IAEA TECDOC SERIES

IAEA-TECDOC-2032

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# Status of Knowledge for the Qualification and Licensing of Advanced Nuclear Fuels for Water Cooled Reactors

IAEA

International Atomic Energy Agency

# IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

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# STATUS OF KNOWLEDGE FOR THE QUALIFICATION AND LICENSING OF ADVANCED NUCLEAR FUELS FOR WATER COOLED REACTORS

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# FOREWORD

IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants, was published in 2019. SSG-52 provides recommendations on meeting the safety requirements for the design of the reactor core, established in IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design. SSG-52, a Safety Guide, is intended mainly for nuclear power plants that use natural and enriched uranium dioxide fuels and plutonium blended uranium dioxide fuel (mixed oxide fuel) with zirconium alloy cladding. The Safety Guide can be applied with judgement to innovative fuel materials or cladding materials other than zirconium alloys.

The development and qualification of advanced cladding materials (including advanced zirconium cladding alloys, advanced steels and silicon carbide composite) and fuel materials (e.g. doped uranium dioxide, high density fuels) have made significant progress in the past decade, since the initiation of various national research and development and industry projects launched after the accident at the Fukushima Daiichi nuclear power plant in March 2011.

Specific safety related attributes that have been researched for some advanced fuels (initially referred to as accident tolerant fuels and now also known as advanced technology fuels) include reduced reaction kinetics with steam, lower hydrogen generation rate and reduction of stored energy in the core.

As some of these fuel technologies are approaching high levels of technology readiness, increased effort is being devoted by regulatory bodies in some Member States to making the necessary preparatory steps for reviewing future applications by licensees.

The aim of this publication is to review the status of qualification and licensing of some advanced fuel technologies for water cooled reactors and to consider the applicability of the current IAEA Safety Requirements and Safety Guides in addressing the safety of advanced fuels in design and operation, identifying specific aspects that might be transposed into recommendations in the future (including a future revision of SSG-52).

The IAEA wishes to thank the experts from Member States involved in the drafting and review for their valuable contributions to this publication. The IAEA officer responsible for this publication was S. Massara of the Division of Nuclear Installation Safety.

## CONTENTS

# 1. INTRODUCTION

## 1.1. BACKGROUND

Among a broad spectrum of activities aiming at enhancing the safety of currently operating nuclear power plants, the accident that occurred at the Fukushima Daiichi nuclear power plant (NPP) in March 2011 also provided momentum to the development of nuclear fuels with enhanced performance and improved behaviour in accident conditions.

Various national research and development and industry projects were launched in 2011 to develop and validate new fuels, initially referred as Accident Tolerant Fuels, now also known as Advanced Technology Fuel (ATF). Such fuels are characterised by an improved behaviour in accident conditions by delaying the onset of fuel melt with respect to the current uranium oxide zirconium alloy fuel system, while maintaining or improving the performance during normal operation including operational transients. Specific researched safety related attributes for ATF included: reduced reaction kinetics with steam; lower hydrogen generation rate; and reduction of stored energy in the core.

Furthermore, researched enhanced safety features have to be associated with acceptable economic performance (in terms of cost per kilo-watt-hour and compatibility across all the fuel cycle stages, from front-end to back-end) — in many cases associated with increased burnup rates — as a necessary condition for operating organizations to initiate licensing applications aiming at loading advanced nuclear fuels into current NPPs.

National projects (in some cases involving direct governmental financial support) are currently being pursued by fuel vendors in many Member States (including China, France, Japan, Russian Federation, Republic of Korea and the United States), often associated with out-of-pile and in-pile experiments developed (or planned) in other Member States (including Belgium, Czech Republic, Germany, Norway and Switzerland), resulting in the down selection of two main categories of materials:

- Evolutionary technologies – featuring low to medium progress with respect to current fuels, deployable in the relative short term (approximately less than 10 years). This category includes coated and improved zirconium alloys; advanced steels; improved uranium oxide.
- Revolutionary technologies – expected to feature higher benefit but requiring more extensive development and qualification, expected to become available in the longer term (typically more than a decade). This category includes SiC-SiC composite cladding and various advanced fuel designs (including high density or metal fuel).

International multilateral activities aiming at supporting such initiatives were initiated at the OECD NEA, with a comprehensive state-of-the-art report on the status of knowledge of ATF [1], published in 2018.

At the IAEA Department of Nuclear Energy, a Coordinated Research Project (CRP) on the Analysis of Options and Experimental Examination of Fuels with Increased Accident Tolerance (ACTOF) [2] was completed in 2019, followed by another CRP on Testing and Simulation of Advanced Technology Fuels (ATF-TS), started in 2021 under the auspices of the IAEA Technical Working Group on Fuel Performance and Technology.

As some evolutionary technologies are approaching high levels of technology readiness, regulatory bodies in some Member States are undertaking preparatory steps to facilitate (or streamline) licensing applications, by evaluating whether current regulatory requirements, acceptance criteria and regulatory guidance documents would be applicable for advanced fuels licensing, and updating these when necessary.

An update of regulatory guidance documents for the licensing of advanced fuels may be associated, in some cases, to a necessary revision of the applicability of currently applied criteria (e.g. maximum oxidation rate and maximum cladding temperature in loss of coolant accident (LOCA) conditions, etc.) to advanced fuels, for example because of a different phenomenological behaviour of advanced fuels with respect to current fuels, requiring the development of new models in fuel thermomechanical numerical codes.

More generally, the validation of numerical codes used for deterministic safety analysis of advanced fuels remains a challenge because of the scarcity of the experimental base on advanced materials, given that models implemented in thermomechanical numerical codes rely more on empirical correlations (derived on the experimental base for the current uranium oxide zirconium alloy fuel system) than on mechanistic models which might be extended more easily to advanced fuels.

## 1.2. OBJECTIVE

The objective of this publication is primarily to identify and collect the status of knowledge in Member States on the applicability to advanced fuels of regulatory requirements, acceptance criteria and regulatory guidance documents that have been developed for the licensing of current nuclear fuel systems.

This publication also aims at reviewing the status of qualification and licensing of some advanced fuel technologies for water cooled reactors, with a focus on evolutionary as well as revolutionary technologies.

In addition, this publication considers the applicability of the current IAEA Safety Requirements and Safety Guides in addressing the safety of advanced fuel in design and operation, identifying possible specific aspects which may be transposed into recommendations in the future. It is anticipated that some recommendations of the IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants [3] will be affected by specific features of advanced nuclear fuels.

## 1.3. SCOPE

The publication encompasses evolutionary and revolutionary advanced nuclear fuel technologies potentially envisaged for the reactor core of water cooled reactors (in particular light water cooled reactors).

The publication addresses the expected impact of advanced nuclear fuels on the current regulatory requirements, acceptance criteria and regulatory guidance documents supporting the licensing of nuclear fuel, focusing on design and operation. Hence, the impact of advanced nuclear fuels on other fuel cycle processes (like manufacturing, transport, interim storage, reprocessing and final disposal) will not be dealt with in this publication.

The publication also identifies the applicability of the relevant IAEA safety requirements and recommendations provided in Safety Guides in addressing the safety of advanced nuclear fuel in the field of design and operation.

## 1.4. STRUCTURE

The TECDOC consists of six sections, two appendices and an annex.

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

Section 2 illustrates related IAEA safety standards pertaining to nuclear fuel design and operation.

Section 3 describes current regulatory requirements, acceptance criteria, and regulatory guidance documents for nuclear fuel licensing in various Member States.

Section 4 summarizes the main features, status of knowledge of expected performances and identified challenges for safety, and the status of development, qualification (and, whether appropriate, licensing) for some evolutionary and revolutionary advanced nuclear fuels under development.

Section 5 evaluates the applicability of the current IAEA safety standards in addressing the safety of advanced nuclear fuel, pertaining to nuclear fuel design and operation.

Section 6 illustrates the applicability of existing models, criteria and regulatory guidance documents for the licensing of advanced nuclear fuels.

Appendix I presents an excerpt of requirements and recommendations from IAEA safety standards, relevant for the design and operation of the reactor core for NPP.

Appendix II documents an evaluation of the applicability of SSG-52 [3] in addressing some advanced nuclear fuel technologies.

An Annex details an individual approach applicable to Canadian Deuterium Uranium (CANDU) reactors in Canada.

## 2. IAEA SAFETY STANDARDS FOR THE DESIGN, QUALIFICATION AND LICENSING OF NUCLEAR FUELS FOR NUCLEAR POWER PLANTS

Several IAEA safety standards provide guidance for the design and safety assessment of the reactor fuel and nuclear fuel for water cooled reactors, as well as for conducting regulatory oversight of processes associated to the nuclear fuel (including its design, transport, in-reactor operation), including the following:

- IAEA Safety Standards Series Nos GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [4], and GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [5], address more specifically the areas of the regulatory framework for safety and of the safety assessment;
- IAEA Safety Standards Series Nos SSR-2/1 (Rev.1), Safety of Nuclear Power Plants: Design [6], and SSR-2/2 (Rev.1), Safety of Nuclear Power Plants: Commissioning and Operation, contain requirements in the areas of design, commissioning and operation;
- SSG-52 [3] provides recommendations on meeting the safety requirements established in SSR-2/1 (Rev.1) [6] in the design of the reactor core for nuclear power plants;
- IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [7] provides recommendations on performing deterministic safety analysis for nuclear power plants;
- IAEA Safety Standards Series No. SSG-12, Licensing Process for Nuclear Installations [8] provides recommendations on meeting the requirements relating to authorization by the regulatory body.

An excerpt of general safety requirements, specific safety requirements, as well as recommendations from IAEA safety guides on how to meet the abovementioned requirements in the design and operation of the reactor core for NPP is illustrated in the Appendix I.

This consistent set of requirements and recommendations is, in most cases, mirrored by national regulations developed in several Member States, which will be illustrated in Section 3 of this TECDOC.

# 3. REGULATORY REQUIREMENTS, ACCEPTANCE CRITERIA AND REGULATORY GUIDANCE DOCUMENTS FOR THE LICENSING OF CURRENT NUCLEAR FUEL

As detailed in Section 2, IAEA safety standards establish fundamental safety principles and requirements on the design and operation of NPPs "primarily for land based stationary nuclear power plants with water cooled reactors designed for electricity generation or for other heat production applications (such as district heating or desalination)" [6]. Similar requirements, as well as acceptance criteria and guidance documents, captured in national regulations, have been used to carry out the safety assessment (and licensing) of the current generation of nuclear fuel (i.e., $UO_2$ ceramic pellets within zirconium alloy cladding arranged in a square fuel bundle array), and various examples will be mentioned in subsection 3.2.

The demonstration of the successful implementation of the fundamental safety principles and safety requirements may be strongly influenced by fuel design and utilization since many of these items important to safety are judged on their ability to protect against or limit damage to the reactor core. Such demonstration also relies on proven fuel performance, fulfilment of acceptance criteria and the ability of analytical models and methods to predict, with a high level of confidence, fuel performance under a wide range of operational states and accidental conditions. For example, to demonstrate the capability to mitigate the consequences of a LOCA and remove long term decay heat, detailed analytical models that predict fuel rod behaviour under LOCA conditions are needed along with analytical limits that ensure acceptable fuel performance (e.g. maximum peak cladding temperature of 1204°C).

Nuclear fuel qualification will be considered in subsection 3.1.

## 3.1. FUEL QUALIFICATION

Fuel development and qualification activities often occur in parallel. Building on past experience, Idaho National Laboratory published a paper describing a detailed approach for the development and qualification of fuel for light water reactor (LWRs) [9], which stated that:

> "the approach is described as four phases, with emphasis on selecting a reference fuel concept, evaluating and improving the fuel to develop a fuel specification for a reference design, obtaining data to support a licensing safety case for the fuel, and final qualification of the fuel for a specific application."

The stated objective of nuclear fuel qualification is the demonstration that a fuel product fabricated in accordance with a specification behaves as assumed or described in the applicable licensing safety case, and with the reliability necessary for economic operation of the reactor.

The United States Nuclear Regulatory Commission (U.S. NRC) published NUREG-2246, Fuel Qualification for Advanced Reactors [10] "to identify criteria that will be useful for advanced reactor designers through an assessment framework that would support regulatory findings associated with nuclear fuel qualification". This framework provides criteria, derived from regulatory requirements, that when satisfied would support regulatory findings for licensing. This framework follows a top down approach, in which a set of base goals supports high level regulatory requirements. Reference [10] provides the bases for the identified goals and clarifying examples for the expected evidence used to satisfy those goals, stating that, specifically, "the assessment framework particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate evaluation models and empirical safety criteria" [10].

In general, once the fuel design and manufacturing specifications have been finalized, the next stage in fuel qualification is to obtain the data needed to support the licensing safety case. The first step is to fully characterize the mechanical, material, thermal, chemical, and nuclear properties, and the impact of irradiation under reactor coolant conditions on these properties. Separate effect tests and integral tests of fresh and irradiated fuel segments will be required to characterize these properties for the full range of operating conditions including design basis accidents (DBA). One of the main goals of this research and testing is to satisfy the following fuel qualification needs:

- Identify all degradation mechanisms and failure modes;
- Establish performance metrics that satisfy the fundamental safety principles, safety requirements, acceptance criteria and regulatory requirements;
- Define acceptance criteria (i.e., analytical limits) that satisfy the performance metrics.

Another major goal of the research and testing is to gather data necessary to calibrate and validate analytical models, as well as establish model prediction uncertainties. The ability of analytical models and methods to predict, with a high level of confidence, fuel performance under normal operation and accident conditions is needed to demonstrate that fundamental safety principles and safety requirements are satisfied, and that safety related structures, systems, and components (SSCs) perform their intended safety functions. Reference [10] provides a systematic evaluation and justification of the qualification of nuclear fuel.

### 3.1.1. Lead test rods and lead test assemblies

Lead test assemblies (LTAs) are a necessary and important step in the fuel development and qualification process. These fuel assemblies may contain new mechanical design features (e.g. grid spacers with mixing vanes), new nuclear design features (e.g. doped $UO_2$ fuel pellets), and/or new materials (e.g. advanced zirconium cladding alloys); however, these have to be designed to be geometrically, mechanically, and thermalhydraulically compatible with coresident fuel assemblies and reactor internals. LTA programs may be comprised of only a few lead test rods (LTRs) in a single fuel assembly or comprise of several full assemblies with many new design features.

LTA irradiation programmes provide knowledge of, and experience with, irradiated material properties and performance, which is critical for qualifying analytical codes and methods and for developing the design bases to license new fuels or design features for unrestricted use. In particular, LTA programmes accomplish the following tasks:

- Collection of data to characterize irradiated material properties and performance;
- Provision of irradiated material for subsequent hot cell examination, characterization, and research;
- Demonstration of in-reactor performance.

The licensee is responsible for assessing its ability to irradiate LTAs in accordance with its license and has to comply with all applicable regulatory requirements to ensure no undue risk to the public. As part of the plant's license, operational limits and conditions define limiting conditions for operation, equipment operability requirements, and surveillance requirements. Often, the technical specifications for operational limits and conditions include a provision for the irradiation of a limited quantity of LTAs. For example, the Standard Technical Specifications[1] in Ref. [11] state that:

> "a limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions."

The specification of a 'limited number' and 'nonlimiting core regions' are purposely vague given the variety of new design features which could be present in an LTA. For the LTA safety demonstration, the degree of characterization of the irradiated material properties and performance of the new design feature dictates the allowable quantity and operation of LTAs in a commercial reactor core. For example, this provision would allow a larger quantity of LTRs containing doped $UO_2$ fuel pellets with extensive in-reactor experience and testing versus the allowed quantity of LTRs containing developmental non $UO_2$ fuel pellets with minimal characterization. As part of this safety demonstration, the licensee has to demonstrate that the presence of the LTAs does not negatively impact the performance of neighbouring co-resident fuel nor interfere with the reliability and performance of SSCs important to safety. Given their restricted quantity and operation, the LTA safety demonstration may often rely on analytical models and methods that are of lower confidence than approved models and methods used to assess overall reactor safety.

