties and resistance to corrosion. However, at high temperatures as during accident scenarios, these materials can experience rapid oxidation and hydrogen generation, leading to potentially hazardous scenarios. Accident tolerant fuel cladding materials seek to address these concerns by providing enhanced performance and durability during normal operation and severe accident scenarios, thereby increasing the safety margins of nuclear power plants. The primary goal is to develop materials that exhibit superior corrosion resistance, that could withstand higher temperatures, and that reduce the risk of hydrogen generation compared to traditional zirconium-based alloys.

Three promising approaches for accident tolerant fuel (ATF) cladding are worldwide under development and investigation, namely, Cr-coated zirconium, ferritic FeCrAl alloys, and SiCf/SiC ceramic composites. They all offer promises for improved performance during operation, design-basis accidents, and, at least partially, for safety gains during hypothetical severe accidents. But they also all have outstanding issues that need to be resolved.

Cr-coated zirconium alloys are the near-term solution for ATF cladding materials. They are developed worldwide according to different coating methods with typical Cr coating thicknesses of 5-30 µm. In general, well-produced Cr coatings provide very good performance under harsh operating conditions and result in a 1-2 orders of magnitude reduction in high-temperature oxidation kinetics compared to zirconium alloys. Cr-coated Zr alloys provide among others improved corrosion and oxidation behaviour, less hydrogen uptake and harder surface providing better resistance to fretting wear, with other properties like neutronic and thermo-mechanical behaviour remaining almost unchanged. This entails only minor technological and licensing modifications. A big issue with Cr-coated Zry is the eutectic interaction between Cr and Zr at around 1330°C, which lead to melt formation and almost immediate failure of the protective effect of the coating. Then the advantage of the only slightly modified cladding can turn into the opposite with still 20-50 tons Zr in the core available for the reaction with steam. Cr-coated cladding is the most advanced technology with lead test rods (LTR) and lead test assemblies (LTA) already implemented in commercial reactors in Europe and the USA. It will take a few additional years before LTR under irradiation would reach their qualification burnup levels and be tested in prototypical accidental conditions. In addition, the industrial scaling of the product is still subjected to cost-benefits analysis, depending on fuel vendors commercial offers (which are not yest fully available).

1 Introduction

During the last decade the development of various Accident Tolerant Fuel (ATF) concepts has come into focus of both research and industry communities in the USA, Europe and Asia. The accident tolerant fuel program is aiming towards improving the safety of nuclear energy by investigating materials that can replace or modify the current uranium-dioxide nuclear fuel and zirconium-based cladding. This research programs are being supported by all major nuclear countries since 2011. Several fuel vendors have announced plans to develop and seek approval for various fuel designs with enhanced accident tolerance (i.e., fuels with longer coping times during loss of cooling conditions). The increase of fuel burnup above licensed limits (which varies by vendor, but roughly corresponds to 62 MWd/kg rod-average), as well as increasing enrichment beyond 5% has also become a focus for the advanced fuels. These programs are described elsewhere, and there is a number of recent papers available for the reader to assess the current status and the guidelines of developments, such as (Khatib-Rahbar, Krall, Yuan, & Zavisca, 2020) (Goldner, et al., 2021), (OECD CSNI, 2022), (Rebak R. , 2023), etc.

There is a wide spectrum of various ATF concepts currently under development, and there are many concepts under investigation, including Cr-coated claddings, Cr-doped UO_2 pellets, FeCrAl cladding, SiC cladding, UN pellets, and metallic fuels. Of these concepts, the coated claddings, doped fuel pellets and steel cladding designs are considered to be nearer with respect to the time to commercial deployment. Silicon carbide cladding, uranium silicide fuel pellets and metallic fuels are considered for longer-term deployment.

ANT International has released a STR Accident Tolerant Fuel – A review in 2021 (Mahmood, et al., 2021) which provides an overview and status on the development of various ATF concepts up to mid-2020. The purpose of this report is to present the advance and current status of the most promising ATF concepts currently under development at various fuel vendors, research institutions and nuclear laboratories around the world and to provide the reader an independent assessment of the state of the art and potential for development and implementation related with each type of the ATF.

This report is divided into several sections, addressing the development of fuel claddings, fuel materials, ATF qualification and licensing and ATF economic considerations.

The editorial date for the date used in this report is October 2023.

2 Update on ATF cladding

Nuclear fuel cladding envelops the fuel and is a major barrier against the release of fission products. Traditional fuel cladding materials, such as zirconium-based alloys, have served the nuclear industry well due to their excellent high-temperature properties and resistance to corrosion. However, under certain extreme conditions, these materials can experience rapid oxidation and hydrogen generation, leading to potentially hazardous scenarios. Accident tolerant fuel cladding materials seek to address these concerns by providing enhanced performance and durability during normal operation and severe accident scenarios, thereby increasing the safety margins of nuclear power plants.

