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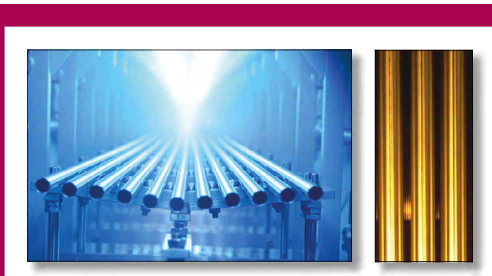
CSNI Technical Opinion Paper No. 19

Applicability of Nuclear Fuel Safety Criteria to Accident-Tolerant Fuel Designs



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Nuclear Safety

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Applicability of Nuclear Fuel Safety Criteria to Accident-tolerant Fuel Designs

NEA No. 7576

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The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;
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Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

Cover photos: Cr-coating of full-length M5Framatome cladding tubes by PVD Process (Framatome); Cr-coated cladding tubes (Framatome).

Foreword

The Nuclear Energy Agency (NEA) Working Group on Fuel Safety (WGFS) is tasked with advancing understanding of nuclear fuel safety issues by assessing the technical basis for current safety criteria and their applicability to high burnup and to new fuel designs and materials. The group aims at facilitating international convergence in this area, including as regards experimental approaches and interpretation and the use of experimental data relevant for fuel safety.

One of the current key topics in the fuel safety area is the applicability of safety criteria to new fuel designs, including for accident-tolerant fuels (ATFs). A dedicated task group worked on this topic from 2018 until 2020, developing its assessment from previous NEA work on nuclear fuel safety criteria, *Nuclear Fuel Safety Criteria Technical Review* [2] and from the compiled knowledge base for new ATFs designs, *State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels* [3].

The task group produced a report that constitutes a reference on the applicability of fuel safety criteria to new ATF designs and on future research and development needed to support ATF licensing.

Acknowledgements

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Special thanks go to Paul Clifford (Nuclear Regulatory Commission, United States) for leading the task group, drafting the report, and for consolidating the various opinions, comments and remarks.

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List of abbreviations and acronyms

AOO	Anticipated operational occurrence
ATF	Accident-tolerant fuel
BWR	Boiling water reactor
CHF	Critical heat flux
CSNI	Committee on the Safety of Nuclear Installations (NEA)
CS	Cold spray technique
DBA	Design basis accident
DNB	Departure from nucleate boiling
ECR	Equivalent-cladding reacted
FG	Fission gas
EGATFL	Expert Group on Accident-tolerant Fuels for Light Water Reactors (NEA)
FFRD	Fuel fragmentation, relocation, and dispersal
FGR	Fission gas release
FP	Fission product
INL	Idaho National Laboratory (United States)
LOCA	Loss-of-coolant accident
LTA	Lead test assembly
LTR	Lead test rod
LWR	Light water reactor
NEA	Nuclear Energy Agency
ODS	Oxide dispersion strengthened
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory (United States)
PCI	Pellet-to-cladding interaction (often associated with stress corrosion cracking)
PCMI	Pellet-to-cladding mechanical interaction
PIE	Post-irradiation examination
PWR	Pressurised water reactor
PVD	Physical vapour deposition
RIA	Reactivity-initiated accident
SCC	Stress corrosion cracking
SEM	Scanning electron microscope
SiC	Silicon carbide
TOP	Technical Opinion Paper
TRL	Technology readiness level
WGFS	Working Group on Fuel Safety (NEA)

Executive summary

Following the 2011 Great East Japan Earthquake and the resulting nuclear accident at the Fukushima Daiichi Nuclear Power Plant, global interest has expanded in researching and developing nuclear fuels with enhanced accident tolerance. Such accident-tolerant fuels (ATF) include improved designs, materials, and performance features beyond the current generation of slightly enriched UO_2 ceramic pellets within cylindrical zirconium alloy cladding.

The Nuclear Energy Agency (NEA) Expert Group on Accident-tolerant Fuels for Light Water Reactors (EGATFL) investigated a large number of advanced fuel designs and materials, and defined a technology readiness level based upon the existing empirical database supporting the characterisation of material properties and performance, reported in the publication *State-of-the-Art Report on Light Water Reactor Accident Tolerant Fuels* [3].

The purpose of this Technical Opinion Paper (TOP) is to (1) evaluate the applicability of existing fuel design and performance requirements to each of the five ATF designs; (2) identify new phenomena or mechanisms which create the need for new or different performance metrics and design requirements; (3) identify data gaps; and (4) discuss opportunities for international collaborative research to fill these data gaps. The information shared in this document reflects the knowledge collected up to October 2020. This paper is intended to inform future international research programmes and support ATF licensing.

By cross-referencing the report of the EGATFL with results from an international survey of most likely ATF candidates for deployment, the scope of this report was narrowed down to the following ATF technologies:

Coated zirconium alloy fuel rod cladding	- Chapter 2
FeCrAl fuel rod cladding	- Chapter 3
Silicon carbide fuel rod cladding	- Chapter 4
Doped uranium dioxide ceramic fuel pellets	- Chapter 5
Uranium silicide ceramic fuel pellets	- Chapter 6

Building upon the nuclear fuel safety criteria and design requirements documented within the second edition of *Nuclear Fuel Safety Criteria Technical Review* [2], this report provides a detailed assessment of each of the above listed ATF technologies against over 50 phenomena important to safety-related fuel design and performance requirements. The evaluation table for each ATF technology appears in Annex 1 available online at: www.oecd-nea.org/7576-annex.

Due to their limited departure from the currently licensed fuel technology, coated zirconium alloy fuel rod cladding and doped UO_2 fuel pellets have the least impact on the applicability of existing fuel design, performance requirements and nuclear safety criteria. Based on the degree of characterisation of irradiation properties and performance, doped UO_2 fuel pellets have already been deployed in reload quantities in at least one country and both of these near-term ATF design concepts are relatively close to licensing and deployment in several other countries. Nevertheless, there are opportunities for collaborative research to fill gaps in the empirical database and further enhance knowledge and understanding.

FeCrAl fuel rod cladding, silicon carbide fuel rod cladding, and uranium silicide fuel pellets introduce more pronounced impacts on existing fuel design and performance requirements and nuclear safety criteria. Larger data gaps exist for these longer-term ATF design concepts, which lack the degree of characterisation available for the short-term ATF design concepts.

A variety of new phenomena were identified which challenged the applicability of existing performance metrics and analytical limits or created the need for new criteria. Chromium-coated zirconium alloy cladding may be susceptible to hydrogen permeability from the coolant through the coating surface into the base material during normal operation and may experience a zirconium-chromium eutectic reaction at the cladding-coating interface and chromium diffusion into the base material under accident conditions. FeCrAl cladding may be susceptible to embrittlement due to chromium-rich particle precipitation during normal operation and unstable oxide formation under accident conditions with high heating rates. Silicon carbide cladding may experience chemical compatibility issues and dissolution during normal operation and generation of methane and carbon monoxide under severe accident conditions. No new phenomena were identified for doped UO_2 fuel, whereas U_3Si_2 fuel exhibited a severe exothermic reaction with water and steam.

Chapter 7 describes opportunities for collaborative international research programmes to fill research needs and data gaps for each ATF technology. Research needs are divided by the type of facilities and capabilities within each facility needed to fill the specific data gap. This cross-cutting format illustrates how any given research facility could support multiple ATF technologies. Priority should be assigned to research supporting the licensing of near-term ATF technologies: chrome-coated zirconium alloy cladding and doped UO_2 fuel pellets. However, recognising the calendar time for long-term irradiation campaigns, radionuclide decay (i.e. cooling), and transportation to hot cell facilities, priority must also be given to research needs for the most commercially viable long-term ATF technologies.

The major findings of this investigation into the applicability of existing fuel design and performance requirements for new ATF designs are summarised in Table ES-1 where the relative impact on existing fuel safety criteria, the number of new phenomena and the relative magnitude of data gaps are defined for each of the five ATF technologies. These relative impacts consider the number and degree of changes in safety criteria, performance and design requirements, and analytical limits. The relative magnitude of data gaps represents the overall effort required to characterise the irradiated material properties and performance considering the extent of the existing empirical database.

Table ES-1: **Summary of ATF design evaluations**

ATF design concept	EGATFL technology readiness level*	Relative impact on existing fuel safety criteria	Number of new phenomena	Relative magnitude of data gaps
Coated zirconium alloy cladding	4	Low	3	Low
FeCrAl cladding	3-4	Medium	2	Medium
Silicon carbide cladding	< 3	High	3	High
Doped UO_2 ceramic pellets	8	Low	0	Low
Uranium silicide ceramic pellets	< 3	High	1	High

* In 2018, the EGATFL report defined a TRL for each ATF design concept from 1 to 9, with 9 defined as *routine commercial-scale operation. Multiple reactors operating.*

Chapter 1. Introduction

1.1. Background

The goal of reactor safety is to minimise the risk of radiation-related damage to the public from the operation of commercial nuclear reactors. Fuel operational or design limits are introduced to avoid fuel failures during normal operation and to mitigate the consequences of accidents in which substantial damage is done to the reactor core.

In most countries, dose rate limits are defined for a possible off-site radiological release following such accidents; fuel safety criteria that relate to fuel damage are specified to ensure that these limits are not exceeded.

Fuel safety criteria, with derivative fuel design requirements, operating limits, and performance requirements, are needed to judge the performance of reactor fuel during normal operation, anticipated operational occurrences (AOOs), and postulated accidents (Class III-IV conditions). Collectively, these fuel design and performance requirements are also used to judge the performance of safety-related systems, structures, and components designed to protect the nuclear power plant and mitigate the consequences of AOOs and postulated accidents, as well as to demonstrate compliance with regulatory requirements. In addition, fuel pellet and cladding material properties, which may evolve with exposure, need to be characterised to accurately predict the fuel's performance during normal operation, AOOs, and postulated accidents. All of these important parameters are the subject of this paper.

In 1996, the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Task Force on Fuel Safety Criteria was given the mandate to review existing fuel safety criteria and to focus on new fuel and core designs, new cladding materials and industry manufacturing processes. The task force was also requested to identify those areas in which additional efforts might be necessary to ensure that the technical bases for fuel safety criteria remain adequate. In 2001, the NEA published the result of this work in a report entitled *Nuclear Fuel Safety Criteria Technical Review* [1].

The NEA Working Group on Fuel Safety (WGFS), a successor to the task force, was subsequently tasked with advancing the understanding of fuel safety issues by assessing the technical basis for current safety criteria and their applicability to current burnup limits and to new fuel designs and materials. A second edition of *Nuclear Fuel Safety Criteria Technical Review* [2] was published in 2012 to document the understanding at that time.

With the development of advanced fuel designs and materials, including accident-tolerant fuel (ATF), the applicability of existing fuel safety criteria, with derivative fuel design requirements, operational limits, and performance requirements is challenged. The objective of this report is to:

- assess the applicability of existing fuel safety criteria, based on the reference technology of UO_2 ceramic fuel pellets encased within a zirconium-based alloy cladding, to these advanced fuel designs;
- identify new phenomena or mechanisms which create the need for new or different performance metrics and design requirements;
- identify data gaps; and
- discuss opportunities for international collaborative research to fill these data gaps.

This report does not address transportation, storage, or beyond design basis accidents (a.k.a. design extension conditions). However, extended operational and design limits (e.g. fuel performance beyond 1 204°C) were considered when targeted for a particular ATF design concept. In addition, it should be highlighted that ATF technologies continue to evolve rapidly. The information shared in this document reflects the knowledge collected up to October 2020.

1.2. Scope of ATF technologies

The NEA Expert Group on Accident-tolerant Fuels for Light Water Reactors (EGATFL) investigated a large number of advanced fuel designs and materials, and then defined a technology readiness level based upon the existing empirical database supporting the characterisation of material properties and performance. The EGATFL published their findings in a report entitled *State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels* [3].