To clarify regulatory requirements and encourage LTA irradiation programmes, fuel vendors have requested approval of more refined definitions of 'limited number' and 'nonlimiting core regions'. For example, Ref. [12] has defined specific performance requirements for three distinct types of lead assemblies: (1) lead test assemblies, (2) lead use assemblies, and (3) high burnup lead use assemblies. This approach provides an acceptable licensing framework for satisfying technical specifications and irradiating LTAs without prior approval from the regulatory body.

## 3.2. FUEL LICENSING

National regulatory documents and/or practices are illustrated in the following subsection.

### 3.2.1. Belgium

In Belgium, as in other OECD NEA countries, the U.S. NRC rules and guidance (e.g. regulatory guides, standard review plan) are generally followed, but with specific requirements for some particular aspects, illustrated below.

For example, in Belgium:

> "new fuel design is licensed on the basis of geometrical, mechanical, neutronic and thermal-hydraulic criteria. For each new fuel, a compatibility document is prepared. This document shows that the new fuel does not violate any criterion (geometrical, mechanical, neutronic and thermal-hydraulic), neither for a full new fuel core as well as for transition reloads. To demonstrate the thermal-hydraulic compatibility of a new fuel design, statistical methods are used by most of the designers. Specific penalties related to some non-rigorous aspects of the approaches are adopted, and additional verifications are

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[1] Standard Technical Specifications may be viewed on the U.S. Nuclear Regulatory Commission website at https://www.nrc.gov/docs/ML1810/ML18100A045.pdf [11].

conducted when statistical combinations of the uncertainty on the departure from nucleate boiling (DNB) correlation is used.” [13]

The technical support organization needs to adopt generic margins to guarantee defence in depth in the process. “Special attention is devoted to the assembly design (stresses, fatigue, vertical load, etc.) and to the rod design (pressure, maximal heat flux, power capacity, transient behaviour, etc.)” [13].

A fuel rod thermal–mechanical design report is then submitted for approval to the technical support organization: it has to demonstrate the adequate behaviour of the fuel rod in normal operation and in anticipated operational occurrence (AOO). In particular, the pellet-cladding interaction induced stress corrosion cracking (PCI-SCC) under AOO has to be considered by either the generic demonstration or a specific verification for any new fuel products. A LOCA fuel safety evaluation report is also to be provided to demonstrate the compatibility of the new fuel to the reference fuel used in the current licensing basis for LOCA safety analysis.

When new materials are introduced in previously approved fuel assemblies (e.g., Zirlo, optimized Zirlo, M5, mixed uranium-plutonium oxide fuel (MOX), gadolinium, etc.), experimental feedback results are needed for the licensing. The introduction of LTRs or LTAs is subject to specific licensing review, as for the application for licensing of new fuel designs.

### 3.2.2. Canada

In Canada:

> “the Canadian Nuclear Safety Commission (CNSC) licensing process includes a thorough assessment of the application submission and, if the licence is being renewed, a verification of the licensee’s compliance performance. All licence applications are assessed based on the risk-ranking of the proposed licensed use type.” [14]

The CNSC outlines the licensing process that covers new licence application, financial guarantees, licence renewal, licence transfer, submitting an application, licence amendment requests, and revoking a licence.

> “A licence application goes through the following steps within the CNSC:
>
> 1. Entry into CNSC’s electronic records system;
> 2. Assessment for relevant cost-recovery fees, if applicable;
> 3. Entry into CNSC’s licensing database;
> 4. Technical assessment by a licensing specialist;
> 5. Quality assurance;
> 6. Sign-off by a designated officer (if the application and the applicant meet all regulatory requirements);
> 7. Licence issuing to licensee.” [14]

### 3.2.3. France

In France, the French Nuclear Safety Authority (ASN) established several guidelines or basic safety rules, based on the state of art and industrial practices.

Some of these safety guides contain safety principles and requirements related to nuclear fuels, such as:

- Guide n°22 [15] concerning the design of pressurized water reactors (PWRs);
- Guide n°28 [16] concerning the validation of scientific computing tools used in the safety case for PWRs – applicable to the first barrier to the release of radioactive materials.

These guides, jointly produced by ASN and the French Institut de Radioprotection et de Sureté Nucléaire, are non binding but provide a coherent set of recommendations for achieving safety objectives.

Furthermore, in France, industrial codes and standards are produced by the nuclear industry and formalized by the French Association for Nuclear Steam Supply Systems Equipment Construction Rules (AFCEN). In the past, the French Association for Nuclear Steam Supply Systems Equipment Construction Rules established the design and construction rules for fuel assemblies of PWR nuclear power plants. Although the development of such industrial codes and standards lies within fuel vendors and licensees' responsibility, ASN and and the French Institut de Radioprotection et de Sureté Nucléaire can sometimes recognize their acceptability in relation to the safety demonstration.

### 3.2.4. Japan

In Japan, regulatory requirements for the licensing of nuclear fuel are contained in the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors [17] (in particular the Article 43-3-6 (Criteria for the Permission) (1)(iv) and the Article 43-3-14 (Maintenance of Power Reactor Facilities).

Regulatory requirements related to fuel design and performance are also contained in two ordinances issued by the Nuclear Regulation Authority (NRA):

- NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities [18], where the Article 15 (Reactor core) is relevant for fuel design and performance;
- NRA Ordinance Prescribing Technical Standards for Commercial Power Reactors and their Auxiliary Facilities [19], where the Article 23 (Reactor core) is relevant for the fuel design and performance.

Additional guidance on ways to meet the regulatory requirements related to fuel design and performance is found in the regulatory guides to the above NRA Ordinances, which includes regulatory criteria related to fuel.

### 3.2.5. Republic of Korea

In the Republic of Korea, the regulatory requirements and acceptance criteria are based on the articles of the Regulations on Technical Standards for Nuclear Reactor Facilities [20]. The relevant articles for fuel design are Article 17 (Reactor design), Article 18 (Inherent protection of reactor), Article 28 (Instrumentation and Control system), Article 30 (Emergency core cooling system), and Article 35 (Reactor core).

Additionally, supporting documents, regulatory standards and safety review guidelines for LWRs are established by the Korean Institute of Nuclear Safety, which are positioned lower than the abovementioned articles in the hierarchy of applicable regulatory documents. Chapter 5.2 (KINS/RS-N05.02) in regulatory standards, and chapter 4.2 (Fuel system design) in the safety review guidelines [21] support the articles of Ref. [20], although these documents are not legally binding. Specific guidance for reviewing the fuel design is described in chapter 4.2 (Fuel system design) in the safety review guidelines [21].

### 3.2.6. Russian Federation

In the Russian Federation, the main requirements for fuel elements and fuel assemblies of water water energetic reactors (WWER) are presented in the federal rules and regulations for the utilization of atomic energy, NP-094-15 [22].

NP-094-15 [22] contains acceptance design criteria for fuel assemblies, that were developed on the basis of past operation experience, including on the following parameters:

- Stress values in fuel cladding;
- Diameter of fuel rods;
- Fuel rod elongation;
- Fuel temperature;
- Oxide coating thickness;
- Gas pressure within the fuel rod;
- Hydrogen content.

In accordance with [22], safety factors have to be established for such parameters.

Safety justification of nuclear fuel is performed in accordance with Ref. [23], which states that bench tests and reactor tests of new nuclear fuel need to be implemented to confirm the fulfilment of safety criteria. However, this document does not contain specific regulatory requirements for such tests, which are performed on a case by case basis.

The general practice is to perform testing of new nuclear fuel in research reactors (for applications in the water water energetic reactors, the multi loop reactor MIR.M1 reactor in the JSC 'SSC RIAR' research centre is used) as well as irradiation of test assembly in power reactors prior to full scale operation. Tests in research reactors are performed in normal operation as well as in various typical accident conditions (e.g. reactivity insertion accident (RIA), LOCA). Irradiations of separate specimens of new materials in commercial reactor are also performed prior to testing of a fuel assembly. Special attention is given to the validation of computer codes, where results of reactor tests play an important role in providing the necessary experimental data.

A similar process is being adopted for the licensing of recycled mixed uranium and plutonium fuel, with currently several test assemblies undergoing irradiation in NPPs. Prior to loading LTA in the reactor core, various LTRs of recycled mixed uranium and plutonium fuel were irradiated and eventually underwent post-irradiation examination.

The same process is currently being applied for LTRs with ATF (Cr coated E110 fuel [24] and Cr nickel 42XHM cladding fuel [25]).

New federal rules and regulations are currently under development, with the aim of formalizing the process for nuclear fuel testing and safety justification: this future regulatory document will cover topics such as criteria and requirements for testing of nuclear fuel prior to operation in NPPs and requirements for pilot operation of new fuel in NPPs.

### 3.2.7. United Kingdom of Great Britain and Northern Ireland

In the United Kingdom, the nuclear site licensee is expected to demonstrate, via the production of a safety case, that the design, manufacture and operation of the nuclear fuel used at their facility reduces risks so far as is reasonably practicable. The starting point for such a demonstration is for the licensee to prove that its proposed fuel design is consistent with relevant good practice (RGP). It is the nuclear site licensee's responsibility to set out the standards and codes that were used and to justify them as RGP. In its supporting evidence, the licensee will need to show how its design and the associated implementation meet the identified standards (e.g., through quality arrangements, safety analysis, inspections etc) and, hence, is consistent with RGP.

Regulatory oversight of new fuel designs is performed based on a graded permitting process of the licensee's supporting safety case conducted by the United Kingdom Office for Nuclear Regulation (ONR). To aid reaching a judgement on the adequacy of the licensee's demonstration, regulatory assessment activities are informed by guidance provided in the following documents:

- ONR's Safety Assessment Principles for Nuclear Facilities [26];
- ONR's Nuclear Safety Technical Assessment Guides (TAGs) Safety of Nuclear Fuel in Power Reactors [27];
- Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable) [28];
- Validation of Computer Codes and Calculation Methods [29].

The regulatory body will also consider international standards and other applicable sources of RGP.

If RGP is not available due to the novelty of the proposed fuel design, the nuclear site licensee has to demonstrate by alternate means that adequate levels of safety are achieved. ONR will need to be assured that such cases demonstrate equivalence to the outcomes associated with the use of the Safety Assessment Principles [26], and such a demonstration may need to be examined in greater depth to gain that assurance.

### 3.2.8. United States of America

In the United States, the United States Code of Federal Regulation Title 10, Energy, Part 50, Domestic Licensing of Production and Utilization Facilities [30], Appendix A, General Design Criteria for Nuclear Power Plants, defines principal design criteria that establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The following criteria, particularly relevant for the design of the nuclear fuel, are of key importance for licensing fuel designs:

- Criterion 10: Reactor design;
- Criterion 11: Reactor inherent protection;
- Criterion 12: Suppression of reactor power oscillations;
- Criterion 17: Electric power systems;
- Criterion 20: Protection system functions;
- Criterion 25: Protection system requirements for reactivity control malfunctions;
- Criterion 26: Reactivity control system redundancy and capability;
- Criterion 27: Combined reactivity control systems capability;

- Criterion 28: Reactivity limits;
- Criterion 29: Protection against anticipated operational occurrences;
- Criterion 33: Reactor coolant makeup;
- Criterion 34: Residual heat removal;
- Criterion 35: Emergency core cooling;
- Criterion 50: Containment design basis.

Guidance for the licensing of nuclear fuel is contained within NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition [31].

In Ref. [31], guidance related to fuel design and performance requirements is captured in Chapter 4.2 (Fuel system design), Chapter 4.3 (Nuclear design) and Chapter 4.4 (Thermal and hydraulic design). Guidance related to the compliance demonstration for SSCs important to safety is mainly found in Chapter 6.3 (Emergency core cooling system) and Chapter 15 (Transient and accident analyses).

Additional guidance documents are cited in [31].

# 4. DEVELOPMENT AND QUALIFICATION OF ADVANCED FUEL TECHNOLOGIES

A variety of advanced nuclear fuel technologies are being actively developed by fuel vendors worldwide, as documented by the broad spectrum of designs included in recent reviews conducted at the OECD NEA [1], [32] and at the IAEA [2].

Without being fully exhaustive, this section focuses on a few advanced nuclear fuel technologies, highlighting general features (by referring, where appropriate, to previous reviews) while providing references accessible in the open literature detailing individual fuel designs developed by various technology developers. The review covers both evolutionary and revolutionary technologies, under development for deployment in water cooled reactors either in the near term (less than 10 years) or on a longer term (more than 10 years).

The description in this section is mainly based on the state-of-the art of advanced fuel technologies for LWRs. In principle, advanced fuel technologies described here are also applicable to pressurized heavy water cooled reactor fuel unless otherwise specified.

For CANDU reactors, specific developments were undertaken with the purpose of enhancing the critical heat flux (CHF) of the fuel bundle, hence restoring margins to dryout in the sagged channel due to ageing phenomena (see Annex for additional information).

## 4.1. CLADDING MATERIALS UNDER DEVELOPMENT

Researched benefits of advanced cladding materials compared to zirconium alloys are an increased resistance to high temperature steam oxidation, with a consequential expected improvement of the behaviour in accident conditions (both DBA and design extension conditions, DEC).

In LOCA conditions, researched improvements are:

- An increase of the coping time (which measures the time from the initiation of an accident to cladding failure due to post quench embrittlement);
- A lower value of the peak cladding temperature, achieved through a much lower contribution from the exothermic oxidation of zirconium;
- A reduction of the amount of hydrogen generation, with a reduced risk of containment failure in the event of hydrogen explosion.

Improved behaviour in transient conditions is often associated with improved characteristics in normal operation, with reduced wear and fretting resulting in increased nuclear fuel reliability.

### 4.1.1. Coated zirconium alloys

The deposition of a coating material on the outer surface of the cladding aims to provide an increased resistance to high temperature oxidation, while also improving fuel reliability in normal operation (reduced wear, fretting and corrosion).

The advantage of adopting a coating material lies is that the cladding mechanical properties are substantially governed by the underlying zirconium alloy, and hence are relatively unaffected (especially in the presence of thin coatings). For the same reason, the neutron penalty on the $^{235}U$ enrichment (due to the higher capture neutron cross-section of the coating material versus zirconium) is relatively low.

Coated zirconium alloys are under development worldwide, including in China, France, Japan, Republic of Korea, Spain, Russian Federation and the United States.

#### 4.1.1.1. *General description*

These alloys are constituted by:

- A monolithic zirconium alloy (such as N36 in China, M5 in France, MDA in Japan, HANA in the Republic of Korea, E110 in the Russian Federation, ZIRLO and Optimized ZIRLO in the United States);
- A layer of either metallic coating (pure Cr, Cr alloys, FeCrAl and Cr and/or FeCrAl multi layer) or ceramic coating (nitrides, MAX phases).

The majority of designs are constituted by pure Cr or CrAl coating:

- Cr coated zirconium alloy, developed by the China Nuclear Power Technology Research Institute [33];
- Cr coated zirconium alloy by the Nuclear Power Institute of China [34];
- Cr coated M5 by Framatome (France and United States) for PWR and coated Zircaloy 2 for Boiling Water Reactor (BWR) [35], [36];
- Cr coated zirconium alloy by Mitsubishi Nuclear Fuel and Mitsubishi Heavy Industries (Japan) [37];
- CrAl or metal Cr coated HANA-6 by KEPCO Nuclear Fuel (Republic of Korea) [38];
- Cr coated E110 by TVEL (Russian Federation) [24];
- ARMOR™ (Coated zirconium alloy) by Global Nuclear Fuel (United States) [39];
- EnCore®[1] Cr coated zirconium alloy cladding by Westinghouse (United States) [40].

Various processes were developed for the coating deposition, including: physical vapour deposition (with coating thickness in the range 10–20 μm), cold spraying (coating thickness of 20–30 μm), 3D laser melting and coating (coating thickness of 40–80 μm).