After the Fukushima Daiichi accident in 2011, there was a global surge in research and development efforts focused on accident tolerant fuel (ATF) cladding materials. Governments, nuclear regulatory bodies, and industry stakeholders recognized the need to enhance the safety of nuclear reactors and mitigate potential severe accident scenarios. Numerous countries, including the United States, Japan, and France, among others, have launched dedicated research programs to explore and develop advanced ATF cladding solutions. The primary goal was to develop materials that exhibit superior corrosion resistance, could withstand higher temperatures, and reduce the risk of hydrogen generation compared to traditional zirconium-based alloys.

The main materials systems that emerged as promising candidates for accident tolerant cladding include chromium-coated zirconium alloys, iron-chromium-aluminium (FeCrAl) alloys, and silicon carbide (SiC) composite cladding (Rebak R. , 2023).

Other potential solutions have been abandoned due to technical reasons (e.g., fabrication and welding issues of molybdenum cladding) or bad high-temperature oxidation resistance or are in an early stage of development progress. These include e.g. other types of coatings (e.g., ceramic, multilayer, and MAX phases).

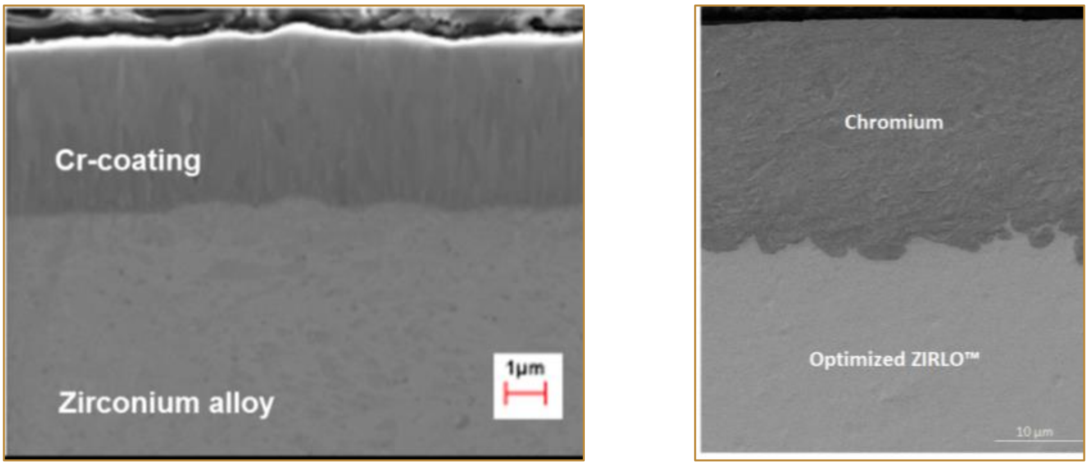
This section will concentrate on the three most promising ATF cladding candidates mentioned above. The accident tolerance of all three materials relies on the formation of protective oxide scales during high-temperature oxidation in water steam atmospheres, namely chromia (Cr_2O_3) on chromium coating, alumina (Al_2O_3) on FeCrAl alloys, and silica (SiO_2) on silicon carbide. These three oxides are known by materials scientists as the only protective ones under the extreme conditions of severe nuclear accidents. For each of the cladding materials, a summary of the basics and the status of knowledge from the 2020 report (Mahmood, et al., 2021) will be given, followed by a description of new research results since then and a discussion of the remaining challenges and gaps.

2.1 Cr-coated zirconium alloys

Cr-coated zirconium alloys are the near-term solution for ATF cladding materials, which should be associated with only moderate changes in technology and licensing procedures. They are developed worldwide according to different coating methods with typical Cr coating thicknesses of 5-30 μm. In general, well-produced Cr coatings provide very good performance under harsh operating conditions and result in a 1-2 orders of magnitude reduction in high-temperature oxidation kinetics compared to zirconium alloys. The first lead test assemblies (LTAs) are already being used in commercial reactors in the US and Europe. Recent reviews have been published, for example, by Bischoff (Bischoff, et al., 2018), Brachet (Brachet, et al., 2019) (Brachet, et al., 2020), Maier (Maier, et al., 2018), Yeom (Yeom, et al., 2019), Yang (Yang, et al., 2022), and Li (Li, et al., 2023).