By cross-referencing the EGATFL report with results from an international survey of most likely ATF candidates for deployment, the scope of this report was narrowed down to the following ATF technologies:

- coated zirconium alloy fuel rod cladding;
- FeCrAl fuel rod cladding;
- silicon carbide fuel rod cladding;
- doped uranium dioxide ceramic fuel pellets;
- uranium silicide ceramic fuel pellets.

1.3. Design-specific evaluation tables

Building upon the nuclear fuel safety criteria and design requirements documented within the second edition of *Nuclear Fuel Safety Criteria Technical Review* (2012), this paper provides a detailed assessment of each of the ATF technologies defined above. Over 50 phenomena and mechanisms important to safety-related fuel design and performance requirements were identified and dispositioned for each ATF technology. The evaluation table for each ATF technology appears in Annex 1 (available online: www.oecd-nea.org/7576-annex).

Listed in Column D of the evaluation table, the safety-related fuel design and performance phenomena are organised into the following categories:

- normal operational and design limits;
- normal operational fuel degradation and damage mechanisms;
- transient fuel failure modes;
- accident fuel performance, control rod insertion, and damaged fuel coolability requirements, and inputs to radiological consequence assessments.

Phenomena may be listed (and presented) more than once if they affect more than one of the above categories. For example, fission gas released from the fuel pellet during normal operations impacts rod internal conditions (e.g. rod internal pressure, gap gas conductivity), the margin to a fuel failure mode (e.g. cladding lift-off), and radiological source terms.

Column E of the evaluation table identifies the safety function (e.g. fission product barrier, reactivity control) for each of the fuel design and performance phenomena. For the current generation of UO_2 fuel within zirconium alloy cladding, column F provides the performance metric and/or analytical limit used to judge the fuel’s performance against each of the key phenomena listed in column D.

For each of the fuel design and performance phenomena, the following questions were identified and discussed:

- Applicability of key phenomenon or mechanism (column G):
  - Are the existing phenomena (e.g. pellet-to-cladding mechanical interaction, PCMI) applicable to the ATF design, material properties, or performance? (yes/no)
- Applicability of performance metric (column H):
  - Is the existing performance metric (e.g. cladding failure) applicable to the ATF design, material properties, or performance? (yes/no)
- Impact of ATF concept on analytical limits (column I):
  - Discuss the impact of the ATF design, material properties, or performance on the existing analytical limit (e.g. 1% cladding permanent strain).
- Types of data needs (column J):
  - Describe the type of testing and facilities needed to characterise the ATF material properties or performance (e.g. power ramp testing conducted at a research reactor on irradiated segments of commercial fuel rods to define cladding failure thresholds).
- Sensitivity to fabrication process (column K):
  - Is the material property or performance characterisation sensitive to fabrication variables and processes which may be commercially sensitive or proprietary (e.g. physical vapour deposition (PVD) versus cold spray coating application)? (yes/no)
- Extent of public database / data gaps (column L):
  - Describe the extent of the publicly accessible empirical database available to characterise the irradiated material properties and performance.
  - Identify gaps in the empirical database needing to be filled prior to licensing and commercial deployment (e.g. prompt critical power excursion testing conducted at a research reactor on irradiated segments of commercial fuel rods to define coolability limits).
- Opportunity for collaborative research (column M):
  - Is there an opportunity for collaborative or multilateral research to fill data gaps? (yes/no)

Each ATF design-specific evaluation table is documented in a separate worksheet tab of the Microsoft Excel spreadsheet linked in Annex 1 and the principal findings and conclusions are summarised in the following chapters.

Chapter 2. Coated zirconium alloy fuel rod cladding

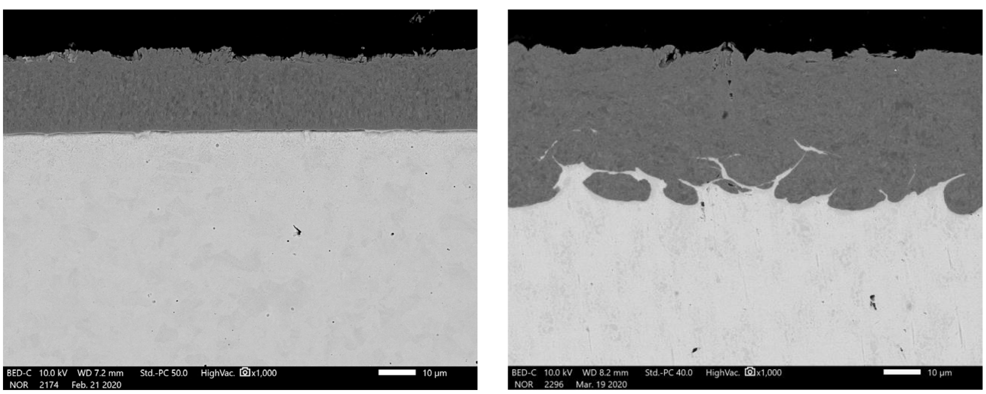
2.1. Background and scope

One solution to further improve the performance of the cladding in normal and accidental conditions is to protect the external surface of the current zirconium alloys through surface treatments such as the deposition of coatings. The concept of improved or coated Zr-alloy claddings has been adopted by many fuel vendors around the world and is considered one of the near-term accident-tolerant fuel (ATF) candidates.

There have been dozens of coating materials proposed and tested around the world, from metallic to fully ceramic coatings, along with a variety of deposition techniques. Cr-based coatings are generally considered the most mature. The typical microstructure of two potential candidates is shown in Figure 2-1.

Figure 2-1: **(a) Typical SEM micrograph of the Cr PVD coated Zr-alloy in the as-deposited condition; (b) Typical SEM micrograph of the Cr CS coated Zr-alloy in the as-deposited condition**

(a) (b)



Source: Reproduced courtesy of ALVEL, a.s. , Czech Republic.

The following coated cladding concepts for light water reactors (LWRs) are being considered by fuel vendors:

- Framatome is developing Cr deposited by physical vapour deposition (PVD) [4], [5].
- Westinghouse Electric Company (WEC) is developing Cr deposited by Cold Spray technique (CS) [6], [7], [31].
- TVEL Fuel Company (TVEL) is developing Cr-based coating deposited by PVD [8], [9], [10].
- KEPCO Nuclear Fuel (KNF) is developing Cr-based metallic CrAl coating deposited by arc ion plating [11], [12].
- Global Nuclear Fuel (GNF) is developing ceramic fretting/oxidation resistant “ARMOR” coating for boiling water reactors (BWRs) with a publicly unknown composition [13], [14].
- China General Nuclear (CGN) is developing Cr deposited by several methods (PVD, CS, laser deposition and plasma spraying) [15], [16].

The Nuclear Energy Agency (NEA) Expert Group on Accident-tolerant Fuels for Light Water Reactors (EGATFL) identified other surface modification concepts such as MAX phase coatings, carbides, nitrides, oxide dispersion strengthened (ODS) treatment or multilayer FeCrAl/Cr coatings [3]. However, as shown above, the fuel vendors are mainly pursuing Cr-based coated claddings that were defined by the EGATFL group on the basis of technology readiness levels (TRL) between three and five on a traditional nine-level scale. Additionally, the EGATFL report summarised attributes of Cr and CrN coated cladding that serve as the starting point when evaluating the impacts on existing nuclear fuel safety and design requirements.

There are many benefits associated with Cr-coated cladding, some of which were confirmed via out-of-pile experiments and others are expected by analytical predictions using available data and models. More data is expected from ongoing lead test rod (LTR) and lead test assembly (LTA) programmes with follow-on post-irradiation examinations (PIE) as indicated by column L in the coated zirconium cladding evaluation table. There have been both positive and negative impacts on coated cladding performance reported in public literature on fatigue, CRUD deposition, heat transfer characteristics and pool-boiling critical heat flux (CHF), as well as ballooning, burst and quenching [24-30]. The inconsistencies of the reported effects seem to depend on the particular deposition parameters and coating design, and some of the available results are not performed in representative conditions.

During normal operations, the main benefits of Cr-coated claddings are expected to be lower waterside corrosion rates, reduced hydrogen uptake, improved wear resistance, and higher thermal creep strength [5, 18, 22, 23, 27, 32]. Under accident conditions, anticipated benefits include enhanced high temperature oxidation resistance resulting in reduced energy release from the exothermic metal-water oxidation reaction, potentially lowering peak cladding temperatures, improved residual post-quench ductility due to less oxygen embrittlement, and reduced combustible gas generation. Another important anticipated benefit is reduced high temperature creep and ballooning and potentially smaller burst opening size [23, 27, 28, 32, 33].

Since there is no public data available on the coated cladding produced by Global Nuclear Fuel (GNF) and China General Nuclear (CGN), this Technical Opinion Paper (TOP) focuses on general Cr-based coatings deposited by PVD, CS or arc ion plating methods on the outer surface of pressurised water reactor (PWR)/water-water energetic reactor (VVER) fuel rods. BWRs are excluded since further testing of Cr-based materials is needed to confirm material compatibility in BWR environments. It is assumed that only outer surfaces of the cladding are coated, while the inside of the cladding and other parts of the fuel assembly (guide tubes, spacer grids, nozzles, etc.) are uncoated. These assumptions cover the majority of the Cr-coated cladding concepts pursued by fuel vendors and research groups around the world.

The TOP also limits the nominal coating thickness to less than 30 microns since most coated cladding designs are within this range. The evaluations performed in this report are based on the hypothesis that the coating does not strongly affect the neutronics of an LWR with respect to fuel safety criteria, and does not negatively affect the mechanical behaviour of the underlying Zr-alloy cladding substrate providing structural function [20, 21]. Thin coatings are also expected to have a limited impact on the thermo-mechanical behaviour of Zr-based cladding, since it is assumed that standard Zr-based substrates do not degrade or change their fundamental properties and microstructure due to the coating deposition (e.g. due to deposition temperature, substrate straining, mechanical interactions such as hardening) or due to the coating presence during in-reactor operation (e.g. material interdiffusion, embrittlement, increased hydrogen uptake) [17-19]. The above assumptions nonetheless will have to be demonstrated by the fuel vendors during the qualification and licensing process.

The design and engineering of the coated cladding should also ensure the integrity and adhesion of the coating when considering potential coating failure mechanisms during fuel operation. For this reason, the performance of a damaged coating (e.g. scratched or cracked) should be also considered in the licensing and design calculations if such a condition is to be expected during normal operation or accidental conditions. Coatings should be adherent during normal operation and off-normal conditions, protecting the substrate from rapid oxidation during high temperature transient conditions. Any impacts of expected or prototypical coating damage on the substrate behaviour, such as the potential for galvanic corrosion between zirconium and chromium, should be addressed and the impacts on safety should be evaluated.

However, it should be noted that even within the scope described above, aspects of the coated cladding performance may be product-specific and any Cr-coated cladding design would have to ensure the basic premises described above in order to fall within the applicability of this evaluation. Since only outer surface coatings on standard Zr alloys are considered, fuel and cladding interactions are comparable to standard Zr and UO_2 fuel. Small variations in fuel performance could potentially occur, such as higher pellet-cladding contact pressure due to a cladding with increased strength, but these variations are not significantly different from those which could occur even with a different Zr-alloy cladding and the performance from the perspective of pellet-cladding interaction is similar to the traditional Zr and UO_2 fuel system.

Throughout the rest of the document, only "Cr-coated cladding" is to be used.