#### 4.1.1.2. *New phenomena*

References [1], [32] identify the following new phenomenological features with respect to standard zirconium alloys, requesting special consideration during the development and qualification of coated zirconium alloys:

- Possible formation of chromium zirconium eutectic. Consequently, the formation and evolution of the eutectic integrity needs to be evaluated.
- Specificities of the coated cladding embrittlement process (due to the diffusion of chrome into the zirconium alloy and the progressive reduction of the protectiveness of the coating due to oxygen diffusion) and related applicability (and/or adaptation) to coated cladding of the standard LOCA equivalent clad reacted (ECR) criterion established for uncoated zirconium alloy.
- Presence of a new mechanism for enhanced hydrogen uptake, i.e. hydrogen permeability from coolant through coating surface into bulk.

Data gaps were identified for the following phenomena:

- Cladding oxidation;
- Hydriding;

- Cladding ballooning (including prototypical peroxidation and hydrogenating) and rupture;
- Cladding fracture due to embrittlement during LOCA and pellet-cladding mechanical interaction (PCMI) or post DNB failure during RIA;
- The effect of coating layer surface condition on PWR CHF (in high flow and high pressure conditions).

*4.1.1.3. Expected performance in normal operation*

Given the relatively low thickness of the coating (generally in the range 20–80 μm), chromium coated cladding generally exhibits similar performances as uncoated zirconium alloys in normal operation.

In addition, coated cladding features reduced wear and fretting, with reduced waterside corrosion kinetics and hydrogen pick-up.

However, due to the increased parasitic neutron capture in the coating, neutronic penalties may arise (resulting in an increase of $^{235}U$ enrichment), particularly in the case of thick coatings (e.g. beyond 30 μm).

A surveillance plan coordinated with the implementation of chromium coated zirconium alloy cladding to monitor the coolant chemistry will mitigate any impact of chromium ions.

Performance expectations and challenges are summarized in Table 1.

TABLE 1. SUMMARY OF PERFORMANCE EXPECTATIONS AND CHALLENGES WITH COATED ZIRCONIUM ALLOYS

| | Performance expectations | Challenges |
|---|---|---|
| Normal operation | Pros<br>- Reduced corrosion kinetics and hydrogen pick-up<br>- Increased wear resistance<br>- Good irradiation resistance<br>- No adverse impact of coating on CHF and CRUD<br>- Acceptable fatigue performance for coating<br>Cons<br>- Neutronic penalty (increase of $^{235}U$ enrichment) particularly with thick coating | - Benign performances to be confirmed also under reactor irradiation and coolant chemistry<br>- Compatibility with current reprocessing process for spent nuclear fuel |
| Accident conditions (DBA and DEC without significant fuel degradation) | Pros<br>- Significantly reduced high temperature steam oxidation leading to reduced heat and hydrogen production (LOCA conditions)<br>- Increased post quench ductility in LOCA conditions, leading to reduced creep and ballooning<br>- Strengthening effect at high temperature, enhanced burst resistance, resulting in increased time to rupture (LOCA)<br>Cons<br>- Potential Cr Zr eutectic formation | |

*4.1.1.4. Expected performance in accident conditions (design basis accidents and design extension conditions without significant fuel degradation)*

Major areas of progress with respect to uncoated zirconium alloy cladding are:

- A significantly reduced high temperature steam oxidation leading to reduced heat and hydrogen production;
- Increased post quench ductility in LOCA conditions, leading to reduced creep and ballooning, enhanced burst resistance (e.g. potentially smaller opening size), resulting in increased time to rupture.

The possibility of formation of chromium zirconium eutectic needs to be addressed, and its potential consequences in accident conditions need to be evaluated to exclude any detrimental effect on the evolution of the accident.

*4.1.1.5. Status of development and qualification*

Coated zirconium alloys are in an advanced stage of development and qualification worldwide, with LTR and LTA undergoing irradiation in commercial reactors:

- In Belgium, ENUSA and/or Westinghouse are irradiating LTRs in the Doel 4 NPP [41], [42];
- In China, the irradiation of LTR and/or LTA is undergoing regulatory review prior to loading into PWR;
- In France, irradiation of LTR and/or LTA is in preparation, after successful irradiations of LTA and/or LTR in the Gösgen NPP (Switzerland) [35], [36], [43];
- In the Republic of Korea, the irradiation of samples in research reactors is underway, while the introduction of LTR and/or LTA in commercial reactors is still in the early planning phase (planned after 2023);
- In the Russian Federation, the irradiation of 12 LTRs with chrome coating in unit No 2 of Rostov NPP (VVER) began in 2021. LTRs are inserted in a standard TVS-2M fuel assembly. In parallel, mid 2022, similar LTRs underwent the fourth irradiation cycle in the multi loop reactor MIR.M1research reactor;
- In the United States, the irradiation of LTR and/or LTA is ongoing, associated either with standard $UO_2$ fuel or in combination with doped $UO_2$:
  - In PWRs: Entergy’s ANO 1, Southern’s Vogtle, Exelon’s Byron 2 and Exelon’s Calvert Cliffs NPPs;
  - In BWRs: Southern’s Hatch, Exelon’s Clinton and Xcel Energy’s Monticello NPPs.

*4.1.1.6. Status of licensing*

Currently, no fuel assembly designs with coated cladding were approved for batch reload in any Member State having contributed to this publication.

### 4.1.2. Advanced steels

For the purpose of this publication, advanced steels are those developed for high temperature applications in industrial fields and are considered to be applicable for nuclear fuel claddings.

Advanced steels feature an excellent resistance to high temperature steam oxidation, combined with a good behaviour in normal operation (in particular a good resistance to debris fretting).

Also, the more benign mechanical properties and generally lower creep rates (compared to zirconium alloy) are beneficial in accident conditions and allow the design of a thinner cladding tube (which also partially counterbalances the increased neutron capture cross-section compared to zirconium alloy).

#### *4.1.2.1. General description*

This group of materials includes various categories of FeCrAl and FeCrAl oxide dispersion strengthened alloys (ODS):

- FeCrAl ODS developed by the China Nuclear Power Technology Research Institute [33];
- FeCrAl by the Nuclear Power Institute of China [44];
- FeCrAl ODS by Global Nuclear Fuel Japan, Hitachi-GE Nuclear Energy and Nippon Nuclear Fuel Development [45], [46];
- IronClad™ (monolithic FeCrAl) by Global Nuclear Fuel (United States) [47].

In addition to advanced steels, alternatives to zirconium alloys are being developed including chromium nickel alloy ferritic martensitic 42XHM by TVEL (Russian Federation) [25].

#### *4.1.2.2. New phenomena*

New phenomena identified in Ref. [32] related to FeCrAl and FeCrAl ODS include:

- Embrittlement at low temperature: in addition to embrittlement due to high temperature oxidation (which may also affect zirconium alloy) FeCrAl may be subjected to chromium rich alpha prime precipitation within the cladding, which may lead to low temperature (lower than 500°C) embrittlement. The risk of chromium rich alpha prime precipitation has to be given due consideration both in normal operation and in accident conditions.
- FeCrAl features a different corrosion behaviour with respect to zirconium alloys: in addition to the formation of an oxide layer (as for zirconium alloys), the corrosion process of FeCrAl also involves its dissolution into the coolant. Hence a new performance metric associated to the cladding wall thinning by corrosion needs to be formulated.
- Phenomena related to high temperature embrittlement:
  - Possible unstable oxidation which includes a direct oxidation of iron in the case of fast LOCA heat up kinetics;
  - Possible eutectic reaction with neighbouring materials, which needs to be analysed as this may induce a possible reduction of the cladding melting temperature.

#### *4.1.2.3. Expected performance in normal operation*

Advanced steels exhibit increased fretting resistance and improved mechanical properties. Nevertheless, the following drawbacks are identified:

- A neutron penalty (due to the increased capture cross-section in the thermal domain with respect to zirconium alloy). This may be compensated by:
  - An increased $^{235}U$ enrichment (in many cases beyond 5%);
  - A reduced cladding thickness, due to the increased mechanical strength of FeCrAl: however, it should be recalled that the final design will result from a trade off of both neutronic and mechanical performances;
  - The possible adoption of higher density fuel pellet materials (e.g. metal, nitride or uranium silicide, see subsection 4.2.2).
- Permeability to tritium, which may necessitate dedicated coatings to avoid tritium contamination of the water coolant. Moreover, additional research needs to be performed to evaluate the risk of selective oxidation of elements like aluminium, molybdenum and tungsten (these could be contained as impurities), which could potentially increase the filter activity.
- Stress corrosion cracking (SCC) susceptibility to caesium (while the absence of susceptibility to iodine was highlighted in scoping studies).

#### *4.1.2.4. Expected performance in accident conditions (design basis accidents and design extension conditions without significant fuel degradation)*

The main recognised advantages of advanced steels (compared to zirconium alloys) are a high corrosion resistance in steam–water coolant and low hydrogen production due to the absence of a steam–zirconium reaction. In addition, advanced steels are not affected by embrittlement through hydrogen and oxygen.

Performance expectations and challenges are summarized in Table 2.

TABLE 2. SUMMARY OF PERFORMANCE EXPECTATIONS AND CHALLENGES WITH ADVANCED STEELS

| | Performance expectations | Challenges |
|---|---|---|
| Normal operation | Pros<br>- Improved resistance to debris fretting and wear<br>- Improved mechanical properties (yield strength, ultimate tensile strength, elastic modulus)<br>Cons<br>- High permeability to tritium from fuel pellet through the cladding<br>- Worsened neutron economy due to increased thermal neutron capture<br>- SCC susceptibility to caesium | - Ductility reduction at high temperature<br>- Radiation induced hardening<br>- Compatibility with current reprocessing process for spent nuclear fuel |
| Accidental conditions (DBA and DEC without significant fuel degradation) | Pros<br>- Slow oxidation rate at temperatures above 1200°C leading to reduced heat and hydrogen production (LOCA conditions)<br>Cons<br>- Lower melting temperature vs. zirconium alloys | Effect of temperature ramp rate on high temperature oxidation |

#### 4.1.2.5. *Status of development and qualification*

Advanced steels are in an advanced stage of development and qualification, with LTR and/or LTA undergoing irradiation in various countries:

- In Japan, recrystallized FeCrAl ODS sheet materials were irradiated in the High Flux Isotope Reactor at ORNL, allowing the investigation of the modification of mechanical properties at beginning of life conditions;
- In the Russian Federation, the irradiation of 12 LTRs with 42XNM chromium nickel alloy coating in unit No 2 of Rostov NPP (VVER) began in 2021. LTRs are inserted in a standard TVS-2M fuel assembly. In parallel, mid 2022, similar LTRs underwent the fourth irradiation cycle in the multi loop reactor MIR.M1research reactor;
- In the United States, LTR associated with standard $UO_2$ are being irradiated in BWRs (Southern's Hatch and Exelon's Clinton NPPs).

#### 4.1.2.6. *Status of licensing*

Currently, no advanced steels designs were approved for batch reload in any Member State having contributed to this publication.

### 4.1.3. SiC-SiC composite

SiC-SiC composite cladding constitutes an attractive solution, thanks to its improved basic properties (absence of phase change, with sublimation temperature at 2700°C, chemical inertness), benign neutronic features (lower thermal neutron capture cross-section) and attractive characteristics at high temperature (retention of strength under irradiation, excellent oxidation resistance with steam) with respect to zirconium alloys.

SiC-SiC composite is being developed in several countries, due to its broad targeted applications, as cladding material for both PWRs and BWRs and as channel box for BWRs:

- China (by the China General Nuclear Power Group (CGN) [33] and the Nuclear Power Institute of China [48]);
- France (by Framatome [49], in a joint effort with the Commissariat à l'Energie Atomique et aux Energies Alternatives);
- Japan (by Toshiba [50] and Hitachi-GE Nuclear Energy [51], [52], [53]);
- United States (by Westinghouse [40] and General Atomics, jointly developing the design named SIGA [54]).

This concept is also being investigated in the Republic of Korea by KEPCO NF and in the Russian Federation by TVEL.

#### 4.1.3.1. *General description*

This material is composed of various layers of fibres ($SiC_f$) and monolithic SiC ($SiC_m$) with design specificities as described below:

- In France, Framatome and the Commissariat à l'Energie Atomique et aux Energies Alternatives's sandwich design consists of two layers of $SiC_f$-SiC composites separated by a thin ductile metal liner (tantalum).
- In the United States, General Atomics is developing a duplex cladding design, with an inner SiC-SiC composite (which carries the tensile loading), densified using chemical

vapour infiltration, and an outer SiC monolith (which ensures hermeticity and provides improved corrosion resistance) using a chemical vapour deposition process. A similar two layer design is being developed by the China General Nuclear Power Group in China and by Toshiba in Japan.
- In Japan, Hitachi-GE Nuclear Energy is investigating the use of titanium coatings as a countermeasure for hydrothermal corrosion [51], [52], [53].

The main challenges for the development and qualification of SiC composite cladding are:

- Hydrothermal corrosion, with production of silica that may dissolve in primary water;
- Thermomechanical stresses which may induce microcracking in the SiC matrix, resulting in a possible release of gaseous fission products;
- Development of end-plug sealing technology, capable of withstanding high neutron radiation doses in a corrosive environment.

*4.1.3.2. New phenomena*

Given that the development of this material is at an early stage, the identification of new phenomena might not yet be fully comprehensive, nevertheless the following are mentioned in [32]:

- Dissolution of silica in water and other environmental effects encountered during the operation of a full SiC core or a mixed core with SiC containing subassemblies.
- All phenomena that might be related to the possible loss of a coolable geometry (other than embrittlement due to oxidation and hydriding, which affects zirconium alloys but not SiC).
- Interactions between SiC, $UO_2$ (or other fuel material) and steam at high temperature (including possible generation of methane and carbon monoxide).
- Absence of creep and reduced cladding ductility to accommodate cladding stresses during normal operation.

*4.1.3.3. Expected performance in normal operation*

SiC-SiC composites are developed primarily for their expected superior performance in accident conditions, and more predictable dimensional stability properties under irradiation with respect to zirconium alloy, saturating at relatively low fluence.

Identified concerns needing special consideration are related to:

- Fission product and tritium retention capability;
- Lower fuel volume fraction due to the gap thickness increase (as a consequence of the reduced ductility), which may introduce a penalty on uranium enrichment versus zirconium alloy;
- Depending on the manufacturing process and the effect of irradiation, a lower thermal conductivity could have an impact on the fuel rod thermal behaviour;
- Absence of creep and reduced cladding ductility.

*4.1.3.4. Expected performance in accident conditions (design basis accidents and design extension conditions without significant fuel degradation)*

SiC-SiC is expected to keep its mechanical integrity and dimensional stability in accident conditions (high temperatures) while featuring a low interaction with steam.

Numerical simulation studies show a sharp improvement of the dynamic response during DEC without significant fuel degradation: results from modelling and simulation show that fuel melting is substantially delayed, with a substantial slowing down of the accident progression, hence allowing additional grace time to mitigate the accident evolution. While quantitative results vary among specific scenarios, the same trend is noted in references [33], [49], [55], [56]. These features require further experimental confirmation.

Performance expectations and challenges are summarized in Table 3.

TABLE 3. SUMMARY OF PERFORMANCE EXPECTATIONS AND CHALLENGES WITH SILICON CARBIDE CLADDING

| | Performance expectations | Challenges |
|---|---|---|
| Normal operation | Pros<br>- Reduced thermal neutron absorption<br>- Higher strength at high temperature<br>- Chemical inertness until 1700°C<br>- Good resistance to radiation damage and dimensional stability<br>- Improved CHF performance due to surface conditions<br>Cons<br>- Fission products retention capability<br>- Higher $^{235}U$ enrichment (> 5%) due to gap thickness increase and consequential fuel pellet diameter reduction<br>- Hydrothermal corrosion<br>- Absence of creep and reduced cladding ductility | - Increase of fission gas release from fuel pellets due to irradiation induced swelling and low thermal conductivity of SiC cladding and increased gap thickness<br>- Possibility of bowing for BWR channel box<br>- Potential PCMI (linked to low ductility) and PCI-SCC issues |
| Accidental conditions (DBA and DEC without significant fuel degradation) | Pros<br>- Outstanding oxidation resistance and lower hydrogen production in high temperature steam, maintaining integrity in most DEC A[a] scenarios<br>- Coolable geometry maintained at high temperature (no ballooning not burst)<br>- Lack of a melting point (SiC sublimates at 2700°C)<br>Cons<br>- Recession in water at high temperature, which may be responsible for wall thinning of SiC cladding, potentially leading to fission products release and cladding failure | - Response under applied loads given the lack of ductility (e.g. seismic loads)<br>- Applicability of traditional performance metrics and corresponding analytical limits for LOCA and RIA |

a DEC without significant fuel degradation.