2.1.1 Basic information and summary of the 2020 report

Chromium coatings on zirconium alloys can be produced by various technologies, including Physical Vapor Deposition (PVD), Chemical Vapor deposition (CVD), Electrodeposition, Thermal Spray Deposition (TS), Cold Spraying (CS) and Three-dimensional Laser Coating, which are described in more detail in the 2020 report (Mahmood, et al., 2021). Different PVD methods are applied as direct current magnetron sputtering (dc-MS), arc ion plating (AIP), and high-impulse-power magnetron sputtering (HiPIMS) (Wang Z. , et al., 2023). The main methods applied for nuclear cladding are PVD and CS. This is also reflected by the two most advanced industrial Cr-coated cladding tubes. Framatome (France and US) applies PVD to manufacture dense and well-adherent 10-20 μm thick chromium coating for its PROtect Cr-coated $M5_{Framatome}$ cladding (Bischoff, et al., 2018) (Vioujard, Lewis, Maxson, & Reed, 2022). On the other hand, Westinghouse within the EnCore® Fuel program (Lahoda, et al., 2018) (Karoutas, et al., 2021) applies Cold Spray and subsequent polishing to get 20-30 μm thick coating with smooth surface, but compared to PVD with more wavy Cr/Zr interface, as seen in Figure 2-1, due to the special features of the CS process. Global Nuclear Fuel (GNF) in the US is developing a so-called ARMOR (Abrasion Resistant, More Oxidation Resistant) proprietary coating applied to the outer surface of normal production Zircaloy fuel rods.



PVD coating by Framatome (Bischoff, et al., 2018) CS coating by Westinghouse (Shah, et al., 2018)

Figure 2-1: Typical chromium coatings produced by PVD and CS

Framatome, Global Nuclear Fuel (GNF) and Westinghouse are working with the US Department of Energy (DOE) to commercialise their ATF concepts by 2025. Further R&D on Cr-coated cladding is conducted in South Korea with Cr, Cr-Al and FeCrAl/Cr coatings on partial oxide dispersion-strengthened (ODS) Zr alloy (Kim H.-G. , et al., 2016). Under the framework of the national ATF R&D program led by CGN (China General Nuclear Power Group), tremendous efforts on ATF research are going on in China (Liu, et al., 2018) with several detailed research papers published every month, but almost no open information on the programmatic progress in China.

2.1.1.1 Normal operation conditions

Although Cr-coated Zr alloys were initially developed to enhance the oxidation resistance of the cladding materials under accident conditions, the coated cladding should provide the same or better properties as conventional cladding during the long-term normal operation in high-temperature and high-pressure aqueous environment.

Generally, well-produced Cr coating are very adherent to the Zr alloy substrate. The advantage of using a coating rather than changing the bulk cladding is that most mechanical properties of the cladding are governed by the substrate and not by the coating, especially with thin coatings. Cr coating is harder than the zirconium alloy with the benefit of potentially protecting the cladding against fretting and wear. Preliminary studies have shown that Cr-coating is very protective against cladding wear and therefore may significantly reduce the risk for cladding damages due to debris or grid-to-rod fretting (Bischoff, et al., 2018) (Brachet, et al., 2019). Tensile strength of Cr-coated samples is close to that of uncoated samples, but the deformation strain of coated samples is lower than of coated ones (Li, et al., 2000). Tensile and compression tests conducted in Korea also showed Cr-coated samples slightly stronger than the uncoated samples (Kim H.-G. , et al., 2015).

The impact of the Cr coating on the thermal hydraulic behaviour in the reactor is shown to be negligible. One could expect negligible effects on the hydraulic resistance, core bypass flow, hydraulic forces and bulk

3 Update on ATF fuel concepts

The objective of this section is to review the fuel materials that are intended to improve the resistance of water reactor fuel to accident conditions or mitigate the effects of such events. This review focuses on fuel for the existing fleet of PWR and BWR's. The fuel in these plants utilize oxide fuel pellets with Zr-alloy cladding in the form of varying arrays of long cylindrical rods. This fuel comprises cylindrical pellets consisting of UO_2, $(U,Gd)O_2$ or similar compounds with other neutron absorbers and, less frequently, $(U,Pu)O_2$ (MOX). These fuel materials are sintered to high densities, ground to specified diameters and placed in Zr-alloy tubes, almost always with high purity helium at fill pressures which vary among reactor types and fuel designs. The review of fuel materials focuses primarily on fuel materials that could be adopted in the near future with either Zr-alloy or Fe-alloy cladding as identified in the earlier ANT report (Mahmood et al, 2021). It also includes discussions of alternatives that require more development but that appear to offer significant advantages.

The potential accident tolerant fuel materials can be categorized based on their respective characteristics such as:

- Improved UO_2 fuel,
- High-density, high thermal conductivity fuel,
- Encapsulated fuel, in which the encapsulation is intended to improve fission product retention and/or to allow use of high-conductivity, high-density fuel material in water-cooled reactors.

Potential ATF materials can also being categorized based on their expected benefits, risks and time necessary for implementation. Near-term alternatives are those involving minor variations from existing UO_2 fuel that can potentially be implemented with minimal development and minimal risk. Longer-term alternatives are those that offer greater potential benefit but are associated with need for larger research efforts and greater risks or uncertainties.