2.2. Impact on design and performance requirements

Due to the thin coating assumed (up to 30 μm), Cr-coated cladding is very similar to traditional Zr-based cladding and therefore all of the phenomena and performance metrics that are PWR-related and cladding-related are applicable for this concept, as shown in columns G and H of the coated zirconium cladding evaluation table. Some of the analytical limits are, however, impacted or the potential impact on the analytical limits is not known to date. Both positive and negative impacts are reported, as shown in column I of the coated zirconium cladding evaluation table. It should be noted that the impact on analytical limits can be product-specific, which is indicated in column K. The most important findings are discussed below.

2.2.1. Normal operations

There are no explicit criteria but Cr-coated cladding should be accounted for in assembly mechanical design evaluations. Changes in hydraulic forces, assembly bow and growth, buckling, or rod to grid fretting or fatigue should be considered since they can be affected due to the presence of coatings.

It has been shown that pool-boiling CHF is affected by surface treatments including coatings with various surface microstructures, but such effects have to be evaluated in PWR CHF conditions at high flow and high pressure since other preliminary results indicate that the effect of coated cladding has no adverse impact on PWR CHF when compared to Zr-alloy cladding [29-30]. Cr coatings have the potential to reduce the high temperature metal-water oxidation reaction. If this can be shown, one may be able to demonstrate acceptable performance for limited times-at-temperature above this critical heat flux, thus leading the way to a potential evolution of the PWR CHF criterion to a time-at-temperature limit instead of the current departure from nucleate boiling margin (DNBR) limit.

Coating oxide spallation and volatilisation could lead to a rapid, but short-lived increase in reactor coolant system activities due to the activation of Cr (e.g. Cr-51 formation). Additionally, it is a potential contributor to refuel floor dose rates during shutdown refuelling activities. This should be evaluated for analyses of radiological consequences with a potential increase in initial source term.

The current analytical limits are applicable to waterside corrosion of Zr-based alloys to preclude other phenomena from occurring. For Zr-alloy cladding, the steady-state cladding oxidation and/or hydrogen limits are established to preclude oxide spallation, which has typically been observed above 75-100 μm and/or to limit the mechanical cladding damage due to the oxidation and hydriding. Zirconium oxide spallation can lead to a local cool spot, which acts as a sink for hydrides, creating notably a local, extremely brittle hydride lens. Coating oxide and coating spallation are not expected to result in localised hydrogen concentrations because their removal would lead to a hot spot due to the formation of ZrO_2, which is a thermal insulator. This nevertheless has to be assessed depending on the Cr-coated cladding design. The impact of the coating on hydrogen uptake must be assessed in light of the thin oxide scale and potential permeability of hydrogen through the coating oxide and coating. The current analytical limits are adequate/conservative, but they should be evaluated in light of changes to the corrosion-related processes.

2.2.2. *AOOs and postulated accidents*

The current loss-of-coolant accident (LOCA) limits are defined in order to ensure the requirements of post-quench ductility or strength-based behaviour, including an additional axial loading during the quench, but are only surrogates for the amount of oxygen in the beta phase of the Zr-alloy cladding following a time-at-temperature exposure in steam. In particular, the equivalent-cladding reacted (ECR) limit is determined by translating the time-at-temperature at which cladding ductility becomes compromised into an ECR as calculated with a correlation based on uncoated cladding oxidation kinetics test data (e.g. Baker-Just correlation or Cathcart-Pawel correlation). It is essentially a surrogate time-at-temperature limit to prevent cladding embrittlement due to oxygen absorption rather than a limit on corrosion itself. The embrittlement of Cr-coated cladding is a more complex process with several transport mechanisms involved [27, 28, 34], so the current ECR limits based on a total weight gain are inappropriate, and an alternative correlation based on time-at-temperature tests and post-quench or strength-based performance demonstration could be evaluated. However, extensive testing has shown that the current ECR_{BJ} (ECR calculated using the Baker-Just correlation) or ECR_{CP} (ECR calculated using the Cathcart-Pawel correlation) and peak cladding temperature limits are highly conservative for the intact Cr-coated cladding with respect to the time-at-temperature required to cause cladding embrittlement.

Finally, since the ECR limit was based on intact cladding tests and the assumption of two-sided oxidation, burst area survival for coated claddings would need to be considered as the relationship between cladding embrittlement and burst area toughness has changed. Moreover, regarding fuel fragmentation, relocation and dispersal (FFRD), especially during LOCAs, it seems to be important to evaluate the coating cladding behaviour in terms of ballooning and burst characteristics.

Limits are set in place to prevent cladding failure due to pellet-to-cladding mechanical interaction (PCMI) that can occur in a reactivity insertion accident (RIA). It is a highly dynamic and integrated phenomenon depending on the pellet, cladding, initial and accident conditions. The Cr coating does not significantly change the response of the cladding and therefore the current phenomena are still relevant. However, the values of current fuel enthalpy or rise enthalpy limits are based on RIA tests that have been performed on irradiated and unirradiated fuel rodlets in various research test reactors (RTRs) using uncoated cladding. The impact of the Cr-based coatings on the relevance and applicability of those tests and results, i.e. the values of the limits, should be assessed. It should be also noted that hydrogen in the cladding due to waterside corrosion discussed in Section 2.2.1 can embrittle the cladding, which can affect other safety limits such as those for anticipated operational occurrence (AOO) cladding strain, LOCA embrittlement, or RIA PCMI cladding failure.

Melt limits have been established as the limiting temperature concern for uncoated Zr-alloy claddings and should be confirmed for coated claddings. In particular, the possibility of eutectic formation (see Section 2.3) should be addressed.

2.3. **New phenomena**

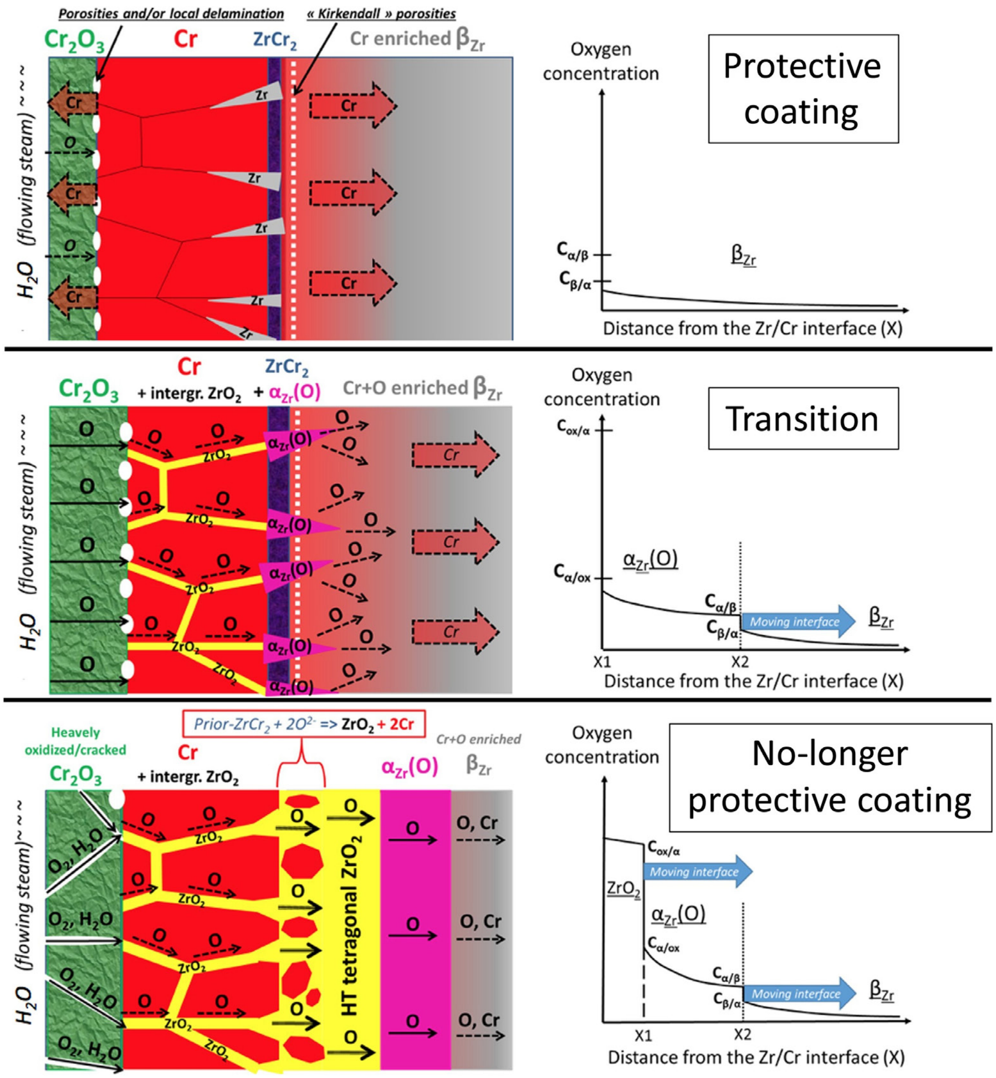
The possibility of eutectic formation should be addressed and the integrity of the Cr-coated cladding post-eutectic should be evaluated. The eutectic formation and its impact on the cladding are influenced by the design and the temperature ramp scenario.

This new phenomenon does not introduce the need for a new regulatory criterion because it is already part of the overarching safety criteria for postulated accidents, such as cladding melt, cladding fracture due to embrittlement, combustible gas generation, etc. Models and acceptance criteria (limits) developed for any cladding product need to demonstrate their applicability through any phenomena that could occur in the associated temperature ranges.

Current design basis accident (DBA) ECR limits have been shown to be conservative with respect to the time-at-temperature, as calculated with the associated correlation (Baker-Just correlation or Cathcart-Pawel correlation), required to cause cladding embrittlement. However, the embrittlement mechanism is more complex for Cr-coated cladding due to diffusion of Cr into the Zr-alloy and the protectiveness of coatings from diffusion of oxygen that may vary with time as shown in Figure 2-2.

This new phenomenon does not introduce the need for a new regulatory basis because it is already part of cladding fracture due to embrittlement for a postulated LOCA accident. One could demonstrate the relevance of current limits and current correlations or elect to change the value of the criteria by developing a product-specific time-at-temperature correlation and performing the tests required to demonstrate the associated retention of acceptable fuel behaviour.

Figure 2-2: **Schematic overview of the HT steam oxidation process of Cr-coated Zr-based alloys**



Source: Reproduced from [28].

Two mechanisms for hydrogen uptake during normal operations exist for uncoated and coated cladding: H permeability from coolant through surface into bulk (large ZrO_2 layer protects uncoated cladding) and H absorption from corrosion reaction (formation of ZrO_2 leads to H absorption); due to a very thin Cr_2O_3 layer the first mechanism's contribution is expected to be dominant while the second's almost non-existent when the coating is protective. This phenomenon does not introduce the need for a new performance metric. Hydrogen uptake models should be specific for each product.

New performance metrics could be introduced to ensure coating integrity if tests show an acceleration of cladding degradation with failed coatings (e.g. non-adherence, scratches, deposition heterogeneities, cracks). One of the main concerns related to coating integrity is the compatibility of Zr-based substrates and coatings due to different physical properties such as coefficient of thermal expansion, creep, swelling, growth, ductility, and galvanic corrosion. This can impact coating integrity during transients and long-term irradiation with an effect that will have to be evaluated. These phenomena do not introduce the need for new regulatory limits but they should be addressed in the models which are developed and validated with product qualification.

2.4. Data gaps

Out-of-pile data on fresh material are generally available but gaps exist when data needs are product-specific. Data on irradiated material are limited; however, it is expected that ongoing LTR and LTA programmes will provide more information. The extent of public databases and data gaps are shown in detail in column L of the coated zirconium cladding evaluation table. The major gaps are described below.

With respect to mechanical tests with irradiated cladding materials, data gaps are related to phenomena such as fatigue, stress/strain, creep, cladding collapse and fuel-clad gap opening, PCMI failure thresholds and fracture due to fuel handling accident (FHA) loads. Available empirical models are based on assumptions and experimental data with fresh cladding material.