#### 4.1.3.5. *Status of development and qualification*

The short term focus is the acquisition of properties through irradiation of samples in research reactors and irradiation of LTR planned for the forthcoming years in commercial reactors:

- In the United States, irradiation of LTR in commercial reactors is planned in the forthcoming years both by Westinghouse and/or General Atomics and by Framatome.
- LTR have been irradiated since 2016 by Framatome in the Gösgen NPP in Switzerland and in the United States (MITR in 2021–2022 and in ATR starting in 2022), and further irradiations in research reactors are under preparation in Belgium (BR-2).
- In Japan, the irradiation of coupon samples in a test reactor was conducted.
- In the Russian Federation, the irradiation of samples in research reactors and of LTR and/or LTA in VVER-1000 is foreseen for 2025 and 2026, respectively.

#### 4.1.3.6. *Status of licensing*

Currently, no SiC-SiC designs were approved for batch reload in any Member State.

## 4.2. FUEL MATERIALS UNDER DEVELOPMENT

Various designs – aimed at enhancing the fission product retention and minimising pellet to clad interaction – were reviewed in [1], including:

- Doped $UO_2$ pellets, where an increased grain size is obtained thanks to dopants (chromia, chromia alumina), seeking an improvement of the viscoplastic behaviour;
- Microcell $UO_2$ pellets [57], where both an enhanced retention of fission products and thermal conductivity increase are obtained through the adoption of a microcell structure (either metallic or ceramic).

In the remainder of subsection 4.2, the focus will be on technology characterised by a higher maturity level, i.e. the doped uranium oxide fuel.

### 4.2.1. Doped $UO_2$

Advanced or doped $UO_2$ pellets are $UO_2$ pellets manufactured with additives incorporated during the blending process such as chromium and aluminium oxides. After sintering, those additives can lead to a pellet microstructure with a higher density and an increased grain size compared to standard $UO_2$.

This can bring improvements on several aspects (like thermal conductivity, fission gas retention, viscoplasticity) which are considered beneficial for the thermomechanical behaviour of the fuel rod in normal operation, AOOs and DBAs. Hence, margins are expected to be improved in terms of fuel temperature limit, internal pressure for normal operation and accidents conditions and PCI-SCC. The addition of dopants in the $UO_2$ induces a reduction in the fissile mass, however the higher density of these pellets can generally compensate for this effect.

Doped $UO_2$ fuel is being developed in various countries, due to expected superior thermal and mechanical properties:

- $Cr_2O_3$ doped $UO_2$ developed by Framatome [35];
- ADOPT$^{TM}$ fuel ($Al_2O_3$ $Cr_2O_3$ doped $UO_2$) by Westinghouse (United States) [40].

In addition, doped $UO_2$ is also being developed by the China General Nuclear Power Group in China [33].

*4.2.1.1. General description*

The addition of dopants to standard $UO_2$ fuel aims at achieving the following expected benefits:

- An increased grain size, which may lead to an increased retention of gaseous fission products (higher intragranular retention, reducing the amount of gas available for release at grain boundaries), with a lower internal pressure at high burnup (expected to reduce ballooning and burst failure risk);
- An increased viscoplasticity, which may lead to enhanced PCI-SCC margin at high temperature;
- Decreased fuel oxidation rate for leaking fuel rods when exposed to reactor coolant, which may lead to reduced washout rate compared to standard $UO_2$;
- A higher density, which may partially counterbalance the increase in $^{235}U$ enrichment with other advanced cladding materials alternatives to zirconium alloys (advanced steels, for example). This might also contribute to earlier gap closure.

*4.2.1.2. New phenomena*

No new phenomena were identified in the review documented in Ref. [32]. Hence the licensing basis is the same as for the standard $UO_2$ fuel.

*4.2.1.3. Expected performance in normal operation*

In normal operation, expected benefits are a reduced washout rate and an increased retention of gaseous fission products, with an improved PCI-SCC behaviour due to an increased viscoplasticity.

Earlier gap closure due to higher density can potentially influence rod growth and swelling.

*4.2.1.4. Expected performance in accident conditions (design basis accidents and design extension conditions without significant fuel degradation)*

In accident conditions, expected benefits are a reduced fission gas release and improved fuel fragmentation relocation and dispersal (including in LOCA conditions). This may need to be confirmed by experimental tests.

Performance expectations and challenges are summarized in Table 4.

*4.2.1.5. Status of development and qualification*

Doped $UO_2$ are in an advanced stage of development in France and in an advanced stage of development and qualification in the United States.
In the United States, LTAs are being irradiated in PWRs (Southern's Vogtle, Exelon's Byron 2 and Exelon's Calvert Cliffs NPP).

*4.2.1.6. Status of licensing*

Several designs of doped pellets have been approved for batch reload in BWR and PWR in the United States and Europe.

TABLE 4. SUMMARY OF PERFORMANCE EXPECTATIONS AND CHALLENGES WITH DOPED $UO_2$

| | Performance expectations | Challenges |
|---|---|---|
| | Pros | |
| Normal operation | - Reduced wash-out rate<br>- Increased pellet-cladding interaction (PCI-SCC) margin at high temperature<br>- Increased density and grain size<br>- Increased viscoplasticity facilitating load following modes | - Lower fission gas release (to be confirmed) |
| Accidental conditions (DBA and DEC without significant fuel degradation) | | - Reduced fission gas release during transients (to be confirmed)<br>- Improved behaviour vs. fuel fragmentation, relocation and dispersal during LOCA (to be confirmed) |

### 4.2.2. High density fuels

Fuel materials with uranium densities higher than $UO_2$ could bring significant economic benefits to ATF concepts, in particular in association with those cladding materials (other than zirconium alloys) that induce a neutronic penalty due to the increased neutron capture cross-section. Hence, the increased fissile density may (at least partially) counterbalance the increase of $^{235}U$ enrichment due to the adoption of advanced cladding materials.

Some of these candidate materials, namely uranium silicide, uranium nitride and metal fuel, also have improved thermal properties compared to $UO_2$ resulting in additional safety benefits. An improved thermal conductivity with respect to uranium oxide provides various advantages:

- A reduced amount of stored energy prior to accidental transients (e.g. LOCA), hence reducing the peak cladding temperature;
- The reduced pellet temperature allows:
  - In normal operation, a reduced pellet cracking and improved pellet-clad interaction behaviour;
  - A reduction of fission gas release in normal operation, with consequential decrease of the plenum volume, and reduced ballooning and burst in LOCA conditions.

Advantages related to an increased thermal conductivity (and hence lower temperature in normal operation) have nevertheless to be weighted based on the possible variation of the melting temperature (in some cases lower than for uranium oxide), which decreases the margin with respect to fuel melting.

Despite development projects underway in various Member States, references in the open literature are relatively scarce (see [1] for additional information on this topic). For this reason, the evaluation of applicability of SSG-52 [3] to high density fuels could not be conducted and will not be presented in Section 5.

# 5. APPLICABILITY OF IAEA SAFETY STANDARDS IN ADDRESSING ADVANCED NUCLEAR FUELS

Section 2 and subsection I.3.1 in the Appendix I describe the recommendations provided in SSG-52 [3] to meet the requirements of SSR-2/1 (Rev. 1) [6] pertaining to the design of the reactor core for NPPs. Section 3 described national frameworks currently established in Member States for the development, qualification and licensing of current nuclear fuels. Both SSG-52 and national frameworks are mainly based on natural and enriched uranium dioxide fuels and plutonium-blended uranium dioxide fuel (mixed oxide fuel) with zirconium alloy cladding.

This section evaluates the applicability of the recommendations provided in SSG-52 [3] in addressing specific features of the four advanced fuel technologies described in Section 4:

- Three of the technologies are expected to complete their development in the short term: chromium coated zirconium alloys (illustrated in subsection 4.1.1), advanced steels (subsection 4.1.2) and doped $UO_2$ (subsection 4.2.1).
- One technology is expected to complete its development in the long term: SiC-SiC composite (illustrated in subsection 4.1.3).

Given that information in the public literature on high density fuels under development is relatively scarce, the applicability of SSG-52 [3] to such technologies was not evaluated.

## 5.1. RATIONALE

The following approach was adopted to evaluate the applicability of the recommendations provided in SSG-52 [3] to these advanced technologies:

1. Gap analysis: based on the new phenomena and degradation mechanisms identified for each technology in Section 4, a screening analysis was first performed to identify the paragraphs of SSG-52 [3] that might be impacted.
2. Technology applicability analysis: for each of the impacted SSG-52 [3] paragraphs, the technical applicability was evaluated, concluding on either of the following:
   - Minor impact: in such case the metrics of the design limits remain applicable, but the quantitative values may change. Additional tests may be needed to verify the applicability, or determine or confirm the values of the design limits.
   - Major impact: in such case, new metrics may need to be introduced and the corresponding new design limits determined by new tests.
3. Identify the need for new recommendations: for each of the impacted SSG-52 [3] paragraphs, the possible need of elaborating new recommendations (in association with the presence of new phenomena) was also evaluated.

The detailed evaluation is documented in the Appendix II, with a table for each innovative technology documenting the applicability of each recommendation and, when appropriate, the suggested modifications allowing to enhance the applicability of SSG-52 [3] recommendations (or the additional phenomena needing consideration).

The remainder of this section summarizes the major conclusions arising from this evaluation.

## 5.2. APPLICABILITY OF SSG-52 IN ADDRESSING THE SPECIFIC FEATURES OF ADVANCED NUCLEAR FUEL TECHNOLOGIES

The following subsections illustrate the applicability of the recommendations provided in SSG-52 [3] in addressing specific features of advanced nuclear fuel technologies (the subsections are labelled following the structure of SSG-52 [3]).

### 5.2.1. Section 1 (Introduction)

This section of SSG-52 [3] describes features generally applicable to all four advanced fuel technologies considered in this evaluation.

Suggested modifications allowing to enhance the applicability of SSG-52 [3] recommendations are considered as minor, e.g. to include advanced fuels designs among those addressed, for example in paras 1.5–1.6, which define the nuclear fuel technologies specifically covered by SSG-52 [3].

### 5.2.2. Section 2 (General safety considerations in the design of the reactor core)

This section describes a general approach that is generally applicable to all four advanced fuel technologies considered in this evaluation.

Suggested modifications allowing to enhance the applicability of SSG-52 [3] recommendations are considered as minor, e.g. to include the advanced fuels designs among those addressed by SSG-52 [3] (for example, adding in para. 2.2 the new phenomena and degradation mechanisms identified for the advanced fuel technologies – e.g., eutectic formation, diffusion of chromium into the zirconium alloy, coating integrity for coated cladding, etc.).

### 5.2.3. Section 3 (Specific safety considerations in the design of the reactor core)

A summary of the results of the evaluation conducted for coated zirconium alloys, advanced steels, doped $UO_2$ and SiC-SiC composite is provided in the following subsections, while the Appendix II provides the detailed results of this evaluation.

#### *5.2.3.1. Coated zirconium alloys*

As discussed in subsection 4.1.1, coated zirconium alloys are among the most promising technologies for short term implementation due to the maturity of the coating technology and the expected benefits to the operating organizations. However, several new phenomena or mechanisms have to be addressed in the design and safety analysis, and some new degradation phenomena identified for coated zirconium claddings (such as the coating integrity during normal operation and accident conditions) also need to be considered.

The evaluation concludes that the recommendations of SSG-52 [3] are mostly applicable to coated zirconium claddings, except when considering the impact of the difference in material properties, which need to be duly considered (or confirmed through additional tests).

The Appendix II provides detailed information concerning the paragraphs of SSG-52 [3] that are impacted by specific features of coated cladding zircaloy alloys. An excerpt is summarized in Table 5, which also presents suggested modifications to address the specific features of coated zirconium alloys.

TABLE 5. SUMMARY OF PARAGRAPHS IN SSG-52 [3] AFFECTED BY SPECIFIC FEATURES OF COATED ZIRCONIUM ALLOYS

| Para | Suggested modifications to enhance applicability to coated zirconium alloys |
|---|---|
| 3.5 | Neutron penalty, mechanical properties, corrosion behaviour, oxidation behaviour at high temperature, susceptibility to SCC. |
| 3.22 | To add ‘these analyses should be performed for all fuel types in the core (e.g. zirconium cladding or Cr coated zirconium cladding, or $UO_2$ or mixed oxide fuel)’. |
| 3.30 | Fuel assembly CHF tests need to be performed with Cr coated zirconium cladding. |
| 3.39 | Add a new performance metric ‘integrity of the coating’. |
| 3.40 | In operational states, limits on the peak fuel temperature are established to prevent fuel melt, the applicability of limits established for uncoated zirconium alloy to coated claddings needs to be confirmed. In particular, the possibility of formation of eutectic needs to be addressed. |
| 3.41 | Some phenomena (relating to cladding corrosion and hydriding) need to be considered, in addition to considering the effect of the difference of material properties. |
| 3.51 | Same as para. 3.41. |
| 3.54 | The impact of coating on the CRUD formation and effects on heat transfer need to be studied or confirmed. |
| 3.59 | Mechanical property, cladding design. |
| 3.60 | Cladding embrittlement due to in-reactor hydriding is material dependent. The effect of the difference of material needs to be considered. |
| 3.63 | The PCI-SCC phenomena in presence of coating needs to be studied with sufficient ramp tests. |
| 3.66 | Material property, corrosion behaviour, cladding design. Need to add performance metric on ‘integrity of coating’. |
| 3.72 | Mechanical property. The impact of coating layer on the ballooning or bursting during LOCA and PCMI or post DNB fuel cladding failures during RIA need to be performed or confirmed, considering the new phenomena or mechanisms. |
| 3.73 | Material property, corrosion behaviour, cladding design, oxidation behaviour at high temperature. In terms of (a)–(c) and (e), some design limits need to be changed considering the properties of cladding material and their modification with burnup (e.g. embrittlement with hydrogen pickup, cladding corrosion and hydriding, enthalpy rise during RIA). |

Subsection 4.1.1.2 identified data gaps related to cladding oxidation, hydriding, cladding fracture due to embrittlement during LOCA, PCMI or post DNB failure during RIA in irradiated and accident conditions, and the effect of coating layer surface condition on PWR CHF. Hence, the applicability of the ‘current’ design and safety limits to these phenomena needs to be adequately verified or confirmed.

#### *5.2.3.2. Advanced steels*

As described in subsection 4.1.2, advanced steels feature various specificities (e.g. mechanical properties, corrosion behaviours, neutron penalty, etc.) with respect to conventional zirconium alloy. Hence, the paragraphs in SSG-52 [3] that closely relate to these features need careful consideration to confirm their applicability also to advanced steels.

The evaluation concludes that recommendations provided in SSG-52 [3] are mostly applicable to advanced steels: the recommendations are applicable to FeCrAl claddings with due consideration of the difference of material properties.