3.1 UO_2 fuel in water cooled reactors

In general, the improved UO_2 fuel can be divided into two groups. First is based on cationic dopants that were developed to mitigate fission gas release and enhance PCI failure resistance. This fuel type typically includes small amounts of Cr_2O_3 or Cr_2O_3+Al_2O_3 added to UO_2 matrix. The fuel already exists as a commercial product by Framatome and Westinghouse. It has been proposed as a near-term, accident-tolerant fuel material. The second category is based on the addition of ceramic or metallic additives that are largely insoluble in the fuel matrix and form grain-boundary phases that surround ("microencapsulate") individual grains of UO_2. The objectives of these microencapsulating additives to either improve the retention of fission products in the fuel matrix or improve thermal conductivity and thereby reduce fuel temperature and improve temperature-related effects such as fission gas release and stored thermal energy. Basic information and summary of the 2020 report

3.1.1 Basic information and summary of the 2020 report

3.1.1.1 UO_2 with chromia-based dopants

Chromia-doped UO_2 is considered as an accident-tolerant fuel material. As such, it provides some benefits relative to standard (small grain) UO_2 with respect to the retention of fission products, greater pellet compliance at high temperatures and PCI resistance. These benefits primarily relate to the conditions of normal operation and anticipated operating occurrences providing improvements in margins to design basis accidents due to greater (nevertheless rather limited) retention of fission gases. With respect to fuel behaviour important in severe accidents, chromia-doped fuel is rather comparable to standard UO_2 and any benefits relative to severe (beyond design-basis) accidents are expected to be limited.

3.1.1.2 UO2 with Si-Ti-Al-O additives

Another type of dopant proposed as an accident tolerant fuel involves the addition of oxides that are largely insoluble in UO_2 and that surround individual grains or agglomerations of grains in the fuel matrix. The resulting pellets are similar to the Al_2O_3+SiO_2 (Al-Si-O) additives which were developed in the 1970s in parallel with zirconium-lined cladding as remedies to the PCI failure mechanism.

The performance of ceramic microcell fuel in a reactor environment is unknown. There are no reports of irradiation tests of ceramic microcell fuel in open publications. Also, emphasis seems to have shifted from the ceramic microcell to pellets with metallic additives based on recent reports.

The potential value of ceramic microcell UO_2 with respect to enhanced accident tolerance is not yet clearly identified. The proposed additives may provide benefits during the conditions of normal operation and anticipated operating occurrences, similar to UO_2 with Al-Si-O additives and chromia dopants; e.g., larger grains, slightly lower fission gas release at normal PWR conditions and increased resistance to the PCI failure process. They do not address issues related to thermal conductivity, stored energy, melting temperature and behaviour at the high temperatures resulting from sustained time without coolant. As a result, the benefits of such additives with respect to severe accidents are likely to be limited (Mahmood et al, 2021).

3.1.2 New information published during the recent 3 years

The Chromium (Cr_2O_3)-doped pellets are at advanced stages of development with many years of accumulated operational experience and the fuel vendors currently are implementing this fuel type into their licensing and analysis methods.

In 2021, Framatome submitted a request to the Nuclear Regulatory Commission (NRC) for the review and approval of a topical report titled “Incorporation of Chromia-Doped Fuel Properties in Framatome PWR Methods.”. The report discusses the implementation of chromia-doped fuel modelling in Framatome Pressurized Water Reactor (PWR) methodologies as part of their Enhanced Accident Tolerant Fuel (EATF) Program. The approval of this supplement by the NRC is important for Framatome’s near-term solution for PWRs. The document also mentions that the approval of chromia-doped pellets for PWRs is a prerequisite for the submission of another topical report for a combined product of chromium-coated M Framatome cladding with chromia-doped pellets, planned for 2023.

Similarly, Westinghouse in 2020 had submitted Topical Report “Westinghouse Advanced Doped Pellet Technology (ADOPT) Fuel” which discusses the development and characteristics of ADOPT fuel. The report concludes that ADOPT fuel can be used as a direct replacement for UO_2 fuel in existing pressurized water reactor designs, offering improved performance in terms of burnup capability and accident tolerance. The report provides detailed information on the microstructure, thermal properties, mechanical properties, and irradiation programs and experience of ADOPT fuel. It also assesses the impact of ADOPT fuel on licensing criteria and nuclear design requirements.

The main conclusions of this report are:

- ADOPT fuel, with its modified uranium dioxide composition incorporating chromia and alumina, offers enhanced properties that allow for higher burnup and improved AOOs and accident tolerance compared to standard UO_2 fuel.
- ADOPT fuel can be used as a direct replacement for UO_2 fuel in existing pressurized water reactor designs. Subsequent licensing submittals will further explore the benefits of ADOPT fuel.
- ADOPT fuel demonstrates similar thermal properties to standard UO_2 fuel, with no significant difference in thermal diffusivity.
- The mechanical properties of ADOPT fuel, including creep behaviour, are comparable to standard UO_2 fuel.

- ADOPT fuel has a minimal impact on density and elastic moduli, and the rule of mixtures can be used to estimate its theoretical density.

- ADOPT fuel meets applicable regulatory requirements, and the report provides a regulatory roadmap for its use, referencing specific sections of NRC guidelines.

The report concludes that ADOPT fuel offers improved performance in terms of burnup capability and accident tolerance, making it a viable fuel option for commercial nuclear reactors.