With respect to long-term irradiation related effects, data is not available for fuel rod growth and bending, irradiation creep, irradiation accelerated corrosion and hydrogen uptake, intergranular stress corrosion cracking, localised hydrogen concentration, CRUD deposition, assembly geometry degradation, assembly bow, and reactor coolant system activity due to coating spallation/dissolution/volatilisation.

With respect to pellet-to-cladding interaction (PCI)/stress corrosion cracking (SCC), phenomena have not been studied and no ramp test data is available. Cladding rupture due to rapid power excursion under RIAs should be considered for the applicability of the current limits, and violent expulsion of fuel fragments from ruptured cladding should be studied.

With respect to cladding melting and combustible gas generation, testing at temperatures around eutectic reactions should be studied for the specific products. Contradictory test results (e.g. cladding melting of thin layer on the surface vs. acceleration of oxidation and further degradation and melting) illustrate the need for further testing.

Product-specific data gaps include boiling crisis, oxidation, embrittlement, balloon/rupture, cladding fracture due to embrittlement, cladding ballooning (including prototypical peroxidation and hydrogenating), and the effect of surface conditioning on PWR CHF (high flow high pressure).

2.5. Opportunities for collaborative research

The opportunities for collaborative research are indicated in column M of the coated zirconium cladding evaluation table.

As shown in columns K and L, the extent of public databases is limited and there are many data gaps that could be supported by collaborative research. There are, however, commercial limitations from fuel vendors that partially restrict the opportunities for research – especially for phenomena that are product-specific.

Collaboration on more general topics with data gaps is possible and needed. The general unresolved issues related to normal operation include CHF testing, cladding creep, and other long-term irradiation related phenomena or conditions.

Opportunities for collaborative research related to AOOs and accident conditions include investigation of RIA-related and LOCA-related requirements and limits, impacts of the reduced high temperature creep and burst characteristics on FFRD especially during LOCAs, and conditions beyond current regulatory design basis.

Chapter 3. Iron-chromium-aluminium (FeCrAl) fuel rod cladding

3.1. Background and scope

The Nuclear Energy Agency (NEA) Expert Group on Accident-tolerant Fuels for Light Water Reactors (EGATFL) State-of-the-Art report [3] also discusses advanced steels as cladding material for Uranium-based oxide fuel pellets. It focusses on the cladding alloys on FeCrAl base with Kanthal APMT as commercial product, the Oak Ridge National Laboratory (United States) (ORNL) alloys (C06M, C26M, C35M, and C36M), and Japanese oxide dispersion strengthened (ODS) FeCrAl as potential candidate materials for reactor operation [35]. Table 3-1 lists the major alloying elements for the three main FeCrAl concepts.

The main reason for the development of FeCrAl cladding is the high temperature oxidation resistance and the lack of an excessive exothermal metal-oxygen reaction above 1 200°C compared to zirconium alloys (Zircaloy), even though FeCrAl has a lower melting temperature compared to zirconium alloys. Furthermore, the mechanical properties for yield strength, ultimate tensile strength, elastic modulus, are higher and the creep rate is lower for FeCrAl compared to zirconium alloys. This results in a beneficial fuel rod behaviour under design basis accident (DBA) as well as design-extended condition accidents and allows a thinner cladding tube thickness. In particular, the FeCrAl cladding leads to an improved barrier for radionuclides (except tritium), lower risk for coolant channel blockage (less ballooning) and orders of magnitude lower hydrogen production due to less high temperature oxidation as they may occur during transients and accidents.

FeCrAl cladding is suitable for the use in boiling water reactors (BWR) and pressurised water reactors (PWR), and also designed to operate in both types of water chemistries [3]. All considerations must take into account the usage in either of these reactor types. The usage of FeCrAl spacer grids and guide tubes depend of specific vendor concepts. BWR fuel channel boxes are not used in current concepts, though FeCrAl is generally applicable for this purpose.

The FeCrAl cladding evaluation table documents an assessment of the applicability of existing fuel safety and design criteria for FeCrAl cladding material in combination with UO_2 fuel pellets. Data gaps and research needed to characterise this advanced fuel design for future licensing are captured in the evaluation matrix. Examination of the evaluation table reveals that the characteristics and expected performance of FeCrAl will have an impact on in-reactor phenomena and many key safety and design criteria. While the full extent of these impacts is yet unknown, some key observations, data gaps, and research needs are discussed below.

Table 3-1: **Composition of different FeCrAl alloys in per cent weight**

	Cr	**Al**	**Fe**
Kanthal APMT	21	5	Bal.
ORNL FeCrAl	10-12	5-6	Bal.
Japanese ODS FeCrAl	12	6	Bal.

3.2. Impact on design and performance requirements

The impact of FeCrAl cladding material on the major fuel design and performance requirements is described in column I of the FeCrAl cladding evaluation table. The most important findings are discussed below.

3.2.1. Normal operations

Steel-based alloys exhibit a higher neutron capture cross section than zirconium-based claddings which impair the neutron economics of this light water reactor (LWR) fuel concept. Strategies for compensating the neutronic effects consist of a thinner cladding thickness as well as an increased fuel density or fuel enrichment as well as an increased fuel diameter due to the thinner cladding.

Both the thermal and irradiation creep rates of FeCrAl are much smaller than for zirconium alloys, leading to slow creep-down during reactor operation. In particular, oxide dispersion strengthened (ODS) FeCrAl has a much lower thermal creep rate at high temperatures compared to zirconium alloys. This lower creep rate influences the time to gap closure and results in a higher pellet centre temperature. However, this temperature increase was calculated to be ~100 K under the representative normal operation conditions and the resultant increase of fission gas release rate was limited to a few percent [36].

FeCrAl alloys exhibit a different corrosion behaviour compared to zirconium alloys. Where zirconium alloys receive a mass gain due to the oxidation of an oxide layer, FeCrAl shows very limited corrosion with the tendency for a mass loss in reducing atmospheres as used in PWRs [37].

The retention of tritium is a primary concern with the FeCrAl cladding. It originates as a ternary fission product in the fuel rods. The high mobility of tritium leads to a high permeability from fuel pellet via the gap through the cladding. This is of particular interest for BWRs, since the permeated tritium becomes the only source of tritium except for the leaking from fuel rods and control rods/blades. For PWRs, the operational reactivity control through neutron absorption by soluble boron and lithium are additional sources of tritium [38]. The tritium permeability is reduced by high chromium (Cr) contents (e.g. 20%-23% in Kanthal APMT) [38]. Moreover, oxide layers formed on the outer surface in the coolant have been shown to be an efficient barrier against tritium for ODS FeCrAl [39].

Heat treatment of either bare clad material or an additional aluminium (Al) layer on the clad forms aluminium oxide (Al_2O_3) and reduces the permeability of some orders of magnitude [38]. Until now, the application of an oxide layer is not intended for FeCrAl in commercial operation in LWRs. Today, there is no safety criterion which applies directly to the tritium retention. Permeability measurements on irradiated fuel rods are needed to quantify the effect on the resulting tritium dose.

3.2.2. AOOs and postulated accidents

Regarding the thermal properties, FeCrAl material has a higher thermal expansion coefficient (~ two times) and lower melting point (~1 780 K) [3, 40] compared to zirconium alloys. The mechanical properties also differ between these two concepts. Limitation to strain is usually given as engineering circumferential strain. Most FeCrAl cladding concepts include reduced thickness ($s_{FeCrAl} \approx 0.3…0.4$ mm) to about half the thickness of zirconium alloy claddings ($s_{Zry} \approx 0.5…0.75$ mm). Displacement measured by cladding tube outer diameter results in higher true strains for FeCrAl compared to zirconium alloys, whereas the thinner cladding needs to be taken into consideration. Whereas no reactivity-initiated accident (RIA) and power ramp tests of FeCrAl cladding have been performed, an analytical evaluation under power ramp condition was examined [40]. The ODS FeCrAl thinner claddings and lower creep rates result in a higher plastic strain of claddings, a lower pellet-to-cladding mechanical interaction (PCMI) contact pressure and a comparable cladding circumferential stress. It is however apparent that analytical predictions need to be verified by experimental studies such as the power ramp test at a test reactor.

The current criterion regarding cladding embrittlement is related to high temperature oxidation of zirconium alloy cladding systems and is practically controlled with the equivalent-cladding reacted (ECR) criterion. This criterion relies on the measure of the oxidised layer, which can be considered an equivalent to 7% (e.g. United States and Europe) or 15% (Japan) of the sound thickness of the cladding as fabricated. Since oxidation processes differ for FeCrAl compared to zirconium alloy and exhibit new phenomena, the ECR criterion is not applicable there. Besides that, FeCrAl cladding is not affected by embrittlement through hydrogen and oxygen [3].

3.3. New phenomena

FeCrAl is affected by embrittlement due to high temperature oxidation, as zirconium alloys are affected, but to a smaller extent. Besides that, there is a second type of possible embrittlement: FeCrAl cladding is susceptible to Cr-rich α' (alpha prime) precipitation within the cladding, which may lead to embrittlement at a low-temperature range (e.g. normal operation temperature). High Cr content (e.g. in Kanthal APMT) leads to high corrosion resistance and the possibility of α' precipitation at low temperatures (T < 500°C). The formation and mechanical properties of Cr-rich α' formation within the α texture needs to be analysed in mechanical testing on irradiated fuel rods. Extensive tests on varying parameters will make it possible to quantify the risk of Cr-induced embrittlement at temperatures < 500°C during normal operation, transients, DBA or spent fuel storage.

The effect of α' precipitation increases with the Cr content. The alloys suitable for reactor operation therefore have reduced Cr content (see Table 3-1), which is part of the design process. The influence of this phenomenon is still under discussion. New regulatory criteria should be introduced covering this low-temperature embrittlement effect of alloying compounds. However, the current criteria, such as a 1% strain limit, would be also applicable to the low-temperature embrittlement.

The current performance metric focusses on peak cladding oxidation and oxidation thickness limit. The corrosion process of FeCrAl material is a competing reaction of the formation of an oxide layer and dissolution into coolant whereas that of the zirconium alloy is only the formation of an oxide layer. A new performance metric would be needed to involve the formulation of thinning of wall-thickness.

Fast loss-of-coolant accident (LOCA) heat-up ramps showed an unstable oxidation which included a direct oxidation of iron as composition of the FeCrAl material in limited cases, whereas a protective aluminium oxide scale was expected to build up and prevent the direct oxidation of iron [41-42] in most cases.

Even though the FeCrAl materials have high resistivity to the material reaction with neighbouring materials (coolant, UO_2 pellet, stainless steel and neutron absorber [B_4C]) [43], the eutectic reaction with oxidised neighbouring materials may lower the melting temperature of FeCrAl and make it sensitive to material combinations [44].

Those two phenomena should be related to the embrittlement (or even loss of fuel integrity) at high temperatures (e.g. DBA LOCA) below its melting temperature. The effect of the temperature ramp rate on the oxidation behaviour of the different alloy compounds addresses the same regulatory limit peak cladding temperatures criterion or the oxygen-induced embrittlement at high temperatures (e.g. DBA LOCA). The index of ECR would be also applicable the FeCrAl materials but would be a different limit value. Hence a new limit should be introduced, (e.g. the heat-up rate) preventing cladding from losing integrity due to embrittlement (or melting), limit the amount of combustible gas due to oxidation and exclude fuel rod, control rod or guide tube melting.