TABLE 6. SUMMARY OF PARAGRAPHS IN SSG-52 [3] AFFECTED BY SPECIFIC FEATURES OF ADVANCED STEELS

| Para. | Suggested modifications to enhance applicability to advanced steels |
|---|---|
| 3.5 | The effect of the difference of material needs to be considered and confirmed; especially, the susceptibility of FeCrAl material to stress corrosion cracking needs to be considered and confirmed. In particular, the susceptibility of FeCrAl material to stress corrosion cracking needs to be considered and confirmed. Also the susceptibility of FeCrAl cladding to the embrittlement at low temperature (e.g. normal operation temperature) by Cr rich α' (alpha prime) precipitation within the cladding needs to be included. |
| 3.22 | To add: 'these analyses should be performed for all fuel types in the core (e.g. $UO_2$ or mixed oxide fuel, with either zirconium alloy or FeCrAl cladding)'. |
| 3.30 | Fuel assembly CHF tests need to be performed with FeCrAl cladding to quantify the margins. |
| 3.40 | In operational states, limits on the peak fuel temperature are established to prevent fuel melt, it needs to be confirmed the applicability of limits established for uncoated zirconium alloy to FeCrAl cladding. In particular, the reduction of melting temperature of FeCrAl cladding compared to zirconium alloy cladding needs to be addressed. |
| 3.41 | The effect of the difference of cladding material between FeCrAl and zirconium alloy on fuel design and phenomena (e.g. due to corrosion, hydriding, etc.) needs to be considered. |
| 3.51 | The extents of hydrogen pick-up and the embrittlement by hydrogen pick-up are dependent on cladding material, and such properties different from zirconium alloy need to be considered. |
| 3.54 | The deposition behaviour of corrosion products on FeCrAl cladding surface needs to be confirmed. |
| 3.59 | The effects of cladding mechanical properties (e.g. creep rates), which are different with respect to zirconium alloy cladding, need to be confirmed. |
| 3.60 | Mechanical properties, corrosion behaviour related especially to hydrogen pick-up: the effect of the difference of material need to be considered because FeCrAl cladding is not affected by embrittlement through the reaction with hydrogen and oxygen. The thinner wall thickness and lower creep rates of FeCrAl cladding vs. zirconium alloy cladding need to be considered during safety analysis of PCMI. |
| 3.63 | The change in the power-ramp failure threshold of FeCrAl cladding by PCI-SCC phenomena needs to be studied with sufficient power-ramp tests. Analytical predictions need to be verified by experimental studies. |
| 3.66 | The impact of different material properties, corrosion behaviour, and cladding design needs to be duly addressed. Some phenomena (mainly relating to (c), (e)–(g) in this para.) are material dependent. Cladding stress and strain are dependent on fuel cladding design. Hence, the effect of the difference of material and design needs to be considered. In terms of (d), the effects of material properties of FeCrAl cladding (e.g. on fuel temperature), need to be confirmed. |
| 3.72 | The difference in mechanical properties (e.g. mechanical properties at high temperature) needs to be considered and confirmed. |
| 3.73 | The impact of different material properties, corrosion behaviour, cladding design, and oxidation behaviour at high temperature needs to be addressed.<br>In terms of items (a)–(c) and (e) of this para., some limits need to be largely modified considering the cladding design and the properties of cladding material and their modification with burnup (e.g. embrittlement with hydrogen pickup, cladding corrosion and hydriding). An unstable oxidation of FeCrAl cladding at fast LOCA heat up condition needs to be considered. A new performance metric would be needed to include the phenomena of wall thickness thinning, considering that the corrosion process of FeCrAl material is a competing reaction of the formation of oxide layer. In this formulation, effects of cladding heat up rate on unstable oxidation may need to be considered. The reaction between cladding and its neighbouring material (e.g. fuel pellet) at high temperature needs to be considered to determine the acceptable limit. |

New phenomena which are attributed to material properties specific to FeCrAl cladding are considered to have major impacts on the recommendations in some paragraphs in SSG-52 [3]:

while the contents of some paragraphs in SSG-52 [3] do not explicitly relate to properties of cladding material itself, there seem to be data gaps between the requirements in the paragraphs and the data which have been obtained so far (e.g. paras 3.30 and 3.63). In such cases, filling the data gaps (e.g. by in-pile and/or out-of-pile experiments) appears necessary.

The Appendix II provides detailed information concerning the SSG-52 [3] paragraphs that are impacted by advanced steels. An excerpt is summarized in Table 6, which also presents suggested modifications allowing to enhance the applicability of SSG-52 [3] recommendations to advanced steels.

#### *5.2.3.3. Doped $UO_2$*

Doped pellets may have different thermal, thermomechanical, and fission gas release properties due to their denser, coarse grained microstructure, but the pellet characteristics remain mainly those of $UO_2$. This technology does not introduce fundamental changes in the phenomena involved in pellet thermal behaviour, fission gas release or rod thermomechanics.

Doped pellets therefore mainly need a characterization of the pellet and rod behaviour to establish relevant behavioural laws for the conditions that fuel has to withstand.

The evaluation of the applicability of SSG-52 [3] in relation to this technology leads to the following conclusion:

- ‘Doped fuel pellets’ are already mentioned in the definition provided for ‘fuel’ in Annex I of SSG-52 [3], as an example of materials used in fuel pellets (see page 59).
- SSG-52 [3], Annex II lists the aspects to be addressed in the design of various SSCs, including the fuel rod and the fuel assembly. As far as the fuel and the thermomechanical behaviour of the rod are concerned, the listed phenomena remain relevant for doped $UO_2$. Furthermore, no new phenomena ought to be added to the list. The behaviour of this type of pellet and its effect on the rod, nevertheless, has to be the subject of appropriate research with the aim of showing whether existing behavioural laws are relevant or to define new ones if needed.
- The limitations throughout the irradiation cycles for normal operation and AOO are unchanged. For instance, clad strain and stresses or fuel temperature limits in normal operation and AOO are still valid to ensure fuel rod integrity. On the other hand, the expected benefits of doped $UO_2$ pellets (for instance, in terms of LOCA or PCI-SCC behaviour) do not exempt them from a margin assessment against the existing criteria (especially if credit of benign features of doped pellets is taken by the licensee).

Doped $UO_2$ fuel may not facilitate reprocessing. For example, some forms of chromium may generate corrosion issues in nuclear fuel reprocessing facilities. Therefore, these issues need to be considered as part of fuel cycle back end analyses.

The Appendix II provides a detailed evaluation of the SSG-52 [3] paragraphs that are impacted by doped pellet behaviour.

This evaluation concludes that all paragraphs of SSG-52 [3] are applicable, with due consideration of the effect of dopants on material properties.

### *5.2.3.4. SiC-SiC composite*

As introduced in subsection 4.1.3, various advanced features of SiC-SiC composite might involve a real paradigm shift for the qualification and licensing of this material, compared to current or evolutionary cladding materials addressed previously in this publication:

- Advanced characteristics of SiC-SiC composite, including the absence of creep and the reduced cladding ductility;
- Potential presence of new phenomena, such as: dissolution of silica in water and other environmental effects, other phenomena potentially leading to the loss of coolable geometry, the interaction between SiC, $UO_2$ and steam at high temperature.

Therefore, the analytical screening of SSG-52 [3] recommendations against specific features of SiC-SiC composite may not lead to an exhaustive identification of the possible modifications to SSG-52 [3] which might be necessary to fully address SiC-SiC composite cladding.

While the main principles behind most SSG-52 [3] recommendations remain applicable, detailed recommendations may need to be developed or adapted.

While such effort is beyond the scope of this publication, the conclusions summarized in the Table 7 were drawn from this preliminary analysis:

- In some paras it would be sufficient to add the material name (‘SiC-SiC composite’) to signify its applicability to SiC (e.g. paras 3.22 and 3.155).
- While the contents of some paras do not explicitly relate to properties of cladding material itself, there seem to be data gaps between the requirements in the paragraphs and the data which have been obtained so far (e.g. paras 3.30 and 3.63). Hence, it is necessary to fill such data gaps through appropriate in-pile and/or out-of-pile experiments.

In addition to these suggested modifications, the inclusion of new recommendations in SSG-52 [3] addressing new phenomena and/or mechanisms pertaining specifically to SiC-SiC composite cladding (mentioned in subsection 4.1.3.2) need to be considered.

Additional details on the results of the evaluation concerning the applicability of recommendations of SSG-52 [3] to SiC-SiC composite are presented in the Appendix II.

TABLE 7. SUMMARY OF PARAGRAPHS IN SSG-52 [3] AFFECTED BY SPECIFIC FEATURES OF SIC-SIC COMPOSITE

| Para. | Suggested modification to enhance applicability to SiC-SiC composite |
|---|---|
| 3.5 | The effect of the difference of material, e.g. chemical compatibility (i.e. hydrothermal corrosion) of SiC with the coolant at about 300°C, no formation of an oxide layer on the cladding surface and the absence of hydrogen uptake, insusceptibility to corrosive fission products that would cause stress corrosion cracking, low pseudo ductility, relatively poor thermal conductivity under neutron irradiation, irradiation growth and swelling, etc., needs to be considered.<br>Some items are neither suitable nor applicable for SiC-SiC composite cladding owing to its intrinsic properties that are different from those of zirconium alloy. |
| 3.22 | To add ‘these analyses should be performed for all fuel types in the core (e.g. $UO_2$ or mixed oxide fuel, with zirconium alloy or SiC-SiC composite cladding)’. |
| 3.30 | Fuel assembly CHF tests need to be performed with SiC-SiC composite cladding to quantify the margins. |
| 3.40 | In operational states, limits on the peak fuel temperature are established to prevent fuel melt. the applicability of limits established for uncoated zirconium alloy to SiC-SiC composite cladding, and the effects of material properties of SiC-SiC cladding, such as thermal conductivity on fuel temperature, needs to be confirmed. |
| 3.41 | The effect of the difference of cladding material on some phenomena (relating to e.g. thermal conductivity, mechanical properties) needs to be considered.<br>Furthermore, the following sentence needs to be included: ‘the leaktightness of cladding should be maintained even when subject to thermal and/or mechanical stresses generated or expected during irradiation’. |
| 3.51 | The impact of material properties needs to be duly addressed.<br>Considering the absence of the hydrogen uptake caused by corrosion, this para. could be either eliminated, or the following wording needs to be included ‘… for each cladding type excluding SiC-SiC composite’. |
| 3.54 | The deposition behaviour of corrosion products on the cladding surface needs to be confirmed. |
| 3.59 | The occurrence of PCMI needs to be confirmed by experiments and/or analysis. The change in gap width during irradiation is expected to be different from zirconium alloys and needs to be taken into account at the design phase, needs to be considered. |
| 3.60 | Due to the absence of hydrogen uptake of SiC, this para. relating to fuel failure mode by hydrogen induced embrittlement could be eliminated. Instead, the failure mode by low pseudo ductility (especially SiC-SiC composites) of cladding needs to be considered and included. |
| 3.63 | The power-ramp failure threshold of SiC-SiC composite cladding by PCI-SCC or other phenomena need to be studied with sufficient power-ramp tests. Analytical predictions need to be verified by experimental studies. |
| 3.66 | The impact of material properties, chemical compatibility with water coolant, cladding design, and leaktightness maintenance of cladding under irradiation need to be duly addressed.<br>The effect of the difference of cladding material needs to be considered because some phenomena (mainly relating to (c) (e)–(g)) are material dependent. Regarding (d) and (f), the formation of microcracks and/or pores during irradiation needs to be considered in relation to the maintenance of leaktightness of fuel rods. In the case of SiC, it may be necessary to consider the effect of (g) by material dissolution into coolant water. |
| 3.72 | The difference in material properties (e.g. mechanical properties at high temperature) needs to be considered and confirmed. |
| 3.73 | The impact of material properties and cladding design needs to be duly addressed.<br>The design limits relating to paras (a)–(e) would significantly change with consideration of e.g. high resistance of SiC against melting, oxidation in high temperature steam, and rupture at elevated temperatures during LOCA, the absence of hydrogen uptake of SiC during normal operation, low pseudo ductility (especially SiC-SiC composites) of cladding, etc. The reaction between fuel pellet and cladding at high temperature needs to be considered to determine the acceptable limit. |

#### *5.2.3.5. High density fuels*

Given that information on ongoing developments of high density fuels is relatively scarce, the applicability of the recommendations provided in SSG-52 [3] to these technologies was not evaluated.

### 5.2.4. Section 4 (Qualification and testing)

This section describes features generally applicable to all four advanced fuel technologies considered in this evaluation.

Only minor modifications are suggested to enhance the applicability of recommendations of SSG-52 [3] to specific designs (for example, adding in para. 4.9 the qualification and testing related to the integrity of the coating layer for coated zirconium cladding).

### 5.2.5. Annex I (Supplementary technical information)

This Annex provides supplementary technical information to clarify the content of some recommendations.

The evaluation concluded that additional specific technical information related to advanced fuel designs would be necessary to enhance the applicability of recommendations of SSG-52 [3]: for example, adding more detailed information about the identified new phenomena and degradation mechanisms for each specific advanced nuclear fuel.

# 6. APPLICABILITY OF EXISTING MODELS, DESIGN CRITERIA AND REGULATORY GUIDANCE FOR THE LICENSING OF ADVANCED NUCLEAR FUELS

While Section 5 illustrates the result of an analysis aiming at evaluating the applicability of IAEA Safety Guide SSG-52 [3] to a few advanced fuel technologies, this section aims at providing a qualitative evaluation of the applicability of existing analytical models, codes and standards, regulatory requirements and guidance, as well as fuel design and acceptance criteria used during the development and qualification process.

Insights from a recently published technical opinion paper produced by the OECD NEA Working Group on Fuel Safety (WGFS), on the applicability of existing nuclear fuel safety criteria to advanced nuclear fuels, will be used, as appropriate [32]. The WGFS technical opinion paper [32] documents a detailed assessment for five different ATFs technologies against over 50 phenomena important to safety in the design and performance of nuclear fuel. In addition, new phenomena, associated with the unique design and performance aspects of ATF technologies, which challenge the applicability of existing performance metrics and analytical limits or created the need for new criteria, were identified.

For Cr coated zirconium alloy fuel rod cladding and doped $UO_2$ fuel pellets, the WGFS technical opinion paper [32] defined the relative impact as low. This 'relative impact' considers the number and degree of changes in safety criteria, performance and design requirements, and analytical limits. Due to their limited departure from currently licensed fuel technologies and degree of characterization of irradiation properties and performance, both near term ATF design concepts are relatively close to licensing and deployment in several States.

Table 8 provides a qualitative assessment of the applicability of existing analytical models and fuel design criteria and associated regulatory guidance to the advanced nuclear fuels described in Section 4. The relative impacts on applicability are defined as follows:

- None – Existing models, criteria, and guidance remain applicable.
- Low – Existing models, criteria, and guidance are likely applicable. However, this has to be confirmed. Data gaps and model updates, if necessary, are expected to be limited.
- Medium – No new phenomena. Data gaps necessitate additional research, to define new or different performance metrics and analytical limits, and update models, criteria and/or guidance, as necessary. Characterization of irradiated material properties and performance may involve a long lead time.
- High – New phenomena necessitate additional research to fill data gaps, define new or different performance metrics and analytical limits, and generate new or different models, criteria and/or guidance, as necessary. Characterization of irradiated material properties and performance may involve a long lead time.

No impact means that the models, criteria and guidance remain applicable. This does not equate to no influence of the technology on calculated results; rather, that existing models and guidance may be employed. For example, adding Chromia to fuel pellets has a minor impact on calculated core physics parameters; however, the existing libraries have neutron cross-sections for chrome. A low impact means that only a small fraction of the overall model or guidance needs to be modified or that further justification is necessary to confirm the applicability.