HALEU fuel contains uranium enriched to between 5% and 20% U^{235} - higher than the uranium fuel used in light-water reactors currently in operation, which typically contains up to 5% U^{235}. Most of the development is currently on-going in the US.

All the three fuel vendors in the US are preparing for the increase of the fuel enrichment. Framatome in 2021 have released a topical report on the increased enrichment for PWR's (Framatome, 2021), Westinghouse has applied for incremental increase of burnup for Westinghouse and Combustion Engineering fuel designs (Westinghouse, 2020) and GNF has applied for a license amendment to request up to 8%wt enrichment of U^{235} (LEU+) in 2022.

Westinghouse Electric Company has received approval from the US Nuclear Regulatory Commission (NRC) to use its ADOPT fuel pellets in US pressurised water reactors (PWRs). An agreement was announced in 2022 with Southern Nuclear Company to load rods using High-Assay Low-Enriched Uranium (HALEU) ADOPT pellets "with licensing and manufacturing in 2023". The fuel is to be placed in Southern Nuclear Operating Company (SNC) Vogtle Electric Generating Plant (Vogtle) Units 1 and 2. Four Lead test Assemblies (LTAs) will be introduced, with coated AXIOM cladding and ADOPT fuel pellets enriched up to 6 wt% U^{235}.

It is stated that the HALEU will be needed by most of the advanced reactor designs being developed under the US Department of Energy's (DOE's) Advanced Reactor Demonstration Program. The lack of a commercial supply chain to support these reactors has prompted the DOE to launch a programme to stimulate the development of a domestic source of HALEU. In October 2023 it was announced that US nuclear fuel and services company Centrus Energy Corp has begun enrichment operations at the American Centrifuge Plant in Piketon, Ohio. The American Centrifuge Plant is the only HALEU facility in the USA licensed by the Nuclear Regulatory Commission (NRC) and the first new US-owned, US-technology uranium enrichment plant to begin production since 1954. Currently the capacity of the 16-centrifuge cascade is rather modest with about 900 kilograms of HALEU output per year but with additional funding and commitments, the company could significantly expand production.

3.1.2.1 Ceramic additives

UO_2-BeO composite fuel

The development activity has included conceptual studies of thermal conductivity, fuel temperature, ^{235}U enrichment requirements, energy generation capabilities and the potential effects of BeO on fuel behaviour during postulated accident conditions. The activities have also involved the development of methods for producing composite fuel pellets and optimizing the spatial distribution of BeO in the composite; i.e., the shape, volume fraction, homogeneity and orientation of the BeO phase.

The experimental work published earlier shows that [Mahmood, 2021):

- Thermal conductivity increases with the concentration of BeO;

- The increase is greater in composite pellets with continuous BeO phase, but also exists in pellets with dispersed BeO particles to a lesser extent;

- The increase in thermal conductivity of composite pellets relative to standard UO_2 decreases with increasing temperature but is still higher even at the upper end of normal operating temperatures for UO_2 pellets.

4 ATF qualification and licensing

4.1 General considerations on licensing processes of new nuclear fuels

Acceptance criteria and guidance documents, captured in national regulations, have been used to carry out the safety assessment (and licensing) of the current generation of nuclear fuel (i.e., UO_2 ceramic pellets within zirconium alloy cladding) (Mahmoodl, 2021), (IAEA TECDOC), (SSG-52) and (Crawford, Porter, Hayes, Meyer, & Petti, 2007). The efficiency of fundamental safety principles and safety requirements have been confirmed over time in their ability to protect against or limit damage to the reactor core, provided licensees ensure fulfilment of acceptance criteria and ability of analytical models and methods to predict, with a high level of confidence, fuel performance under a wide range of operational and accidental conditions. For example, to demonstrate that the fuel can mitigate the consequences of a Reactivity Initiated Accident (RIA), validated analytical models, enabling good prediction of fuel rod behavior are needed, along with analytical limits that ensure acceptable fuel performance (e.g., maximum enthalpy, maximum injected energy, and maximum cladding temperature).

The ATF initiative launched by the American Department of Energy (DOE) in 2012, and the need expressed by the operators to reduce fuel cycle costs by using more robust fuels, incented US-NRC to update the usual licensing process to facilitate the implementation of promising ATFs (which are supposed to be higher performance).

4.1.1 Fuel Qualification

Fuel development and qualification activities often occur in parallel. Building on past experience, the Idaho National Laboratory published a paper describing a detailed approach for the development and qualification of new fuel for light water reactor (LWRs) (Crawford, Porter, Hayes, Meyer, & Petti, 2007) , which stated that: "*the approach is described as four phases, with emphasis on selecting a reference fuel concept, (i) evaluating and improving the fuel, (ii) developing fuel specification for a reference design, (iii) obtaining data to support a licensing safety case, and (iv) qualifying the fuel for a specific application*."