3.4. Data gaps

The publicly available database about FeCrAl cladding behaviour under reactor operation conditions is limited. Several degradation mechanisms and their influence on material properties (e.g. ultimate tensile strength, creep rate, alloy component diffusion, oxygen and hydrogen diffusion) should be studied on irradiated material to develop material models for this cladding concept. Further, the database on fatigue, fretting wear and stress corrosion cracking needs to be expanded from results of lead test assembly (LTA) used in commercial nuclear power plants. The lead test rod (LTR)/LTA irradiation campaigns with FeCrAl materials have already started in US commercial nuclear power plants and the data described above would be obtained in the near future [45].

The interaction of fuel with the cladding has been examined in irradiation tests of fuel pins in the Halden boiling water reactor (HBWR) [46] and advanced test reactors (ATR) [47] but the range of those experiments is still limited. Possible interdiffusion of fuel and cladding components should be investigated in post-irradiation examination (PIE) after irradiation in test reactors and commercial nuclear power plants.

Some integral LOCA tests using unirradiated claddings have been performed [39, 48]. Compared to zirconium alloy tubes, Oak Ridge National Laboratory (ORNL) FeCrAl claddings demonstrated a similar burst temperature at a similar internal pressure but a higher resistance to embrittlement owing to their reduced high temperature oxidation. In the case of Japanese ODS FeCrAl claddings, in addition to the higher resistance to embrittlement, a higher burst temperature at higher internal pressure was confirmed [39]. A separate effect test using ring-type specimens also demonstrated the higher resistance to embrittlement compared to zirconium alloys, and no embrittlement by steam oxidation at 1 473 K for 24 hours followed by water quenching [40]. More detailed LOCA studies using irradiated fuel pins are however required to assess fuel cladding interaction and its impact on fuel fragmentation, relocation, and dispersal (FFRD). Further, the phenomenon of clad ballooning during LOCA scenarios needs to be studied and compared to zirconium alloy cladding to assess the risk of coolant channel blockage and fuel dispersal. The sensitivity of iron oxidation regarding the LOCA – especially the temperature ramp – conditions need further experimental data [42]. Integral tests with zirconia pellet simulators showed a possible fuel cladding interaction at high temperatures, which should be investigated [44]. Moreover, the correlations for hydrogen production under LOCA conditions should be studied in integrated tests without the influence of eutectic oxidation [44].

The behaviour of FeCrAl in rapid transients and power excursions leading to PCMI failure modes should be carried out to analyse new failure modes and determine enthalpy limits depending on burnup and fuel type. Also, the shape and size of the cladding burst openings after PCMI failure should be studied to assess the risk of fuel expulsion into the coolant.

Considering the moderate technology readiness level evaluated by the EGATFL, the FeCrAl cladding development programme still needs to provide necessary experiments and validation data to quantify the impact on the existing nuclear fuel safety and design requirements.

3.5. Opportunities for collaborative research

For FeCrAl, with its moderate technology readiness levels (TRL), there are several data gaps and manifold ways to fill these gaps. A general lack of information covers the field of irradiation and its influence on the mechanical behaviour, which is a similar situation to the data gaps of a new Zr-based alloy. The discussed points below can be classified to the specific fuel safety criteria in the FeCrAl cladding evaluation table.

Regarding the mechanical behaviour of irradiated and non-irradiated material, there are open questions that could be answered through collaborative research, namely about fatigue, stress/strain behaviour, fretting wear, waterside corrosion, inter-granular stress corrosion cracking and cladding creep. Especially for waterside corrosion, a new, reliable method for poolside measurements should be developed and tested. Regarding cladding creep, the in-reactor creep behaviour as well as the thermal creep behaviour of irradiated and non-irradiated FeCrAl should be analysed (e.g. hardening relaxation tests, in-reactor creep tests).

The α' embrittlement of the cladding is a new phenomenon which should be analysed in a testing campaign covering variations in alloy compounds, temperature, stress, corrosion state, and irradiation. Further, mechanical tests need to be conducted to derive a limit of cladding ductility in the context of α' precipitation.

The corrosion effects at high temperature should be studied in detail. For the excessive oxidation reliable oxidations, correlations need to be developed and tested in integral fuel rod (bundle) tests. Further, the combustible gas generation from oxidation needs to be investigated regarding the influence of temperature ramp rates. Open questions remain regarding the effect of melting of neighbouring components (e.g. control rod and channel box) on fuel rod integrity, which should be evaluated separately and verified in integral tests. Studies on ballooning and FFRD will complete the open questions regarding the FeCrAl behaviour during LOCA.

Large gaps in understanding remain regarding the PCMI and pellet-clad interaction (PCI)/stress corrosion cracking (SCC) damage in power ramps and RIA. Only in-pile reactor tests will be suitable to answer open questions, even though out-of-pile simulation tests can be a good reference for the in-pile tests. Included are the fuel failure enthalpies of a FeCrAl rod, the size of burst openings to assess the risk of fuel expulsion after RIA, and the fuel cladding interaction with and without an inner oxide layer of the cladding.

Chapter 4. **Silicon carbide fuel rod cladding**

4.1. **Background and scope**

SiC ceramic matrix composite cladding is a candidate accident-tolerant fuel (ATF) cladding material considered for light water reactors (LWRs) due to its desirable high temperature mechanical properties, high oxidation resistance, and small neutron absorption cross section. Test results showed that SiC performs well in steam at temperatures up to 2 000°C [49]. Modular accident analysis program (MAAP) calculations also showed that SiC cladding may avert events such as the TMI-2 [50]. Figure 4-1 shows some SiC ceramic matrix composite cladding tubes under development.

Figure 4-1: **Cladding tubes made of a composite of SiC fibres and SiC matrix**

Source: Reproduced from [51].

In addition to the oxidation resistance at high temperatures, SiC cladding may also offer benefits for design basis accidents and normal operation. Due to its high temperature strength and no-creep behaviour, SiC cladding may maintain fuel rod integrity under loss-of-coolant accident (LOCA) conditions without radioactive materials release. Radiation damage in SiC typically saturates at around 1 dpa, which is about 6 months of irradiation in a commercial reactor, and the rod growth stops at a maximum value of about 0.7% [52]. Thus, SiC cladding can maintain its dimensional stability and mechanical strength after ~1 dpa. This enables ultra-high burnup capabilities if high-density fuel or higher enriched fuel are used for longer fuel cycles. SiC cladding may also offer benefits in terms of flexible power operation since the SiC is expected to be insensitive to corrosive fission products causing pellet-clad interaction (PCI)/ stress corrosion cracking (SCC). In addition, because of its oxidation resistance, mechanical strength, and no-creep behaviour at high temperature, SiC cladding may withstand departure from nucleate boiling (DNB) conditions for a period of time without rod failure, which may lead to DNB margin improvement.

Because SiC is a novel cladding material and its behaviour is completely different from metal cladding materials such as zirconium alloys, it is expected that the applicability of some of the existing fuel safety requirements and criteria would be different and new criteria may need to be established for SiC. In this chapter, the applicability of the current criteria for SiC cladding is evaluated and reported. The objective concept of SiC cladding in scope is a composite of SiC fibres and SiC matrix (SiC_f/SiC). The design and dimensions of cladding covered in this chapter are assumed to be compatible with zirconium alloy claddings used in the current pressurised water reactor (PWR) and boiling water reactor (BWR) fuel assembly products. Manufacturing processes of SiC/SiC are assumed to be chemical vapour infiltration (CVI), polymer infiltration and pyrolysis (PIP), melt infiltration (MI), nano-infiltration, and transient eutectic-phase (NITE).

4.2. Impact on design and performance requirements

The impact of SiC cladding material on the major fuel design and performance requirements is described in column I of the SiC cladding evaluation table. The most important findings are discussed below.

4.2.1. *Normal operations*

Operational limits under normal operating conditions for fuel with zirconium alloy claddings (namely reactivity coefficients, criticality/shutdown margins and reactor coolant system activity limits) will not be directly impacted by SiC_f/SiC cladding. However, differences in thermal conductivities are expected to result in temperature effects. Cladding-based thermal design limits established to preclude boiling crisis may be less important given the high temperature performance of SiC based cladding designs. Nevertheless, as SiC presents very different behaviours from zirconium alloys, its use may entail changes to existing safety criteria and requirements. As presented in literature [3], the main identified challenges are (1) chemical compatibility of SiC with the coolant at about 300°C, (2) low pseudo-ductility, and (3) relatively poor thermal conductivity under neutron irradiation.

Under normal operating conditions, zirconium alloys and SiC corrosion mechanisms differ highly. Whereas the corrosion behaviour of zirconium alloys will tend to form a protective oxide layer, the hydrothermal corrosion behaviour, caused by chemical interactions of SiC with the coolant, brings a significant recession of the contact surface. Nevertheless, as the corrosion mechanisms of SiC and zirconium alloys are not similar, a new criterion should be defined. In addition, hydrothermal corrosion behaviour of the interface material between SiC fibres and matrix has to be considered.

The absence of hydrogen uptake eliminates the failure mode by hydrogen-induced embrittlement of the composite structure. As there is no formation of an oxide layer, steady-state limits on oxide thickness (e.g. 100 microns) to ensure stability of the oxide layer (i.e. avoid spallation) are no longer necessary. Moreover, contrary to zirconium alloys, SiC is inert and insensitive to corrosive fission products that normally cause SCC in zirconium alloys. Fuel manoeuvring guidelines to reduce the risk of cladding failure due to PCI/SCC may no longer be necessary.

As SiC_f/SiC composites' mechanical behaviour is very different from zirconium alloy's, the stress/strain and fatigue mechanisms should be accounted for with a new approach, comprising a re-evaluation of the existing requirements. On the one hand, radiation damage saturates after ~6 months of irradiation, implying that the rod growth stops at a maximum value of about 0.7%. On the other hand, an uneven distribution of neutron flux could possibly cause bowing, which could be mitigated by the spacer grids. In both cases, results of post-irradiation examinations (PIE) should be accounted for in safety analyses to identify and factor in their level of influence on the overall mechanical strength and behaviour of the cladding. Then, as SiC is a much harder material than zirconium alloys, its resistance to rod failure by fretting mechanisms is highly increased. Therefore, SiC will have a positive influence on fuel rod robustness against fretting. Finally, pellet-cladding mechanical interaction (PCMI) has to be assessed, as the low pseudo-ductility of SiC_f/SiC composites may be problematic. In case of PCMI, as it presents a limited elastic domain, the cladding is less able to accommodate the stresses applied by the pellet. This influence should be determined with calculations and mechanical testing (e.g. inner pressure tests).

Although the thermal conductivity of SiC is higher than that of zirconium alloys at room temperature, it decreases significantly with irradiation and increasing temperature [53]. Therefore, the decreased thermal conductivity under neutron irradiation under normal LWR operating conditions could potentially result in significant thermal stresses in the cladding, leading to early formation of micro-cracks and potential bowing.

In addition to micro-cracks, initial pores that form during the manufacturing process exist in SiC_f/SiC composites. The micro-cracks and pores degrade the leak-tightness of SiC_f/SiC cladding. Therefore, leak-tightness should be evaluated and a metallic layer or coating on the composites is needed in case the leak-tightness is insufficient.

The differential swelling generated by the high temperature side of a SiC_f/SiC fuel cladding may cause high tensile stress rates. The influence of this phenomenon, which could be increased by irradiation-induced degradation of the thermal conductivity, must be accounted for in safety analyses by collecting irradiation data. In addition, the potentially low thermal conductivity of SiC_f/SiC composites leads to elevated centreline temperatures of the fuel. On one hand, it may reduce the cladding's coolability as fuel could reach its melting temperature faster, while on the other hand it may result in an increase in the fission gas release that will affect the inner gas pressure. In both cases, the phenomena should be taken into account in safety analysis by collecting irradiation data.