TABLE 8. RELATIVE IMPACT OF ADVANCED NUCLEAR FUEL TECHNOLOGIES ON APPLICABILITY OF EXISTING MODELS, DESIGN CRITERIA AND REGULATORY GUIDANCE

| Existing models, criteria and guidance | Sample | Relative impact of advanced nuclear fuel technologies on applicability | | | | |
|---|---|---|---|---|---|---|
| | | Coated zirconium cladding | Doped $UO_2$ fuel | FeCrAl cladding | SiC-SiC cladding | High density fuel ($U_3Si_2$, UN) |
| Core Neutronics and Physics Parameters | | | | | | |
| Cross-section libraries | ENDF/B-VII | Low | Low | Low | Low | Low |
| Lattice physics models | CASMO-4 | Low | Low | Low | Low | Low |
| Reload depletion models | SIMULATE-4 | Low | Low | Low | Low | Low |
| Decay heat models | ANS-5.1-2014 | Low | Low | Low | Low | Medium |
| Fuel Assembly Design Calculations | | | | | | |
| Fuel rod thermomechanical models | Vendor specific PAD5, PRIME, GALILEO | Low | Low | Medium | High | High |
| Dimensional stability | Design specific | Low | Low | Medium | High | High |
| Flow induced vibration, grid-to-rod fretting, debris fretting | Design specific | Low | Low | Low | High | Low |
| Power manoeuvring guidelines | Fuel preconditioning and power ascension limits | Low | Low | Medium | High | High |
| Cladding swelling and rupture | NUREG-0630 | Low | Low | Medium | High | Low |
| Fuel Design and/or Safety Limits | | | | | | |
| Rod internal pressure | No clad lift-off | Low | Low | Medium | High | Low |
| Critical heat flux safety limit | Design specific DNBR/CPR thermal limits | Low | Low | Medium | Medium | Low |
| Transient cladding strain failure threshold | 1.0% permanent diametral | Low | Low | Medium | High | Low |
| Fuel melting point | 2860 °C | Low | Medium | Low | Low | High |
| RIA cladding failure thresholds | RG 1.236 PCMI cladding failure thresholds | Low | Medium | Medium | High | High |
| RIA damaged core coolability criteria | RG 1.236 limit on deposited energy | Low | Medium | Medium | High | High |
| LOCA cladding embrittlement criteria | 10CFR50.46 | Medium | Low | Medium | High | High |
| Seismic design and control rod insertion | Design specific applied load limits | Low | Low | Medium | High | Medium |
| Radiological Consequence Assessments | | | | | | |
| Radionuclide inventories | ORIGEN-S | Low | Low | Low | Low | Medium |
| Core releases, timing, composition | RG 1.183 tables 1 and 2 | Low | Low | Medium | High | High |
| Fuel rod release fractions | RG 1.183 table 3 | Low | Low | Medium | Medium | High |
| Short lived radionuclide release fractions | ANS-5.4-2011 | Low | Low | Low | Low | High |
| Allowable consequences | 10CFR50.67 (25 remTEDE) | None | None | None | None | None |

In 2022, the U.S. NRC completed an assessment of the applicability of existing regulations and guidance for the licensing of near term ATF concepts, higher burnup, and increased $^{235}U$ enrichment fuel designs [58]. The U.S. NRC report [10] documents an assessment of the applicability of regulations within 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, and Part 70, Domestic Licensing of Special Nuclear Material, as well as other regulatory guidance documents (e.g. regulatory guides and NUREG series). The report documents the bases of the applicability determination, a closure plan to fill data gaps, update analytical models and/or guidance, and assigns ownership and priority. Near term ATF concepts included Cr coated zirconium alloy fuel rod cladding and doped $UO_2$ fuel pellets.

For Cr coated cladding, the potential impacts for Cr coated cladding and doped $UO_2$ fuel pellets on continued applicability are either none or low (with the exception of LOCA cladding embrittlement criteria). Cr coated cladding has superior high temperature steam oxidation behaviour relative to standard, uncoated zirconium alloy. In addition, within the rupture region of the fuel rod (if predicted to occur), the uncoated cladding inner diameter surface will oxidize at a different rate than the coated cladding outer diameter. These differences impact the rate of cladding embrittlement and hence impact the applicability of existing safety criteria.

For advanced steel cladding (i.e. FeCrAl), the WGFS technical opinion paper [32] characterized the relative impact on existing regulatory requirements and analytical criteria as medium. Similar to zirconium cladding alloys, FeCrAl has favourable combinations of strength, ductility, and corrosion resistance. And while calibration variables and analytical limits need to be refined to capture FeCrAl's material properties and performance, the architecture of the analytical models do not involve significant changes.

Table 8 provides an assessment of the applicability of existing analytical models, fuel design criteria, and regulatory guidance for FeCrAl cladding. The overall impact is rated as medium. There is almost no impact on analytical models and guidance associated with core physics and radiological consequences, as these items are more closely related to fuel pellet properties than cladding properties.

For SiC-SiC composite cladding and higher density fuel pellets, the WGFS technical opinion paper [32] identified large data gaps, new phenomena, and significant impact on existing fuel design and performance criteria. SiC-SiC composite cladding material offers superior high temperature corrosion resistance and strength. However, many of its performance characteristics, most notably ductility and gas permeability, differ dramatically from metallic alloys. In addition, new phenomena, such as mass loss due to hydrothermal corrosion, may necessitate new analytical models and acceptance and performance criteria. As a result, the overall impact on the applicability of existing analytical models, fuel design criteria, and regulatory guidance is rated as high. With the exception of the timing of releases due to core melting during severe accidents, there is almost no impact on analytical models and guidance associated with core physics and radiological consequences.

Higher density fuel pellets (e.g., $U_3Si_2$, UN) offer increased uranium density and higher thermal conductivity. Unfortunately, the irradiated material properties and performance are not yet fully characterized. Given the large data gaps, a rating of high was assigned to the areas where the applicability of existing analytical models, design criteria, and regulatory guidance is unknown.

## Appendix I

## IAEA REQUIREMENTS AND RECOMMENDATIONS APPLICABLE IN THE DESIGN, QUALIFICATION AND LICENSING OF NUCLEAR FUELS FOR NUCLEAR POWER PLANTS

The IAEA's Statute:

> "authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application" [59].

IAEA Safety Standards Series comprise three categories:

- Safety Fundamentals, which "present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements" [59].
- Safety Requirements, which "establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals" [59].
- Safety Guides, which

> "provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety" [59].

The following subsections introduce general safety requirements (subsection I.1), as well as specific safety requirements (subsection I.2) pertaining to design and operation of NPPs, which are applicable to nuclear fuel qualifications and licensing. Finally, recommendations from IAEA safety guides on how to meet the abovementioned requirements in the design of the reactor core are presented in subsection I.3.

### I.1. GENERAL SAFETY REQUIREMENTS

Among the seven General Safety Requirements to be met to ensure the protection of people and the environment in all facilities and activities, GSR Part 1 (Rev. 1) [4] and GSR Part 4 (Rev. 1) [5] address more specifically the areas of the regulatory framework for safety and of the safety assessment.

GSR Part 1 (Rev. 1) [4] (revised in 2016) establishes "requirements in respect of the governmental, legal and regulatory framework for safety" [4]. This publication "covers the essential aspects of the governmental and legal framework for establishing a regulatory body and for taking other actions necessary to ensure the effective regulatory control of facilities and activities" [4].

The objective of GSR Part 4 (Rev. 1) [5], as stated in para. 1.3, is:

> “to establish the generally applicable requirements to be fulfilled in safety assessment for facilities and activities, with special attention paid to defence in depth, quantitative analyses and the application of a graded approach to the ranges of facilities and of activities that are addressed. [GSR Part 4] also addresses the independent verification of the safety assessment that needs to be carried out by the originators and users of the safety assessment”.

Para. 1.7 of GSR Part 4 (Rev. 1) [5] states that:

> “safety assessment plays an important role throughout the lifetime of the facility or activity whenever decisions on safety issues are made by the designers, the constructors, the manufacturers, the operating organization or the regulatory body. The initial development and use of the safety assessment provides the framework for the acquisition of the necessary information to demonstrate compliance with the relevant safety requirements, and for the development and maintenance of the safety assessment over the lifetime of the facility or activity.”

## I.2. REQUIREMENTS APPLICABLE TO NUCLEAR FUEL DESIGN AND OPERATION

Relevant requirements in SSR-2/1 (Rev.1) [6] and SSR-2/2 (Rev.1) [60] are introduced in subsections I.2.1–I.2.2.

### I.2.1. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design

The following subsections provide first an overview (subsection I.2.1.1), a description of the categories of plant states (subsection I.2.1.2) and an introduction of requirements for the design of the reactor core (subsection I.2.1.3).

#### *I.2.1.1. Overview*

SSR-2/1 (Rev. 1) [6]:

> “establishes design requirements for the structures, systems and components (SSCs) of a nuclear power plant, as well as for procedures and organizational processes important to safety that are required to be met for safe operation and for preventing events that could compromise safety, or for mitigating the consequences of such events, were they to occur”.

SSR-2/1 (Rev. 1) [6] was published in 2016, with strengthened requirements in the following areas:

- “Prevention of severe accidents by strengthening the design basis for the plant;
- Prevention of unacceptable radiological consequences of a severe accident for the public and the environment;
- Mitigation of the consequences of a severe accident to avoid or to minimize radioactive contamination off the site” [6].

The structure of SSR-2/1 (Rev. 1) [6] is as follows:

- Section 2 “elaborates on the safety objective, safety principles and concepts that form the basis for deriving the safety function requirements that must be met for the nuclear power plant, as well as the safety design criteria” [6]. In this section, requirements related to the following topics pertaining to the design of nuclear power plants are introduced:

- Achievement of the safety objective, through the application of the ten safety principles;
- Radiation protection in design;
- Safety in design;
- The concept of defence in depth;
- Maintaining the integrity of design throughout the lifetime of the nuclear power plant;

- Sections 3–6 establish eighty-two overarching requirements, organised in four sections:

  - "Section 3 establishes the general requirements to be satisfied by the design organization in the management of safety in the design process;
  - Section 4 establishes:
    i. Requirements for principal technical design criteria for safety, including requirements for the fundamental safety functions, the application of defence in depth and provision for construction;
    ii. Requirements for interfaces of safety with nuclear security and with the State system of accounting for, and control of, nuclear material;
    iii. Requirements for ensuring that radiation risks arising from the plant are maintained as low as reasonably achievable.
  - Section 5 establishes requirements for general plant design that supplement the requirements for principal technical design criteria to ensure that safety objectives are met and the safety principles are applied. The requirements for general plant design apply to all items (i.e. SSCs) important to safety.
  - Section 6 establishes requirements for the design of specific plant systems such as the reactor core, reactor coolant systems, containment system, and instrumentation and control systems" [6].

*I.2.1.2. Categories of plant states*

Requirement 13 of SSR-2/1 (Rev. 1) [6] states that:

> "**plant states shall be identified and shall be grouped into a limited number of categories primarily on the basis of their frequency of occurrence at the nuclear power plant."**

Paragraphs 5.1 and 5.2 of SSR-2/1 (Rev. 1) [6] state that:

> "plant states shall typically cover:
>
> - Normal operation;
> - Anticipated operational occurrences, which are expected to occur over the operating lifetime of the plant;
> - Design basis accidents;
> - Design extension conditions, including accidents with core melting.

Criteria shall be assigned to each plant state, such that frequently occurring plant states shall have no, or only minor, radiological consequences and plant states that could give rise to serious consequences shall have a very low frequency of occurrence."

As far as plant states considered for the design are considered, the major evolution in SSR-2/1 (Rev. 1) [6] is the consideration of DEC – including accidents with core melting – instead of beyond design basis accidents.

The concept of DEC, and the consideration of such conditions in the design process, is further developed in other requirements of SSR-2/1 (Rev. 1) [6], including Requirements 7, 20 and 33. These requirements clarify that the notion of DEC implies a far reaching extension of the plant states envelope considered for the design, to fully include DEC within the design basis.

Requirement 20 of SSR-2/1 (Rev. 1) [6] states that:

"**a set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences."**

Paragraph 5.28 of SSR-2/1 (Rev. 1) [6] states that:

"the design extension conditions shall be used to define the design specifications for safety features and for the design of all other items important to safety that are necessary for preventing such conditions from arising, or, if they do arise, for controlling them and mitigating their consequences".

*I.2.1.3. Requirements for the design of the reactor core*

Requirements specifically related to the design of nuclear fuel, reactor core and control and shutdown systems are formulated in the following:

- Requirement 43 of SSR-2/1 (Rev. 1) [6] states that:

"**fuel elements and assemblies for the nuclear power plant shall be designed to maintain their structural integrity, and to withstand satisfactorily the anticipated radiation levels and other conditions in the reactor core, in combination with all the processes of deterioration that could occur in operational states."**

Paragraph 6.1 of SSR-2/1 (Rev. 1) [6] states that:

"the processes of deterioration to be considered shall include those arising from:

- Differential expansion and deformation;
- External pressure of the coolant;
- Additional internal pressure due to fission products and the build-up of helium in fuel elements,
- Irradiation of fuel and other materials in the fuel assembly;

- Variations in pressure and temperature resulting from variations in power demand;
- Chemical effects;
- Static and dynamic loading, including flow induced vibrations and mechanical vibrations;
- Variations in performance in relation to heat transfer that could result from distortion or chemical effects."

- Requirement 44 of SSR-2/1 (Rev. 1) [6] states that:

  "**the fuel elements and fuel assemblies and their supporting structures for the nuclear power plant shall be designed so that, in operational states and in accident conditions other than severe accidents, a geometry that allows for adequate cooling is maintained and the insertion of control rods is not impeded.**"

- Requirement 45 of SSR-2/1 (Rev. 1) [6] states that:

  "**distributions of neutron flux that can arise in any state of the reactor core in the nuclear power plant, including states arising after shutdown and during or after refuelling, and states arising from anticipated operational occurrences and from accident conditions not involving degradation of the reactor core, shall be inherently stable. The demands made on the control system for maintaining the shapes, levels and stability of the neutron flux within specified design limits in all operational states shall be minimized.**"

- Paragraphs 6.4, 6.5 and 6.6 of SSR-2/1 (Rev. 1) [6] state that:

  "adequate means of detecting the neutron flux distributions in the reactor core and their changes shall be provided for the purpose of ensuring that there are no regions of the core in which the design limits could be exceeded.

  In the design of reactivity control devices, due account shall be taken of wear out and of the effects of irradiation, such as burnup, changes in physical properties and production of gas.

  The maximum degree of positive reactivity and its rate of increase by insertion in operational states and accident conditions not involving degradation of the reactor core shall be limited or compensated for, to prevent any resultant failure of the pressure boundary of the reactor coolant systems, to maintain the capability for cooling and to prevent any significant damage to the reactor core."

- Requirement 46 of SSR-2/1 (Rev. 1) [6] states that:

  "**means shall be provided to ensure that there is a capability to shut down the reactor of the nuclear power plant in operational states and in accident conditions, and that the shutdown condition can be maintained even for the most reactive conditions of the reactor core.**"

Additional requirements are established in paras 6.7–6.12 of SSR-2/1 (Rev. 1) [6], including on diversity and independence of at least two systems. Paragraphs 6.9 and 6.10 of SSR-2/1 (Rev. 1) [6] state that:

“the means for shutting down the reactor shall consist of at least two diverse and independent systems.

At least one of the two different shutdown systems shall be capable, on its own, of maintaining the reactor subcritical by an adequate margin and with high reliability, even for the most reactive conditions of the reactor core.”

Other requirements in SSR-2/1 (Rev. 1) [6] — applicable to the design of various SSCs — are also relevant to the design of the reactor core, including the following:

- Requirement 51 of SSR-2/1 (Rev. 1) [6] states that:

  “**means shall be provided for the removal of residual heat from the reactor core in the shutdown state of the nuclear power plant such that the design limits for fuel, the reactor coolant pressure boundary and structures important to safety are not exceeded.”**

- Requirement 52 of SSR-2/1 (Rev. 1) [6] states that:

  “**means of cooling the reactor core shall be provided to restore and maintain cooling of the fuel under accident conditions at the nuclear power plant, even if the integrity of the pressure boundary of the primary coolant system is not maintained.”**

- Paragraph 6.18 of SSR-2/1 (Rev. 1) [6] states that:

  “the means provided for cooling of the reactor core shall be such as to ensure that:

  (a) The limiting parameters for the cladding or for integrity of the fuel (such as temperature) will not be exceeded;
  (b) Possible chemical reactions are kept to an acceptable level;
  (c) The effectiveness of the means of cooling of the reactor core compensates for possible changes in the fuel and in the internal geometry of the reactor core;
  (d) Cooling of the reactor core will be ensured for a sufficient time.”