As such, qualify a new fuel consists in demonstrating that the new fuel product, which must be fabricated according to well defined technical specifications, behaves as expected (i.e., as described in the applicable licensing safety case), with a high level of reliability, enabling economic operation of the reactor.

The United States Nuclear Regulatory Commission (U.S. NRC) published the NUREG-2246, Fuel Qualification for Advanced Reactors (Nuclear Regulatory Commission, March 2022) "*to identify criteria that will be useful for advanced reactor designers through an assessment framework that would support regulatory findings associated with nuclear fuel qualification*". This framework provides criteria, derived from regulatory requirements, which, once satisfied, will support the licensing by the safety authority.

Reference (Nuclear Regulatory Commission, March 2022) defines the US-NRC general licensing process, "*the assessment framework[2] particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate evaluation models and empirical safety criteria".*

In general, once the fuel design and manufacturing specifications have been finalized, the next stage in fuel qualification is to obtain the data needed to support the licensing safety case. The first step is to fully characterize the mechanical, material, thermal, chemical, and nuclear properties, and the impact of irradiation under reactor coolant conditions on these properties. Separate effect tests and integral tests of

[2] The term "framework" is intended to include the laws, regulations, requirements, guidelines and practices for the review, assessment and approval of nuclear fuel by the NRC.

fresh and irradiated fuel segments will be required to characterize these properties for the full range of operating conditions including design basis accidents (DBA). One of the main goals of this research and testing is to satisfy the following fuel qualification needs:

- Identify all degradation mechanisms and failure modes[3] (US-NRC).
- Establish the fuel performance metrics (i.e., the evaluation parameters) adapted to the fundamental safety principles, safety requirements, and regulatory requirements.
- Define the acceptance criteria (i.e., analytical limits) adapted to the performance metrics.

Another major and complementary goal of these experimental investigations is to gather data necessary to calibrate and validate analytical models, as well as establish model prediction uncertainties. The ability of analytical models and methods to predict, with a high level of confidence, fuel performance under normal operation and accident conditions is needed to demonstrate that fundamental safety principles and safety requirements are satisfied, and that safety related structures, systems, and components (SSCs) perform their intended safety functions. Reference (Nuclear Regulatory Commission, March 2022) of US-NRC provides a systematic evaluation and justification of a new fuel, such enabling its qualification.

US-NRC licensing framework inspired other international documents on fuel licensing processes. As an example, the Specific Safety Guide SSG-52 of IAEA "*Design of the Reactor Core for Nuclear Power Plants*" (IAEA SSG-52 Design of reactor core for LWR) (SSG-52) defines and list the main safety and design limits relevant to LWR reactor core design. IAEA is about to release an extension of the SSG-52, whose purpose is to analyze the applicability of the current safety and design limits to Advance Fuel Technology or Accident Tolerant Fuel. The title of this Technical Document (TECDOC), to be issued shortly, is "*Status of Knowledge for the Qualification and Licensing of Advanced Nuclear Fuels for Water Cooled Reactors*" (IAEA TECDOC).

OECD-NEA has been working since 2012 on ATFs assessment, together with DOE who launched its initiative at the same period. The Expert Group on ATF for Light Water Reactors (EG-ATFL) overviewed the ATF concepts, assessed their TRL (Technical Readiness Levels) and defined the metrics to be used to quantify the ATF performances. An OECD-NEA State-of-the-Art Report, written by the EG-ATFL, has been issued in 2018 (OECD CSNI, 2018) : "*State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels*" (OECD-NEA SOAR-Accident-Tolerant-Fuels-2018.pdf). The ATF review is quite comprehensive and gives a good idea of the state of knowledge and the remaining gaps.

This document has been followed in 2022 by the CSNI Technical Opinion Paper N°19 "*Applicability of Nuclear Fuel Safety Criteria to Accident-Tolerant Fuel Designs*" (CSNI-technical-opinion-paper-no-19) (OECD CSNI, 2022).

Both the IAEA TECDOC and the NEA Technical Opinion Paper are focusing on the applicability of the current licensing approach (i.e., the applicability of the current fuel safety and design limits) to the proposed ATF variants. In other words, NEA and IAEA considered that identifying the potential gaps and therefore defining new criteria or design limits, were beyond their scope of work.

4.1.2 Lead Test Rods and Lead Test Assemblies

Lead test rods (LTR) and lead test assemblies (LTAs) are a necessary and important step in the fuel development and qualification process. LTRs and LTAs are required when the design evolution is significant. To evaluate whether a design modification is significant or not significant, is always debatable.