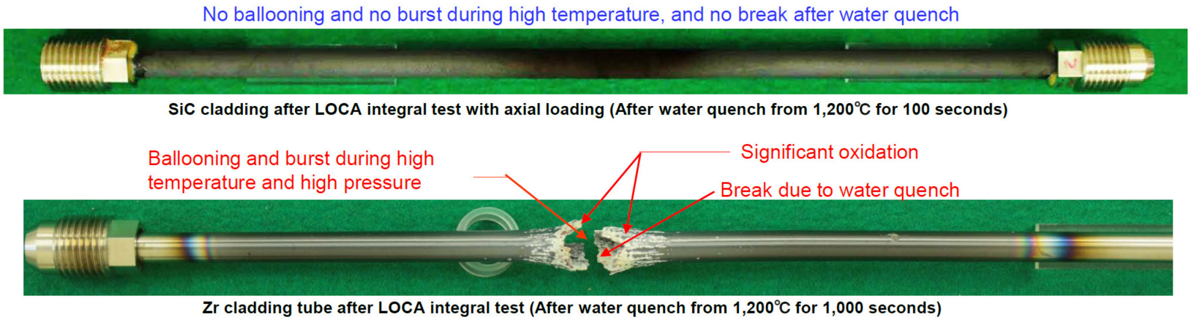
*4.2.2. **AOOs and postulated accidents***

The reactivity-initiated accident (RIA) acceptance criteria are prescribed to prevent fuel rods from failure under anticipated operational occurrence (AOO) conditions and to preserve the integrity of the core and reactor core pressure boundary from mechanical energy generated from the reaction between melted or dispersed fuel and coolant water, etc. in the postulated accidents. The current RIA criteria which are specified in terms of the fuel enthalpy, cal/g or J/g, are based on test results obtained on zirconium alloy claddings under unirradiated and irradiated conditions in test reactors [54]. Therefore, the current RIA criteria are not applicable to SiC_f/SiC cladding. Several failure modes – melting, high temperature oxidation, rupture with ballooning, and PCMI failure – have been observed during tests on fuel rods with zirconium alloy cladding [55]. SiC_f/SiC cladding is expected to have good resistance against melting, high temperature oxidation, and rupture at elevated temperatures. However, the resistance of SiC_f/SiC cladding to PCMI failure has to be studied because of SiC_f/SiC susceptibility to brittle failure.

Emergency core cooling system (ECCS), the function of which is to preserve core coolability, must be capable of providing adequate core cooling before cladding reaches the LOCA criteria of peak cladding temperature 2 200°F (~1 200°C) and the cladding oxidation limit 17% equivalent-cladding reacted (ECR) in the United States [56]. The LOCA criteria are prescribed to prevent zirconium alloy cladding from oxidation-induced embrittlement leading to loss of coolable geometry. The high temperature steam oxidation behaviour of SiC_f/SiC is significantly different from that of zirconium alloys. Therefore, the current LOCA criteria are not applicable to SiC_f/SiC cladding and need to be adapted.

SiC is expected to withstand high temperature oxidation under LOCA conditions significantly more than zirconium alloys [57]. This is illustrated in Figure 4-2, which shows a SiC_f/SiC cladding segment in good condition following a high temperature steam exposure that would have significantly degraded a zirconium alloy cladding. Accordingly, an extension of the safety margin is expected in LOCA conditions. In addition, oxidation-related safety criteria could be potentially removed completely for SiC_f/SiC cladding. However, high temperature oxidation and the balance of phenomena associated with SiC_f/SiC claddings that could be damaging to the coolable geometry may need to be reinvestigated to define a new LOCA criteria.

Figure 4-2: **A SiC_f/SiC cladding and a zirconium alloy cladding after LOCA test at 1 200°C for 100 seconds followed by water quench**



Source: Reproduced from [51].

Recently, the dispersal of fragmented fuel from a rupture opening of zirconium alloy cladding was observed in LOCA simulation tests [58]. SiC_f/SiC cladding would not balloon because it is not susceptible to creep or strength degradation at high temperatures [57]. Accordingly, the size of a rupture opening is expected to be very small even if the SiC_f/SiC cladding ruptures due to excess internal pressure, etc. Consequently, SiC_f/SiC cladding has potential to resolve the issue of the dispersal of fragmented fuel during LOCA events. However, it is necessary to investigate possible mechanisms resulting in rupture and rupture/fracture morphologies for the confirmation of no fuel dispersal at LOCA events.

Hydrogen generation resulting from oxidation in the presence of steam under LOCA conditions is also expected to be lowered considerably by the use of SiC_f/SiC cladding [59]. However, the potential generation of methane (CH_4) and carbon monoxide (CO), which are not considered in the current safety assessment methods, have to be assessed.

In postulated accidents, the ability to insert control rods must be secured to control criticality. The key components for the control rod insertion are thimble tubes and spacer grids for PWR fuel, and channel boxes for BWR fuel. If the same materials used in the current fuel products are also used, the current criteria and existing test data related to control rod insertion are available for fuel assemblies with SiC_f/SiC claddings.

For radiological consequences, the amount of fission product (FP) gas and volatile FPs released in a fuel rod during normal operation would increase in SiC_f/SiC clad fuel since the release of FP gas and volatile FPs depends on the fuel temperature. The lower thermal conductivity of irradiated SiC can cause temperature elevation [60] and increase fission gas release during normal operation. The amount of FP gas and volatile FPs in the pellet-cladding gap and the release behaviour need to be addressed.

During events such as fuel handling accidents, fuel transport accidents, seismic events, etc., external mechanical loads are applied to the cladding. Irradiated zirconium alloy claddings, with remaining ductility, could absorb these external mechanical loads. However, SiC_f/SiC claddings, which are highly brittle compared to zirconium alloy claddings, may lose their geometry due to similar external mechanical loads. In addition, the fracture morphology of SiC_f/SiC claddings is expected to be different from that of zirconium alloy claddings. The amount of radioactive material released during such an event depends on the fracture morphology. Therefore, mechanical testing of irradiated SiC_f/SiC claddings needs to be performed under fuel transport and handling accident conditions. Since SiC_f/SiC cladding is lighter in weight than zirconium alloy cladding, external loads applied to fuel assembly components by fuel assembly's own weight at the accidents are expected to be reduced.

For the long-term decay heat removal at post-LOCA, chemical stability during the sump water cooling is required to maintain cladding integrity. SiC, which is a ceramic, is expected to be more stable in water than zirconium alloys. However, a potential corrosion issue of SiC_f/SiC under PWR coolant conditions has been observed [61]. The chemical stability of a ceramic such as SiC might change with neutron irradiation, more than that of metals. In addition, the corrosion behaviour of the interface material between the SiC fibres and SiC matrix is expected to be different from that of SiC. The interface material in the SiC_f/SiC cladding damaged by an accident might be directly exposed to the coolant. To ensure the preservation of coolable geometry, corrosion of the interface material also has to be assessed.

4.3. New phenomena

Under normal operation and AOOs, the main questions that should be raised when using SiC_f/SiC cladding fuel rods concern its environment. Answers to these questions depend highly on the core and assembly designs, and the operating conditions, including the primary system chemistry, especially when considering their use in a full SiC core reload or in a mixed core. For instance, the mechanical and thermal behaviours of an assembly could be different, based on the core composition, and therefore should be assessed for both cases. In addition, the management of water chemistry would need to be evaluated to make sure it takes into account the dissolution of silicon in water.

The preservation of coolable geometry is the key safety function of cladding during postulated accidents where some amount of fission products is released into the primary coolant. Currently, the embrittlement of the zirconium alloys due to oxidation and hydriding is considered to be the critical phenomenon leading to loss of coolable geometry. For SiC_f/SiC cladding, oxidation and hydriding could no longer be applicable metrics for the preservation of coolable geometry. Therefore, critical phenomena leading to the loss of coolable geometry of SiC_f/SiC cladding must be assessed and verified through tests. Based on the results, new performance metrics and regulatory criteria to preserve coolable geometry need to be developed for SiC_f/SiC cladding.

SiC is more stable mechanically and chemically at high temperatures than zirconium alloys. Accordingly, some extension of the currently available grace period, until the onset of severe accidents, is expected. However, a design base accident condition can progress to a severe accident condition even in a reactor using SiC_f/SiC claddings if accident management fails. Therefore, interactions between SiC and UO_2, steam, and the fuel component behaviour at very high temperatures have to be studied. The generation of methane (CH_4) and carbon monoxide (CO) could be an issue that needs to be addressed. The results of the studies are utilised to formulate action plans in accident management.

4.4. Data gaps

Information on fuel, cladding, and structure under normal operating conditions should be determined by collecting out-of-pile test data, which may be consolidated with PIE data. Thus, representative irradiation tests should be performed to study the influence of the LWR environment on SiC_f/SiC performance, followed by not only specific hot cell tests including mechanical behaviour, but also ramp tests. Iterations between experimental tests and modelling studies will be necessary to confirm the identified fuel failure conditions and modes. Indeed, by using feedback from irradiation tests, calculations should be performed to identify limits and criteria to avoid the loss of cladding integrity and fuel structure, and also to make sure that the model predictions are accurate enough to be used for licensing analysis.

Regarding postulated accidents, tests simulating LOCA and RIA and separate effects tests focusing on the critical phenomena that could be damaging to the coolable geometry, are necessary to develop safety criteria for SiC_f/SiC clad fuel. Mechanical tests addressing fuel handling and transport accident conditions are also necessary for safety evaluations that consider the fracture morphologies of SiC_f/SiC claddings.

The risk of severe accidents would be decreased by using SiC_f/SiC cladding. However, the behaviour of SiC_f/SiC cladding under severe accident conditions would be completely different from those of zirconium alloy claddings. Data on the behaviour of SiC_f/SiC cladding at very high temperatures are necessary for safety evaluations.

4.5. Opportunities for collaborative research

The development of SiC_f/SiC cladding is in its initial stages. Some tests using small samples of SiC_f/SiC are being performed by non-nuclear organisations as well as nuclear organisations, and the results are released in publications such as those provided in the references of this report. The next stage is to perform tests using prototypes of SiC_f/SiC claddings under conditions simulating actual operational and accident environments.

The performance of SiC_f/SiC cladding under normal operations, AOOs and even postulated accidents depends more or less on its composite design and fuel rod design. Their designs could be a good topic for collaborative research between the vendors and utilities, but not across a larger group comprising vendors, utilities, institutes, and regulators. On the other hand, identifying failure modes and the critical factors that result in damage to the geometry of the SiC_f/SiC cladding is a common goal for any design and would therefore benefit from international collaborative research.

Chapter 5. Doped uranium dioxide ceramic fuel pellets

5.1. Background and scope

Among all the variants of oxide additives in UO_2, the NEA Expert Group on Accident-tolerant Fuels for Light Water Reactors (EGATFL) report focused solely on two evolutionary concept types that can be deployed in a short time frame, i.e. in the coming years:

- doped UO_2 pellets, which aim to increase grain size and enhance the viscoplastic behaviour and fission product (FP) retention:
  - Cr_2O_3 doped UO_2 pellet;
  - Al_2O_3-Cr_2O_3 doped UO_2 pellet (ADOPT™);
- microcell UO_2 pellet, which aims to enhance FP retention capability:
  - Si-based oxide-doped UO_2 pellet (ceramic microcell UO_2).

This report will limit its analysis to the above-mentioned doped UO_2 fuel pellets, i.e. Cr_2O_3 doped UO_2 pellet and Al_2O_3-Cr_2O_3 doped UO_2 pellet. Considering that both doped fuel types are already in commercial use, the EGATFL report [3] ranks this fuel pellet design concept at a high Technological Readiness Level of 8.

Throughout the rest of the document, the term "doped UO_2 fuel" will be used.