- Requirement 53 of SSR-2/1 (Rev. 1) [6] states that “the capability to transfer heat to an ultimate heat sink shall be ensured for all plant states”.

- Paragraphs 6.19A and 6.19B of SSR-2/1 (Rev. 1) [6] state that:

  “systems for transferring heat shall have adequate reliability for the plant states in which they have to fulfil the heat transfer function. This may require the use of a different ultimate heat sink or different access to the ultimate heat sink.

  The heat transfer function shall be fulfilled for levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site.”

- Requirement 61 of SSR-2/1 (Rev. 1) [6] states that:

  “**a protection system shall be provided at the nuclear power plant that has the capability to detect unsafe plant conditions and to initiate safety actions**

**automatically to actuate the safety systems necessary for achieving and maintaining safe plant conditions."**

- Requirement 80 of SSR-2/1 (Rev. 1) [6] states that:

  "**fuel handling and storage systems shall be provided at the nuclear power plant to ensure that the integrity and properties of the fuel are maintained at all times during fuel handling and storage."**

The following requirements in SSR-2/1 (Rev. 1) [6] are also relevant to the design of the reactor core:

- Fundamental safety function (Requirement 4);
- Proven engineering practices (Requirement 9);
- Safety assessment (Requirement 10);
- Design limits (Requirement 15) and operational limits and conditions for safe operation (Requirement 28);
- Postulated initiating events (Requirement 16);
- Internal and external hazards (Requirement 17);
- Engineering design rules (Requirement 18);
- Safety classification (Requirement 22);
- Reliability of items important to safety (Requirement 23);
- Containment system for the reactor (Requirement 54).

Recommendations on meeting the requirements for the design of the reactor core are provided in SSG-52 [3] and are presented in subsection I.3.1.

*I.2.1.4. Applicability of SSR-2/1 (Rev. 1)*

The scope of applicability of SSR-2/1 (Rev. 1) [6] is as follows:

- Applicability to reactor technologies: para. 1.6 of SSR-2/1 (Rev. 1) [6] defines the scope as follows:

  "it is expected that this publication will be used primarily for land based stationary nuclear power plants with water cooled reactors designed for electricity generation or for other heat production applications (such as district heating or desalination). This publication may also be applied, with judgement, to other reactor types, to determine the requirements that have to be considered in developing the design".

- Applicability to operating nuclear power plants: para. 1.3 of SSR-2/1 (Rev. 1) [6] states that:

  "it might not be practicable to apply all the requirements of this publication to nuclear power plants that are already in operation or under construction. In addition, it might not be feasible to modify designs that have already been approved by regulatory bodies. For the safety analysis of such designs, it is expected that a comparison will be made with the current standards, for example as part of the periodic safety review for the plant, to determine whether the safe operation of the plant could be further enhanced by means of reasonably practicable safety improvements".

Consistent with the above scope, the advanced nuclear fuels considered in this publication are those in use in water cooled reactors (see subsection I.3.1 for more details on the recommendations contained in SSG-52 [3]).

**I.2.2. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation**

The following subsections provide an overview (subsection I.2.2.1) and an introduction of requirements relevant to the design of the reactor core and the nuclear fuel (subsection I.2.2.2).

*I.2.2.1. Overview*

As stated in para. 1.5 of SSR-2/2 (Rev. 1) [60] (citation omitted), the objective is:

> "to establish the requirements which, in the light of experience and the present state of technology, must be satisfied to ensure the safe commissioning and operation of nuclear power plants. These requirements are governed by the safety objective and safety principles that are established in the Fundamental Safety Principles."

Paragraph 1.8 of SSR-2/2 (Rev. 1) [60] states that "the requirements are mainly applicable to water cooled reactors, but they may also be used as a basis for establishing specific requirements for other reactor designs."

*I.2.2.2. Requirements relevant for the design of the reactor core and the nuclear fuel*

- Requirement 6 of SSR-2/2 (Rev. 1) [60] states that: "the operating organization shall ensure that the plant is operated in accordance with the set of operational limits and conditions."

  Further details on the definition, revision, and training of personnel related to operational limits and conditions are given in paras 4.6–4.15 of SSR-2/2 (Rev. 1) [60], which also establish requirements related to processes that apply to possible deviations from these.

- Requirement 11 of SSR-2/2 (Rev. 1) [60] states that: "the operating organization shall establish and implement a programme to manage modifications".

  Further requirements are provided in paras 4.39 and 4.41 of SSR-2/2 (Rev. 1) [60], which are also relevant for the loading of LTR and LTA, and which state that:

  > "a modification programme shall be established and implemented to ensure that all modifications are properly identified, specified, screened, designed, evaluated, authorized, implemented and recorded. Modification programmes shall cover: structures, systems and components; operational limits and conditions; procedures; documents; and the structure of the operating organization. Modifications shall be characterized on the basis of their safety significance. Modifications shall be subject to the approval of the regulatory body, in accordance with their safety significance, and in line with national arrangements.
  >
  > Temporary modifications shall be limited in time and number to minimize the cumulative safety significance. Temporary modifications shall be clearly identified at their location and at any relevant control position. The operating organization shall establish a formal system for informing relevant personnel in good time of temporary modifications and of their consequences for the operation and safety of the plant".

- Requirement 13 of SSR-2/2 (Rev. 1) [60] states that:

  “**the operating organization shall ensure that a systematic assessment is carried out to provide reliable confirmation that safety related items are capable of the required performance for all operational states and for accident conditions.”**

- Requirement 30 of SSR-2/2 (Rev. 1) [60] states that: “the operating organization shall be responsible and shall make arrangements for all activities associated with core management and with on-site fuel handling”.

- Paragraph 7.18 of SSR-2/2 (Rev. 1) [60] states that:

  “provision shall be made to ensure that only fuel that has been appropriately manufactured is loaded into the core. In addition, the fuel design criteria and fuel enrichment shall be in accordance with design specifications and shall be subject to approval by the regulatory body as required. The same requirements shall be applied before the introduction of fuel of a new design or of a modified design into the core.”

  Further requirements are provided in paras 7.19–7.29 of SSR-2/2 (Rev. 1) [60].

- Requirement 31 of SSR-2/2 (Rev. 1) [60] states that: “the operating organization shall ensure that effective programmes for maintenance, testing, surveillance and inspection are established and implemented.”

  Further requirements are provided in paras 8.1–8.17 of SSR-2/2 (Rev. 1) [60].

## I.3. RECOMMENDATIONS PROVIDED IN SAFETY GUIDES

### I.3.1. SSG-52, Design of the Reactor Core for Nuclear Power Plants

The following subsections provide an overview (subsection I.3.1.1), a description of plant states considered for the design of the reactor core (subsection I.3.1.2), consider the applicability of SSG-52 (subsection I.3.1.3), and describe recommendations related to the qualification and testing of prototype assemblies and lead use assemblies (subsection I.3.1.4).

#### *I.3.1.1. Overview*

SSG-52 [3] provides recommendations on meeting the safety requirements established in SSR-2/1 (Rev.1) [6] for the design of the reactor core for nuclear power plants.

SSG-52 [3] adopts the following structure:

- Section 2 “describes general considerations for safe core design based on requirements for the management of safety, principal technical requirements and general design requirements established in Sections 3, 4 and 5 of SSR-2/1 (Rev. 1), respectively” [3].
- Section 3 “describes specific considerations for the safe design of fuel rods, fuel assemblies, core structures and core components, and the core control system and the reactor shutdown system based on specific design requirements” [3] (i.e. requirements 43–46, 51–53, 61, 80 of SSR-2/1 (Rev. 1), see subsection I.2.1.3).
- Section 4 provides recommendations on the qualification and testing of the SSCs of the reactor core.

*I.3.1.2. Plant states considered for the design of the reactor core*

Paragraph 2.10 of SSG-52 [3] states that:

> "the plant states typically considered for the design of the reactor core are normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation. These four states are referred to as 'all applicable plant states' throughout [SSG-52]. Accidents with significant core melting are outside the scope of the design of the reactor core."

*I.3.1.3. Applicability of SSG-52*

The scope of applicability is as follows:

- Applicability to reactor technologies: the applicability of SSG-52 [3] is derived from SSR-2/1 (Rev. 1) [6] (see subsection I.2.1.4):
- Applicability to the design of SSCs: para. 1.4 of SSG-52 [3] states that:

> "specifically, the following structures, systems and components (SSCs) are covered:
>
> (a) Fuel rods, containing fuel pellets with or without burnable absorbers in cladding tubes, which generate and transfer heat to the coolant.
> (b) Fuel assemblies, comprising bundles of fuel rods, along with structures and components (e.g. guide tubes, spacer grids, bottom and top nozzles, fuel channels) that maintain the fuel rods and fuel assemblies in a predetermined geometrical configuration.
> (c) The reactor core control system, the shutdown system and the monitoring system, including components and equipment used for reactivity control and shutdown, comprising neutron absorbers (solid or liquid), the associated structure and the drive mechanism.
> (d) Support structures that provide the foundation for the core within the reactor vessel (within the calandria for pressurized heavy water reactors), the structure for guiding the flow (for pressurized water reactors) and the guide tubes for reactivity control devices (for pressurized heavy water reactors).
> (e) The coolant.
> (f) The moderator.
> (g) Other core components, such as steam separators (for boiling water reactors) and neutron sources, are considered only to a limited extent in [SSG-52]".

In terms of applicability to fuel/cladding systems, paras 1.5 and 1.6 of SSG-52 [3] state that:

> "[SSG-52] is intended mainly for NPPs that use natural and enriched $UO_2$ fuels and plutonium-blended $UO_2$ fuel (mixed oxide fuel) with zirconium based alloy cladding. Unless otherwise specified, all recommendations apply to these fuel types.
>
> For innovative fuel materials, such as uranium nitride fuel or inert matrix fuel, or cladding materials other than zirconium based alloys, [SSG-52] can be applied with judgement."

Section 5 of this TECDOC presents the results of an evaluation of the applicability of the recommendations provided in SSG-52 [3] in addressing some advanced nuclear fuel currently under development in Member States (as described in Section 4).

*I.3.1.4. Qualification and testing*

Section 4 of SSG-52 [3] provides recommendations on the development of a "robust programme for qualification, inspection and testing of the equipment design and analysis process" (para. 1.4 of SSG-52 [3]).

Paragraph 4.4 of SSG-52 [3] states that:

> "methods of qualification should include:
>
> (a) Performance of a type test on the SSCs representative of those to be supplied;
> (b) Performance of a test on the SSCs supplied;
> (c) Use of pertinent past experience;
> (d) Analysis based on available and applicable test data;
> (e) Any combination of the above methods."

With regard to the testing of prototype assemblies and lead use assemblies, para. 4.8 of SSG-52 [3] states that:

> "out-of-reactor tests should be performed on fuel assembly prototypes, when practical, to determine the characteristics of a new design. The following out-of-reactor tests are generally performed for this purpose:
>
> (a) For light water reactors:
>   (i) Spacer grid tests (including pressure drop tests, crush strength tests and other structural tests such as seismic resistance tests);
>   (ii) Control rod structural and performance tests;
>   (iii) Fuel assembly structural tests (lateral, axial and torsional stiffness, and frequency and damping);
>   (iv) Fuel assembly hydraulic flow tests, including the determination of pressure drop and fuel assembly lift-off force, control rod vibration and wear, fuel assembly vibration, grid-to-rod fretting (with account taken of the relaxation of spacer grid springs), and evaluations of the wear and the lifetime of fuel assemblies;
>   (v) Fuel assembly thermohydraulic tests, including the determination of critical heat flux correlations.
>
> (b) For pressurized heavy water reactors:
>   (i) Fuel bundle string pressure drop tests;
>   (ii) Cross-flow endurance tests;
>   (iii) Mechanical endurance tests;
>   (iv) Bundle impact tests;
>   (v) Bundle strength tests;
>   (vi) Wear tests;
>   (vii) Seismic qualification tests;
>   (viii) Wash-in and wash-out tests (where applicable);
>   (ix) Critical heat flux tests."

In terms of in-reactor testing, para. 4.9 of SSG-52 [3] states that:

> "in-reactor testing of design features through irradiation of fuel rods or fuel assemblies in materials test reactors or through irradiation of lead use fuel assemblies in power reactors

should be used to justify the specified maximum burnup limit for a new design. The following phenomena may be tested in this manner:

(a) Fuel and burnable absorber rod growth;
(b) Fuel rod bowing;
(c) Fuel rod, spacer grid and fuel channel (if present) oxidation and hydride levels;
(d) Fuel rod fretting and spacer (for pressurized heavy water reactors) fretting;
(e) Fuel assembly growth;
(f) Fuel assembly bowing;
(g) Fuel channel (for boiling water reactors) wear and distortion;
(h) Fuel rod ridging (i.e. pellet–cladding interaction);
(i) Fuel rod integrity;
(j) Hold-down spring relaxation (for pressurized water reactors);
(k) Spacer grid spring relaxation (for light water reactors);
(l) Control rod and guide tube wear (for pressurized water reactors).”

Paragraph 4.10 of SSG-52 [3] states that:

“in cases where in-reactor testing of a new fuel assembly design or a new design feature cannot be performed, special attention should be given to analytical evaluations and to augmented inspection or surveillance plans to validate the fuel design capability and performance features.”

### I.3.2. SSG-2 (Rev 1), Deterministic Safety Analysis for Nuclear Power Plants

The following subsections provide an overview (subsection I.3.2.1) as well as a description of available options for performing deterministic safety analysis (subsection I.3.2.2).

#### *I.3.2.1. Overview*

As stated in para. 1.4 of SSG-2 (Rev. 1) [7] (citations omitted), the objective is:

“to provide recommendations and guidance for designers, operating organizations, regulatory bodies and technical support organizations on performing deterministic safety analysis and on its application to nuclear power plants. [SSG-2 (Rev. 1)] also provides recommendations on the use of deterministic safety analysis in:

(a) Demonstrating or assessing compliance with regulatory requirements;
(b) Identifying possible enhancements of safety and reliability”.

The recommendations are provided to meet the applicable safety requirements established in SSR-2/1 (Rev. 1) and GSR Part 4 (Rev. 1), and are supported by current practices and experience from deterministic safety analysis being performed for nuclear power plants around the world.

Paragraph 1.6 of SSG-2 (Rev. 1) [7] (citations omitted) states that:

“[SSG-2 (Rev. 1)] focuses primarily on deterministic safety analysis for the safety of designs for new nuclear power plants and, as far as reasonably practicable or achievable, is also applicable to the safety re-evaluation or reassessment of existing nuclear power plants when operating organizations review their safety assessment. The

> recommendations provided are intended to be consistent with the scope of applicability indicated in paras 1.3 and 1.6 of SSR-2/1 (Rev. 1) and are particularly based on experience with deterministic safety analysis for water cooled reactors."

Paragraph 1.11 of SSG-2 (Rev. 1) [7] states that:

> "[SSG-2 (Rev. 1)] focuses on neutronic, thermohydraulic, fuel (or fuel channel for pressurized heavy water reactors) and radiological analysis. Other types of analysis, in particular structural analysis of structures and components, are also important means of demonstrating the safety of a plant. However, detailed guidance on performing such analysis is not included since such information can be found in specific engineering guides. Neutronic and thermohydraulic analysis provides the necessary boundary conditions for structural analysis."

The structure of SSG-2 (Rev. 1) [7] is as follows:

- Section 2 introduces some basic concepts and terminology used in deterministic safety analysis, as a basis for the specific recommendations provided in the other sections.
- Section 3 describes methods of systematic identification, categorization and grouping of postulated initiating events and accident scenarios to be addressed by deterministic safety analysis, and includes practical advice on the selection of events to be analysed for the different plant states.
- Section 4 provides a general overview of acceptance criteria to be used in deterministic safety analysis for design and authorization of nuclear power plants, and describes the rules for determination and use of acceptance criteria.
- Section 5 provides guidance on verification and validation, selection and use of computer codes and plant models, together with input data used in the computer codes.
- Section 6 describes general approaches for ensuring adequate safety margins in demonstrating compliance with acceptance criteria for all plant states, with a focus on AOO and DBA.
- Section 7 provides specific guidance on performing deterministic safety analysis for each individual plant state.
- Section 8 includes guidance on the documentation, review and updating of deterministic safety analysis.
- Section 9 provides guidance on independent verification of safety assessments, including verification of deterministic safety analysis.