[3] To support this approach US-NRC has sponsored PIRT (Phenomenon Identification and Ranking Tables) exercises on various ATF concepts and operating conditions (the PIRT on Cr coated cladding and FeCrAl has been completed in 2019, the PIRT on higher Burnup, higher enrichment and source term has been completed in April 2021). Currently, the NRC is sponsoring a PIRT exercise regarding Fuel Fragmentation Relocation and Dispersal (FFRD). The final report is expected in Spring 2024. In the near future, additional PIRT exercises are being considered to cover spent fuel storage and transport of fuel with iron-chromium aluminum (FeCrAl) cladding and with chromium-coated cladding. See (US-NRC) https://www.nrc.gov/reactors/power/atf/pirt.html#completed

5 ATF economics considerations

In section 4 of the present document and in Section 7 of the 2021 ANT International ATF report-a review (ANT International, 2021), it was mentioned that US regulator has agreed to make an exception enabling accelerated ATF licensing (see a detailed description of the process in section 7 of (ANT International, 2021). In European countries, potential benefits brought by the ATF have been recognized by the European Commission[12], however European regulators do not seem inclined to facilitate the licensing process of ATFs to accelerate their commercial deployment.

Most of the short-term ATF concepts (with a high Technical Readiness Level) are still under irradiation through experimental irradiation programs in commercia nuclear reactors (LTRs and LTAs). Mass production of these mature ATFs (e.g., Cr coated claddings, FeCrAl and doped pellets for PWR) have not started yet, the stakeholders are waiting for the final outcomes of the in-reactor experimental irradiation programs and a declaration of interest from the nuclear operators.

As a matter of fact, the latest R&D results on ATFs have shown that the resistance of proposed ATF fuel at high temperatures (HT) transients, following a loss of coolant accident (LOCA), is a little better than the standard fuel but do not provide significant additional coping time to the plant operator. The gain in terms of available grace time before fuel core degradation occurs (before ECCS plays its role) is only a few minutes (or a few tenths of minutes).

In other words, the candidate ATFs are not the real game changers expected by the nuclear industry to mitigate the consequences of LOCAs and simplifying LWRs by declassifying some of the safety systems, to reduce the operational costs. Nevertheless, it must be recalled that post-LOCA grace time is not a safety criterion, meaning that potential benefits offered by ATFs could be evaluated against other parameters. A good example is given by the Cr coated cladding: The gain on the coping time is negligible but LOCA tests are showing that balloon size and burst opening of Cr coated claddings are reduced as compared to standard Zr Alloy cladding, such reducing the opportunity for the fuel to fragment and relocate in the ballooned area and to disperse in the coolant. This is very beneficial regarding the usual LOCA safety limits (i.e., PCT, ECR, hydrogen generation, and radiological source term).

Therefore, ATFs benefits should be found elsewhere, mainly in normal operation and AOOs conditions (e.g., better fretting wear resistance, better resistance to boiling transition, lower risk of accelerated waterside corrosion, higher BU, longer cycles, etc.…). To illustrate this change of paradigm, the US nuclear industry is promoting now AFTs (Advanced Fuel Technologies) rather than ATFs (Accident Tolerant Fuels). The strategy is to convince the end-users that it is worth switching from current fuel designs, although they are reasonably performant, to advanced fuel technologies, to reduce operating costs. Therefore, to achieve this goal, US nuclear industry is now focused on:

1. Increasing ^{235}U enrichment to enable longer fuel cycles and higher Burnups,
2. Increasing fuel management flexibility by relaxing some of the design constrains.

The first objective requires in depth investigations to confirm higher burnups and increased fuel enrichments can be demonstrated in the current regulatory framework. One of the key topics to be analyzed is the FFRD (Fuel Fragmentation, Relocation and Dispersal) issue: it is known that burnup significantly increases the risk of fine fragmentation and relocation of the fragments in the ballooned area of the fuel rods, such affecting PCT (peak cladding temperature) in case of LOCA like transients. The current FFRD threshold for standard UO_2 is around 60GWd/tM (fuel rod average burnup). Since nuclear industry wants to reach burnups in the order of 75 GWd/tM to minimize fuel cycle costs, it is paramount to demonstrate that AFT is going to behave better. To this aspect, Cr coated cladding (which minimize

[12] When defining the Taxonomy of energy sources eligible to “green” investments, the European Commission (EC) first dismissed nuclear energy. Then, after a lot of debates, and after JRC issued its report arguing that nuclear energy doesn’t harm environment, EC decided to partially include nuclear energy in the taxonomy, provided ATFs be used by the nuclear operators.

balloon sizes) and doped fuel pellet (whose lower propensity to fine fragmentation at high burnup still need to be confirmed) are viewed as good means to reach this goal.

The second objective is to challenge the current regulation implying that any fuel rod experiencing a boiling transition (BT) phase, in normal operation or AOOs conditions, must be considered as failed. This is not a criterion per se, but such a requirement significantly restrains fuel core management flexibility. As a matter of fact, testing is showing that fuel rods can easily survive a BT phase and be re-used in a safe way. The criterion would be then the maximum allowable "time at temperature" without fuel damage. This time at temperature would be evaluated using mechanistic calculation tools and appropriate DNB (Departure of Nucleate Boiling) correlations. The idea is then to use the longer time at temperature authorized by the AFTs to relax the "systematic post-DNB failure requirement"[13] and gain some fuel management flexibility. The stakeholders think that additional flexibility provided by these higher performance AFTs, could significantly reduce fuel cycle costs, although figures are not yet available.