The objective of the doped UO_2 fuels is to increase fuel robustness and efficiency thanks to their enhanced performance, while ensuring greater safety margins. In comparison to standard UO_2, doped UO_2 fuels make it possible to reach higher density and to have a higher fission gas (FG) retention capability, a behaviour widely agreed based on the more developed gas precipitation in the fuel [62]. They also exhibit better pellet-clad interaction (PCI) and improved secondary degradation behaviour. Therefore, doped UO_2 fuels are anticipated to increase safety margins in accidental conditions:

- A better intragranular gas retention capability will make it possible to decrease the rod's internal pressure prior to the accident. This is beneficial in reducing the clad ballooning and the (burst) failure risk.
- A reduced amount of gas available for immediate release at the grain boundaries of the doped fuel pellets is likely favourable to limit the fuel fragmentation and dispersal in case of rod burst, since fuel fragmentation is likely generated by over pressurising inter-granular gas bubbles.

Two commercial processing routes are used. In the first one, the UO_2 is doped with chromium oxide only and the maximum dopant concentration is approximately 2 200 ppm Cr_2O_3 (i.e. µgCr_2O_3/gUO_2) [63]. In another route, aluminium (up to 200 ppm) is used to enhance the grain enlargement function of the chromium oxide (up to 1 000 ppm) [65]. Doped UO_2 fuel is manufactured with the same dimensions and enrichment as conventional UO_2. Light water reactor (LWR) cores fuelled with doped UO_2 fuel have basically the same neutron spectrum as those with UO_2 fuel, due to the amount of additives kept at a minimum in these designs [63, 64]. In addition, the targeted slightly higher densities obtained with doped UO_2 fuels will have limited impact on the neutron spectrum compared to UO_2 fuel. On the other hand, if higher densities are targeted with the doped UO_2 fuels, this remains in the same order of magnitude as for UO_2 fuel. Inconsequential effects on reactivity coefficients or criticality and shutdown margins are anticipated.

5.2. Impact on design and performance requirements

The impact of doped UO_2 fuels on the major fuel design and performance requirements is described in column I of the doped UO_2 evaluation table. Examination of the evaluation table reveals that the characteristics and expected performance of doped UO_2 fuel pellets have no detrimental impact and a likely favourable impact on in-reactor phenomena and many key safety and design criteria. The most important findings are discussed below.

5.2.1. *Normal operations*

With regard to thermo-physical properties, the low amounts of fuel additives (Chromia with or without Alumina) do not cause appreciable effects in melting temperature, heat capacity and thermal expansion behaviour in comparison with standard UO_2 fuel, irrespective of the dopant concentration for the range considered here [65, 67, 70]. It should be noted that doped UO_2 fuel exhibits a slightly lower thermal conductivity than UO_2; however, the measured values for fresh and irradiated fuel have no impact on the analytical limit.

The higher FG retention allows for a lower rod internal gap pressure, which helps maintain the cladding's integrity. The retention in doped UO_2 depends on the doping element, the dopant concentration and the temperature range. The effect of additives on FG behaviour is complex, and the FG retention is usually seen as the result of two competitive phenomena [65]. First, the larger grains in doped UO_2 increase the diffusion path for gaseous fission products to the grain boundaries and are therefore beneficial for gas retention in the fuel. Second, the increased FG diffusivity caused by the dopants may offset the former beneficial effect, especially at high temperature. While the doped fuels exhibit a higher FG retention capability, the reduction of the FG release is not directly proportional to the grain size enlargement, notably in baseload conditions and as commonly expected from the Booth sphere model. Therefore, the overall gain of fission gas release (FGR) provided by the dopants is difficult to quantify, as shown both in experiments [69] and computations [66, 68]. Due to their larger grain size the doped UO_2 fuels also undergo a lower densification and therefore an earlier clad-pellet contact time in operation.

5.2.2. *AOOs and postulated accidents*

Doped UO_2 fuel is anticipated to have an overall favourable impact on the cladding failure mechanisms through slightly increased FG retention and reduced mechanical stresses. In addition, the enhanced thermal creep of doped UO_2 fuel also reduces the pellet-cladding mechanical interaction (PCMI)-induced mechanical stresses on the cladding [69, 70]. Since creep deformation cannot occur within very short periods, experiments are ongoing to address the behaviour of doped UO_2 fuel in reactivity-initiated accident (RIA) conditions [71, 72].

FGR is generally lower for doped UO_2 [63]. The large grain fuel also exhibits lower oxidation and washout rates, which is relevant in case the fuel cladding fails [67].

5.3. New phenomena

No new phenomenon has been identified for doped UO_2 fuels.

5.4. Data gaps

Overall, comprehensive databases are already available enabling thorough investigations of the material properties and in-pile behaviour of the doped UO_2 fuel. The databases include in-pile and out-of-pile analytical experiments, both separate effects and integral, and extensive global demonstration programmes performed in commercial boiling water reactors (BWRs) and pressurised water reactors (PWRs). The maximum rod burnup achieved in commercial reactors depending on the doped UO_2 fuel product is about 75 MWd/kgU. Current material characterisation and irradiations have been conducted on lab-scale specimens and more importantly on production-scale fuel pellets.

However, depending on the fuel variant, the evaluation tables identified some additional data needed for the completeness of the assessment. Recommended activities include acquiring irradiation behaviour at very high burnup levels, e.g. fuel thermal conductivity by direct measurement on irradiated pellets or through online measurement of fuel centreline temperature. Additionally, testing under various power ramp and anticipated operational occurrence (AOO) transients should be performed to refine technological thresholds resulting from pellet-cladding interaction.

Fuel rod transient irradiation testing under loss-of-coolant accident (LOCA) and rapid power excursion conditions should be conducted to establish analytical thresholds for fuel pellet fragmentation and transient FGR. Also, under these postulated accident conditions evaluation of the source term and radiological consequences need to be assessed up to fuel melting.

5.5. Opportunities for collaborative research

Fuel fragmentation, relocation and dispersal (FFRD) is a mechanism accounting for over pressurisation of gas into bubbles and increasing burnup. This mechanism is being actively implemented in fuel performance codes [74, 75] since it limits the final burnup of the fuel.

The 2016 NEA Report on Fuel Fragmentation, Relocation and Dispersal [73] concluded:

> Experiments are still required to complete the phenomenological picture given some divergence on the temperature/burnup criteria for FFRD and the evaluation of the coolability of dispersed fuel.

Since neither product has extensive amounts of data in this area for doped UO_2, common efforts could be spent on experimental and modelling activities.

Several testing facilities are currently available to analyse the behaviour of irradiated fuels in LOCA conditions. Two types of tests can be distinguished: heating tests and integral tests. Heating tests are separate effects tests usually performed on a small open segment of fuel corresponding to the length of one or two pellets, with its cladding. They are well suited to identify and study the main physical parameters that may have an influence on the fuel fragmentation mechanisms (temperature, hydrostatic pressure, heating rate, local burnup, power history, microstructure, etc.). They are particularly convenient when trying to deconvolute individual phenomena.

Integral tests are performed on several centimetres’ length of rodlets, which are pressurised to lead to clad ballooning and burst. Unlike heating tests, they make it possible to study the influence of the cladding strain, cladding burst, rod inner pressure, and segment depressurisation on FFRD. Their results could also provide valuable data to validate the models developed from the results of heating tests. Due to the strong coupling between each phenomenon, modelling and simulation are needed to interpret and transpose the test results to in-reactor LOCA transients.

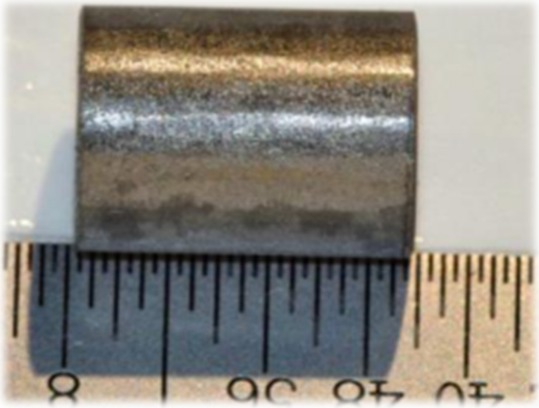
Both heating tests and integral tests should be associated with fine characterisations of fuels and advanced simulations of the fission gases behaviour to characterise precisely the fuel microstructure, particularly the location of fission gases. The tests must be also adequately instrumented in order to simulate and interpret them accurately.

Chapter 6. **Uranium silicide ceramic fuel pellets**

6.1. **Background and scope**

Of the series of compounds in the U-Si phase diagram, the Nuclear Energy Agency (NEA) Expert Group on Accident-tolerant Fuels for Light Water Reactors (EGATFL) report focused solely on U_3Si_2 fuel pellets following rejection of the higher density silicides based on unacceptable swelling and/or low melting point. Likewise, this report will limit its investigation to U_3Si_2 fuel pellets. Note that many of the properties and phenomena discussed below for U_3Si_2 also apply to other high-density, high thermal conductivity fuels such as $U^{15}N$ and $U^{11}B_2$. These other fuels have even higher thermal conductivities and U-235 density. $U^{15}N$ requires use of >95% N15 and $U^{11}B_2$ would require >99.99% B11 in order to avoid neutronic penalties.

Figure 6-1: **As-sintered uranium silicide pellet**



Source: Reproduced courtesy of INL, United States.

The key beneficial characteristics of U_3Si_2 relative to UO_2 fuel pellets are higher thermal conductivity and increased uranium density. The improved thermal conductivity (along with reduced heat capacity) reduces fuel operating temperatures and stored energy. The improved thermal conductivity also decreases the thermal stresses in the fuel and may have positive impacts with respect to fission gas release. In addition, the lower amounts of stored energy in the fuel could reduce the number of fuel rods that will balloon and burst due in loss-of-coolant accident (LOCA) transients. The increased uranium density has a positive impact on fuel cycle economics without going over the 5% enrichment limit of many fuel enrichment and fabrication facilities, as well as commercial nuclear fuel storage pools.

The U_3Si_2 evaluation table documents an assessment of the applicability of existing fuel safety and design criteria, based on UO_2 ceramic fuel pellets encased within a zirconium-based alloy cladding, to U_3Si_2 fuel pellets. Data gaps and research needed to characterise this advanced fuel design for future licensing are captured in the evaluation matrix. Examination of the U_3Si_2 evaluation table reveals that the characteristics and expected performance of U_3Si_2 fuel pellets will have an impact on in-reactor phenomena and many key safety and design criteria. While the full extent of these impacts is yet unknown, some key observations, data gaps, and research needs are discussed below.

6.2. Impact on design and performance requirements

The impact of U_3Si_2 fuels on the major fuel design and performance requirements is described in column I of the U_3Si_2 evaluation table. The most important findings are discussed below.

6.2.1. *Normal operations*

Light water reactor (LWR) cores fuelled with U_3Si_2 will have a harder neutron spectrum than those with UO_2 fuel due to the fuels' higher density and change in hydrogen to heavy metal ratio [76]. The result is that the value of the doppler temperature coefficients are slightly less negative, but the values of the moderator temperature coefficients are more negative. Additionally, while the thermal conductivity of the U_3Si_2 is much greater than UO_2, the heat capacity is lower [77]. Detailed neutronics evaluations should be performed to determine the net effect of reactivity increases on core power and fuel temperature to rule out the possibility of power instabilities as a result of these combined effects. It is also possible that the harder neutron spectrum could affect the swelling and growth of other fuel assembly components and reactor core internals. As the spectral shift primarily affects the thermal neutron spectrum (<1 eV) and changes in fast neutron flux (>1 MeV) at a given power level are relatively minor, these effects are likely to be inconsequential. However, supporting empirical evidence should be generated to substantiate this argument.

Examination of the U_3Si_2 evaluation table reveals that many of the impacts of U_3Si_2 fuel pellets on fuel properties and performance during normal operations are unknown. For example, the higher thermal conductivity will reduce the fuel centreline temperature compared to UO_2, though the melting temperature is much lower. Therefore, it is hard to assess the impact of this concept of margin to melt. Section 6.4 below identifies some key data gaps needed to characterise fuel properties and performance and fully comprehend potential impacts.

6.2.2. *AOOs and postulated accidents*

Examination of the U_3Si_2 evaluation table reveals that many of the impacts of U_3Si_2 fuel pellets on fuel properties and performance under anticipated operational occurrence (AOO) and postulated accident conditions are unknown. Section 6.4 below identifies some key data gaps needed to characterise fuel properties and performance and fully comprehend potential impacts.

6.3. New phenomena

The exothermic reaction of U_3Si_2 with water and steam is expected to be more severe than UO_2. The long-term stability of U_3Si_2 fuel pellets while operating within fuel rods with breached cladding needs further investigation. Similarly, uranium silicide fuel pellet behaviour following postulated accidents for the extended period of time required to remove long-lived decay products needs to be considered.

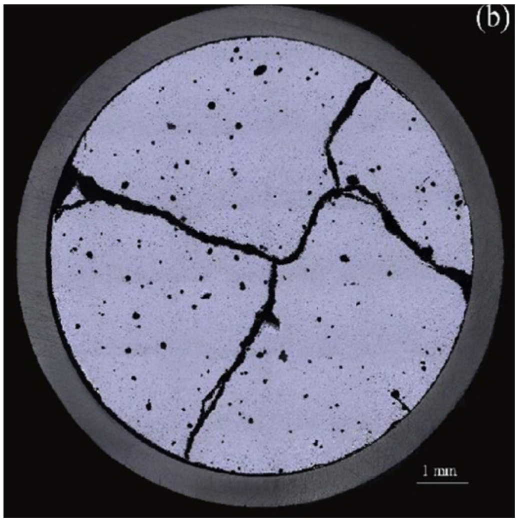
A new performance metric should be considered to address these phenomena. The metric may concern the amount of U_3Si_2 that is allowed to interact with the coolant before the reactor is required to shut down, or additional metrics assuring that such interactions do not occur. Current regulatory criteria require that fuel damage be excluded in normal operations and operational transients due to known failure modes. However, fuel failures do occasionally occur in plants due to unique and unanticipated conditions. Additional metrics or protections may be required for reactors operating with U_3Si_2 due to the increased consequence of fuel rod failure. Given the lower melting temperature of this fuel, there is also an increased vulnerability of interaction between molten fuel and water in severe accident conditions.

6.4. Data gaps

While, the existing empirical database is limited, the evaluation tables identified the types of data needed to characterise the material properties and performance of U_3Si_2 fuel in normal operations, AOO transients, and design basis accidents. Recommended activities include acquiring irradiation behaviour at intermediate burnups on monolithic pellets in order to establish material behaviour models (e.g. swelling, fracturation, fission gas release, creep, thermal conductivity). Irradiation experiments performed at the Idaho National Laboratory (INL) irradiated to ~20 GWd/MTU provide some early indications on fuel performance. The tests showed low fission gas release rates (0.05%) and small fission gas bubbles corresponding to relatively small swelling rates [78].

In-pile testing with online measurement of fuel centreline temperature and plenum pressure could be used to validate some of these material models. Additionally, given the known reactions of U_3Si_2 in water and steam environments, irradiation testing on leaking, irradiated fuel rods is needed to appropriately characterise washout behaviour. U_3Si_2 will likely have different pellet cracking behaviour at both high and low linear heating rates and the release of corrosive fission products will likewise be affected. The INL irradiation of U_3Si_2 showed a low density of radial cracks for irradiations to ~20 GWd/MTU for linear heat rates of ~400 W/cm. Results are shown in Figure 6-2. In-pile testing under various power ramp and AOO transients should take place to assess how changes in the pellet composition affect stress corrosion cracking thresholds resulting from pellet-cladding interaction.

Figure 6-2: **Metallography mount from irradiated U_3Si_2 fuel under moderate heating rates**



Source: Reproduced courtesy of INL [81].

Fuel rod transient irradiation testing under LOCA and rapid power excursion conditions should be conducted to characterise any new, unanticipated failure modes, as well as to establish analytical thresholds for fuel pellet fragmentation and transient fission gas release. Aside from the neutronic effects discussed above, the lower heat capacity and lower melt point of U_3Si_2 fuel will result in high thermal expansion strains for a given enthalpy rise and much lower enthalpies to fuel melting. In slower transients, however, the increase in thermal conductivity of the fuel may result in smaller fuel enthalpy rises for a given energy release. The effects of molten fuel interaction with the cladding need to be carefully assessed to ensure that a coolable geometry is maintained following the transient. These interactions will depend on the cladding type that is paired with this fuel concept. Some initial transient irradiations were conducted at the Idaho National Laboratory in the fall of 2019. Transients were above the enthalpy required to melt the fuel (~450 J/g) [79]. Initial radiography results indicate that the fuel geometry is maintained following irradiation but more data is needed to confirm these results. For transients where cladding rupture is anticipated as part of the accident sequence, the effect of fuel-coolant interaction will also need to be assessed. Uranium Silicide fuel will likely react exothermically with steam and water. These reaction enthalpies will need to be developed so they can be included in the transient analysis.

Furthermore, current material characterisation and irradiation programmes have been conducted on lab-scale specimens. These specimens are manufactured using an arc-melting process to generate U_3Si_2 powders from Uranium and Silicon metal [80]. The resulting phase purity, stoichiometry, and grain size distribution may vary somewhat from pellets produced via a commercial manufacturing process starting from UF_6. Therefore, the applicability of these current data sets to future production-scale fuel is unknown. Some validation experiments should take place to ensure that material models, failure modes and analytical limits developed from experiments with lab-scale samples are applicable to commercially produced fuel pellets. For many of these reasons, the EGATFL report ranks this design concept at a technology readiness level (TRL) below 3.

Given the low TRL of U_3Si_2 fuel pellets, this report was unable to quantify the impacts on existing nuclear fuel safety and design requirements. Therefore, data gaps and research needs were only described qualitatively. The fundamental irradiation testing described in this chapter is detailed in the U_3Si_2 evaluation table next to the appropriate fuel safety criteria.

6.5. Opportunities for collaborative research

Many of the data gaps identified can be filled with detailed investigations of irradiated fuel rods at various burnup intervals. Many unknown elements identified in the evaluation matrix concern unknown swelling and fission gas release behaviour as well as the evolving thermal properties and depletion characteristics. A well fleshed out steady-state irradiation testing campaign with varying heat generation rates and burnup targets with well-planned post-irradiation examination efforts can fill most of these data gaps.

A principal concern with U_3Si_2 is the washout performance. Irradiation ramp testing to failure would provide needed data not only in terms of pellet-cladding interaction performance, but also allow for the evaluation of the effects of operating with a leaking rod after pellet-cladding contact has occurred. This could provide a more realistic look at U_3Si_2 washout performance than experiments with a large pellet-cladding gap, in which there is a much higher exposed fuel surface area than would be likely in an operating reactor.

Finally, both in-pile reactivity-initiated accident (RIA) and LOCA transient testing could provide useful data to the performance of U_3Si_2 in off-normal conditions. The lower melting point of U_3Si_2 combined with the lower heat capacity means that this fuel will perform much differently in both over power and undercooling transients. It is possible that new performance metrics, or at least updates to the existing analytical limits will be required to accommodate this new concept. A collaborative testing campaign in RIA and LOCA conditions can provide the needed data to support the development of these new metrics.

Chapter 7. Recommendations

Recognising that many research projects were ongoing within various international organisations such as the NEA, the International Atomic Energy Agency (IAEA), and the European Union (EU), this Technical Opinion Paper (TOP) identified data gaps in the existing empirical database needed to fully characterise the properties and performance for each of the accident-tolerant fuel (ATF) technologies. In this chapter, potential opportunities for collaborative international research programmes are discussed to fill these data gaps.

Table 7-1 summarises the research needs and data gaps for each ATF technology. Research needs are divided by the type of facilities and capabilities within each facility needed to fill the specific data gap. This cross-cutting format illustrates how any given research facility could support multiple ATF technologies.

Table 7-1: **Opportunities for collaborative research**

Opportunities for collaborative research Research needs	Data gaps				
	Coated zirconium cladding	FeCrAl cladding	SiC Cladding	Doped UO_2 pellets	U_3Si_2 pellets
Commercial reactors					
Lead test assembly (LTA) long-term irradiation	✓	✓	✓		✓
Poolside examination of commercial LTA (e.g. growth, distortion, profilometry, corrosion)	✓	✓	✓		✓
Irradiation-assisted creep and growth measurements	✓	✓			
Hot cell facilities					
Examination and microscopy (e.g. hydrogen, grain size, rim)	✓	✓	✓		✓
Mechanical testing (e.g. YS, uniform elongation, fatigue)	✓	✓	✓		
Fission gas release (FGR) measurements (rod puncture)					✓
Fuel pellet melting analysis					✓
Chemical assay of depleted fuel pellets					✓
Electron probe microanalysis (EPMA) of depleted fuel pellets					✓
Integral loss-of-coolant accident (LOCA) testing (e.g. ballooning, embrittlement, fuel fragmentation, relocation, and dispersal [FFRD])	✓	✓	✓	✓	✓
Research test reactors					
Long-term irradiation	✓	✓	✓		✓
Online FGR measurements and mass spectrometer gas analysis					✓
Online fuel temperature measurements			✓	✓	✓
Irradiation-assisted creep and growth measurements	✓	✓			
Power ramp testing	✓	✓	✓	✓	✓
Reactivity-initiated accident prompt pulse testing (e.g. cladding failure, melt, rod fracture, FFRD, fuel-to-coolant interaction)	✓	✓	✓	✓	✓
Integral LOCA testing (e.g. ballooning, embrittlement, FFRD)	✓	✓	✓	✓	✓
Irradiation of damaged fuel			✓		✓

Priority should be given to research supporting the licensing of near-term ATF technologies: chrome-coated zirconium alloy cladding and doped UO_2 fuel pellets. Examination of Table 7-1 reveals several data gaps for these near-term ATF technologies.

Recognising the calendar time for long-term irradiation campaigns, radionuclide decay (i.e. cooling), and transportation to hot cell facilities, priority must also be given to research needs for the most commercially viable long-term ATF technologies.

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Annex 1. Evaluation tables

The following evaluation tables are available on the NEA website, at www.oecd-nea.org/7576-annex:

- Table A.1: Evaluation table for coated zirconium alloy fuel rod cladding
- Table A.2: Evaluation table for iron-chromium-aluminum (FeCrAl) fuel rod cladding
- Table A.3: Evaluation table for silicon carbide fuel rod cladding
- Table A.4: Evaluation table for doped uranium dioxide ceramic fuel pellets
- Table A.5: Evaluation table for doped uranium silicide ceramic fuel pellets.

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CSNI Technical Opinion Paper No. 19

Following the accident at the Fukushima Daiichi Nuclear Power Plant, many countries began funding research and development on nuclear fuel designs with enhanced accident tolerance (ATFs). ATFs have improved designs, materials and performance features compared with those of the current generation of slightly enriched UO_2 ceramic pellets within cylindrical zirconium alloy cladding.

This report evaluates the applicability of existing fuel design and performance requirements to some of the new ATF designs (coated zirconium alloy fuel rod cladding, FeCrAl fuel rod cladding, silicon carbide fuel rod cladding, doped uranium dioxide ceramic fuel pellets, uranium silicide ceramic fuel pellets), identifies new phenomena which create the need for new or different performance metrics and design requirements, identifies data gaps and discusses opportunities for international collaborative research to fill them.

A variety of new phenomena were identified for the examined ATF designs which challenged the applicability of existing performance metrics and analytical limits or created the need for new criteria. Recommendations to address these challenges are provided with the intention to inform future international research programmes and support ATF licensing.

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