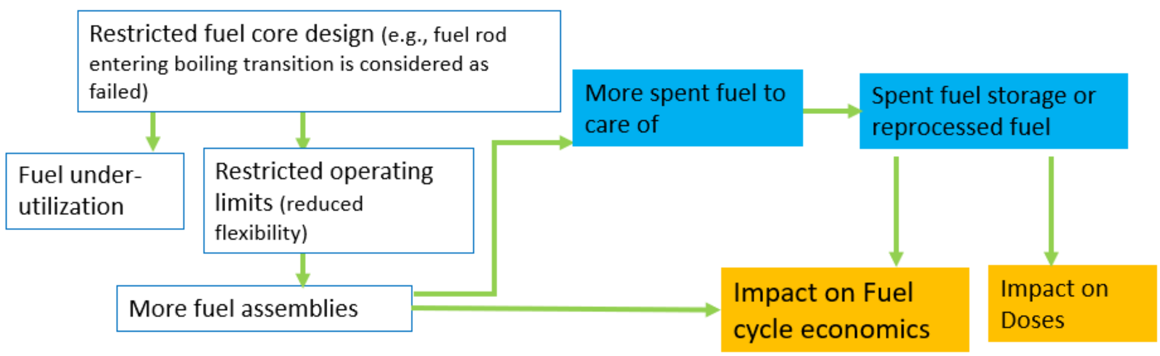


Figure 5-1: Excess conservatism in fuel integrity criteria places an inordinate burden on fuel cycle economics (EPRI, 2023)

All in all, it is a complex task to assess the potential savings brought by a new type of fuel. Obviously, the gains (if any) will be shared among the various stakeholders (i.e., the fuel vendors and the nuclear operators). To quantify the additional manufacturing costs of a new fuel is difficult for proprietary reasons, and the additional operational savings brought by the same new fuel implies many actors, including the domestic regulators. EPRI tried to quantify the benefits in 2018-2019 (ANT International, 2021), using speculative assumptions which have not yet been confirmed. To our knowledge, no new evaluations have been made publicly available since 2020; it seems that the stakeholders are on standby, waiting for the other party, the fuel vendors on one side and the nuclear utilities on the other side, to make the first move.

A few factors are explaining this status:

- DOE has stopped recently to fund R&D on high TRL ATFs (e.g., Cr coated Zr Alloys, doped fuels), and subsidies are now reserved to long term fuel concepts (e.g., SiC/SiC, FCM or TRISÖ fuel), leaving to the industry the responsibility to finalize the licensing and commercialization of these fuel products.

- To license complete fuel reloads, the results of the PIEs to be performed on LTRs, LTAs under irradiation must be completed and integrated in the licensing process of the new fuel products. To reach a bounding BU range, it takes around 6 years in a commercial nuclear power plant and an additional couple of years to cool down and ship the selected irradiated fuel rods, and to perform the PIEs in the hot labs. In addition, tests must be performed in test reactors (whose availability is scarce), e.g., in-pile RIA tests for high BU doped fuel rods and LOCA FFRD

[13] Such an alternative criterion exists already for low frequency transients' applications in NUREG-0562, "Fuel Rod Failure as a Consequence of Departure from Nuclear Boiling or Dry-out," June 1979. The idea is to generalize this approach for frequent AOOs.

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Unit conversion

TEMPERATURE		
°C + 273.15 = K	°C × 1.8 + 32 = °F	
T(K)	T(°C)	T(°F)
273	**0**	32
289	16	61
298	25	77
373	**100**	212
473	**200**	392
573	**300**	572
633	360	680
673	**400**	752
773	**500**	932
783	510	950
793	520	968
823	550	1022
833	560	1040
873	**600**	1112
878	605	1121
893	620	1148
923	650	1202
973	**700**	1292
1023	750	1382
1053	780	1436
1073	**800**	1472
1136	863	1585
1143	870	1598
1173	**900**	1652
1273	**1000**	1832
1343	1070	1958
1478	1204	**2200**

Radioactivity	
1 Sv	= 100 Rem
1 Ci	= 3.7×10^{10} Bq = 37 GBq
1 Bq	= 1 s^{-1}

MASS	
kg	lbs
0.454	**1**
1	2.20

DISTANCE	
x (µm)	x (mils)
0.6	0.02
1	0.04
5	0.20
10	0.39
20	0.79
25	0.98
25.4	**1.00**
100	3.94

PRESSURE		
bar	MPa	psi
1	0.1	14
10	**1**	142
70	7	995
70.4	7.04	**1000**
100	10	1421
130	13	1847
155	15.5	2203
704	70.4	**10000**
1000	100	14211

STRESS INTENSITY FACTOR	
MPa√m	ksi√inch
0.91	**1**
1	1.10