*I.3.2.2. Options for performing deterministic safety analysis*

Recommendations on the possible approaches to deterministic safety analysis are provided in paras 2.8–2.15 of SSG-2 (Rev. 1) [7], and summarized in Table 9 (reproduced from SSG-2 (Rev. 1) [7]), with different levels of conservatism depending on the combination of computer code type (conservative or best estimate), assumptions on the availability of systems and the type of initial and boundary conditions.

Section 7 of SSG-2 (Rev. 1) [7] provides recommendations on the rules and methods for performing deterministic safety analysis for different plant states (normal operation, AOO, DBA, DEC without significant core degradation, DEC with core melting), focusing on the specific objectives of each type of analysis, acceptance criteria, assumptions on the availability of systems and operator actions, analysis assumptions and the treatment of uncertainties.

TABLE 9. OPTIONS FOR PERFORMING DETERMINISTIC SAFETY ANALYSIS [7]

| Option | Computer code type | Assumptions about systems availability | Type of initial and boundary conditions |
|---|---|---|---|
| 1. Conservative | Conservative | Conservative | Conservative |
| 2. Combined | Best estimate | Conservative | Conservative |
| 3. Best estimate plus uncertainty | Best estimate | Conservative | Best estimate<br>Partly most unfavourable conditions |
| 4. Realistic* | Best estimate | Best estimate | Best estimate |

* For simplicity, the terms ‘realistic approach’ or ‘realistic analysis’ are used to mean best estimate analysis without quantification of uncertainties.

AOO and DBA are analysed with conservative rules and methods, with the aim “to demonstrate with a high level of confidence that there are significant margins to the safety limits” (para. 7.40 of SSG-2 (Rev. 1)[7]).

Paragraph 7.54 of SSG-2 (Rev. 1) [7] states that:

> “for design extension conditions without significant fuel degradation, in principle the combined approach or the best estimate approach with quantification of uncertainties (best estimate plus uncertainty), as applicable for design basis accidents, may be used. However, in line with the general rules for analysis of design extension conditions, best estimate analysis without a quantification of uncertainties may also be used, subject to consideration of the caveats and conditions indicated in paras 7.55 and 7.67 [of SSG-2 (Rev. 1)].”

### I.3.3. SSG-12, Licensing Process for Nuclear Installations

SSG-12 [8] provides recommendations on meeting the requirements relating to authorization by the regulatory body (Requirements 7, 23 and 24) established in GSR Part 1 (Rev. 1) [4].

Paragraph 1.5 of SSG-12 [8] states that:

> “[SSG-12] describes how the licensing process should be applied at the various stages of the lifetime of a nuclear installation, with discussion of the topics and required documents to be considered at each stage (siting and site evaluation, design, construction, commissioning, operation, decommissioning and release from regulatory control). Some of these stages may be grouped together, depending on national regulations.”

SSG-12 [8] has the following structure:

- Section 2 provides general recommendations on the licensing process, including basic licensing principles, the content of a licence, public participation, and the roles and responsibilities of the regulatory body, applicant and licensee;
- Section 3 contains recommendations specific to the various steps of the licensing process.

### I.3.4. Other Safety Guides

The following Safety Guides contain additional recommendations or constitute useful references for the topics discussed in this publication:

- IAEA Safety Standards Series No. GSG-13, Functions and Processes of the Regulatory Body for Safety [61];
- IAEA Safety Standards Series No. SSG-63, Design of Fuel Handling and Storage Systems for Nuclear Power Plants [62];
- IAEA Safety Standards Series No. SSG-73, Core Management and Fuel Handling for Nuclear Power Plants [63].

# Appendix II

## APPLICABILITY OF SSG-52 IN ADDRESSING ADVANCED NUCLEAR FUELS

This Appendix presents the detailed evaluation of the applicability of recommendations of IAEA SSG-52 [3] to the advanced nuclear fuel technologies described in Section 4.

## II.1. COATED ZIRCONIUM ALLOYS

The results of the detailed evaluation of the applicability of recommendations of SSG-52 [3] to coated zirconium alloys are provided in Table 10.

TABLE 10. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO COATED ZIRCONIUM ALLOYS

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to coated zirconium alloys | Suggested modification to SSG-52 [3] to enhance applicability to coated zirconium alloys |
|---|---|---|---|
| 1.5 | "This Safety Guide is intended mainly for NPPs that use natural and enriched $UO_2$ fuels and plutonium-blended $UO_2$ fuel (mixed oxide fuel) with zirconium alloy cladding. Unless otherwise specified, all recommendations apply to these fuel types" [3]. | Applicable with minor modification. | To add '(including advanced technology fuels, such as Cr coated zirconium cladding)'. |
| 2.26 | "Safety analysis for the reactor core should be performed to verify that fuel design limits are not exceeded in all applicable plant states. For accident conditions, the effect of fuel behaviour on core cooling should be included in the safety analysis (e.g. ballooning and rupture of the cladding, exothermic metal–water reactions, distortions of fuel rods and fuel assemblies). The effects of hydrogen accumulation (as a result of a metal–water reaction between the zirconium alloy cladding and water at high temperature) on the boundary of the reactor coolant system should be evaluated" [3]. | Applicable with due consideration of the difference in material properties (mechanical and thermal properties, the reaction with water, etc.). | The new degradation phenomena identified for coated zirconium cladding need to be also considered (e.g., eutectic formation, diffusion of Cr into the zirconium alloy, coating integrity). |
| 3.22 | "The maximum reactivity worth of the reactivity control devices (e.g. control rods and/or chemical and volume control systems) should be limited, or interlock systems should be provided, so that any resultant power variations do not exceed specified limits for relevant reactivity insertion transients and accidents, such as the following:<br>(a) Control rod ejection;<br>(b) Control rod Ip;<br>(c) Boron dilution;<br>(d) Uncontrolled withdrawal of control banks.<br>Such reactivity limits should be determined via safety analyses to ensure that the fuel design limits described in paras 3.65–3.76 are not exceeded. These analyses should be performed for all fuel types in the core (e.g. $UO_2$ or mixed oxide fuel) or a representative core with appropriate margins, and for all allowable operating conditions and fuel burnup values" [3]. | Applicable as far as the fuel design limits described in paras 3.65–3.76 are not exceeded, considering the difference in cladding material. | To add 'these analyses should be performed for all fuel types in the core (e.g. zirconium cladding or Cr coated zirconium cladding, or $UO_2$ or mixed oxide fuel)'. |

TABLE 10. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO COATED ZIRCONIUM ALLOYS (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to coated zirconium alloys | Suggested modification to SSG-52 [3] to enhance applicability to coated zirconium alloys |
|---|---|---|---|
| 3.30 | “The design should ensure that the minimum ratio of operating power to critical power (i.e. a minimum critical heat flux ratio, a minimum departure from the nucleate boiling ratio, a minimum critical channel power ratio or a minimum critical power ratio) takes into account that critical heat flux correlations have been developed from representative tests performed at steady state conditions” [3]. | Applicable with due consideration of the difference in cladding material and surface conditions. | Fuel assembly CHF tests need to be performed with Cr coated zirconium cladding. |
| 3.40 | “In operational states, the design should ensure that the peak fuel temperature is lower than the fuel melting temperature by an adequate margin to prevent melting of the fuel, when appropriate provisions and uncertainties are considered. For design basis accidents (e.g. reactivity initiated accidents) and for design extension conditions without significant fuel degradation, incipient fuel melting can be allowed (e.g. fuel centreline melting limited to a small fraction of fuel pellet volume). The design and safety assessments should take into account the effects of fuel burnup on the fuel rod and fuel assembly properties (see Annex I for supplementary information)” [3]. | Applicable with due consideration of the difference in cladding material. | In operational states, limits on the peak fuel temperature are established to prevent fuel melt, the applicability of limits established for uncoated zirconium alloy to coated claddings needs to be confirmed. In particular, the possibility of formation of eutectic needs to be addressed. |
| 3.41 | “Straining of the cladding is caused by overpressure of internal gases in the fuel rod or by gaseous swelling of the fuel or thermal expansion of the fuel as a consequence of fuel burnup or local power increases. The design should ensure that cladding stresses and strains are limited. Limits for cladding stress, accumulated cladding strain, and cladding corrosion and hydriding should be specified for all applicable plant states and should be applied throughout the reloading cycle” [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | Some phenomena (relating to cladding corrosion and hydriding) need to be considered, in addition to considering the effect of the difference of material properties. |
| 3.51 | “Hydrogen pick-up correlations should be determined as a function of the corrosion of the cladding in normal operation for each cladding type, so that appropriate fuel design limits, such as for reactivity initiated accidents and loss of coolant accidents, can be expressed as a function of the pre-transient hydrogen content of the cladding (see Annex I for supplementary information)” [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | Same as para. 3.41. |

TABLE 10. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO COATED ZIRCONIUM ALLOYS (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to coated zirconium alloys | Suggested modification to SSG-52 [3] to enhance applicability to coated zirconium alloys |
|---|---|---|---|
| 3.54 | "The design analyses should take into account the degradation of the heat transfer from fuel rods due to the formation of deposits on the surface of the cladding via corrosion products coming from the reactor coolant system or other chemical changes. For pressurized water reactors, in the event that boron is trapped in the crud layer, its potential impact on the neutronic performance of the core should be assessed and addressed in the core design analyses" [3]. | Applicable with due consideration of the difference in the formation of deposits on cladding surface. | The impact of coating on the CRUD formation and effects on heat transfer need to be studied or confirmed. |
| 3.60 | "In design basis accidents that lead to rapid power transients (e.g. a reactivity initiated accident), the fuel cladding can fail due to excessive pellet–cladding mechanical interaction combined with cladding embrittlement due to in-reactor hydriding at high burnup levels. Fuel failures corresponding to this failure mode should be considered in safety analysis" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | Cladding embrittlement due to in-reactor hydriding is material dependent. The effect of the difference of material needs to be considered. |
| 3.63 | "The power-ramp failure threshold should be established, if applicable, in test reactors by means of power-ramp tests for each type of fuel or cladding. The data collected should cover the entire burnup range (see Annex I for supplementary information)" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The PCI-SCC phenomena in presence of coating needs to be studied with sufficient ramp tests. |

TABLE 10. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO COATED ZIRCONIUM ALLOYS (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to coated zirconium alloys | Suggested modification to SSG-52 [3] to enhance applicability to coated zirconium alloys |
|---|---|---|---|
| 3.73 | “To ensure that the structural integrity of the fuel rods is preserved, the following design limits should be defined and justified:<br>(a) The peak cladding temperature in accident conditions should not exceed a level at which cladding oxidation causes excessive cladding embrittlement or accelerates uncontrollably. In addition, for light water reactors, effects on the peak cladding temperature due to fuel fragmentation and its axial relocation within the ballooned area of the fuel rod should be assessed as appropriate. Possible effects of the dispersal of fuel particles on doses to workers and on core coolability should also be addressed.<br>(b) The total oxidation of the cladding should remain below limits such that the cladding can still withstand accident induced loadings (e.g. in the quenching phase of a loss of coolant accident). Such limits should be determined by experiments that take into account pre-transient in-reactor cladding oxidation and transient oxidation (outer side oxidation and possibly inner side oxidation), pre-transient and transient hydrogen absorption, as well as chemical interactions between the fuel pellets and cladding material.<br>(c) The allowable enthalpy rise for reactivity initiated accidents should be limited to values that take into account initial fuel rod conditions (e.g. pre-transient hydrogen content of the cladding and fuel burnup).<br>(d) If applicable, fuel centreline melting should be limited to a small fraction of fuel pellet volume.<br>(e) Fuel rods should be designed to withstand loadings resulting from post-transient fuel assembly handling, storage and transport to a reprocessing or disposal facility” [3]. | Applicable with due consideration of the difference in material properties (e.g. mechanical properties at high temperature). | In terms of (a)–(c) and (e), some design limits need to be changed considering the properties of cladding material and their modification with burnup (e.g., embrittlement with hydrogen pickup, cladding corrosion and hydriding, enthalpy rise during RIA). |

TABLE 10. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO COATED ZIRCONIUM ALLOYS (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to coated zirconium alloys | Suggested modification to SSG-52 [3] to enhance applicability to coated zirconium alloys |
|---|---|---|---|
| 4.9 | "In-reactor testing of design features through irradiation of fuel rods or fuel assemblies in materials test reactors or through irradiation of lead use fuel assemblies in power reactors should be used to justify the specified maximum burnup limit for a new design. The following phenomena may be tested in this manner:<br>(a) Fuel and burnable absorber rod growth;<br>(b) Fuel rod bowing;<br>(c) Fuel rod, spacer grid and fuel channel (if present) oxidation and hydride levels;<br>(d) Fuel rod fretting and spacer (for pressurized heavy water reactors) fretting;<br>(e) Fuel assembly growth;<br>(f) Fuel assembly bowing;<br>(g) Fuel channel (for boiling water reactors) wear and distortion;<br>(h) Fuel rod ridging (i.e. pellet–cladding interaction);<br>(i) Fuel rod integrity;<br>(j) Hold-down spring relaxation (for pressurized water reactors);<br>(k) Spacer grid spring relaxation (for light water reactors);<br>(l) Control rod and guide tube wear (for pressurized water reactors)" [3]. | Applicable. | After the safety analysis for new cladding material is conducted, and the integrity of fuel rod (and assembly) is confirmed, in-reactor testing needs to be conducted.<br>The data and knowledge of irradiation behaviour of new cladding material need to be obtained in order to conduct a safety analysis. |

## II.2. ADVANCED STEELS (FeCrAl)

The results of the detailed evaluation of the applicability of recommendations of SSG-52 [3] to advanced steels (FeCrAl) are provided in Table 11.

TABLE 11. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO ADVANCED STEELS (FeCrAl)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to FeCrAl | Suggested modification to SSG-52 [3] to enhance applicability to FeCrAl |
|---|---|---|---|
| 1.6 | “For innovative fuel materials, such as uranium nitride fuel or inert matrix fuel, or cladding materials other than zirconium alloys, this Safety Guide can be applied with judgement” [3]. | Applicable with judgement. | To add ‘cladding materials such as advanced steels (e.g. FeCrAl)’ |
| 2.26 | “Safety analysis for the reactor core should be performed to verify that fuel design limits are not exceeded in all applicable plant states. For accident conditions, the effect of fuel behaviour on core cooling should be included in the safety analysis (e.g. ballooning and rupture of the cladding, exothermic metal–water reactions, distortions of fuel rods and fuel assemblies). The effects of hydrogen accumulation (as a result of a metal–water reaction between the zirconium alloy cladding and water at high temperature) on the boundary of the reactor coolant system should be evaluated” [3]. | Applicable with due consideration of the difference in cladding material properties (mechanical and thermal properties, the reaction with water, etc.). | New phenomena specific to FeCrAl need to be considered and added appropriately, e.g. the cladding embrittlement at low temperature (e.g. temperature in nominal operation) which is induced by Cr rich α’ (alpha prime) precipitation, the thinning of cladding wall thickness by a competing reaction of formation of oxide layer and dissolution into coolant, etc. |
| 3.5 | “Cladding materials should be selected with consideration of the following properties (examples of cladding materials are provided in Annex I):<br>(a) Low absorption cross-section for thermal neutrons;<br>(b) High resistance to irradiation conditions;<br>(c) High thermal conductivity and high melting point;<br>(d) High corrosion resistance and low hydrogen pick-up;<br>(e) Low oxidation and low hydriding in high temperature conditions;<br>(f) Adequate resistance to breakaway oxidation at high integrated-time temperature conditions;<br>(g) Adequate mechanical properties (e.g. high strength, high ductility, low creep rate in normal operation, high relaxation rate in transients);<br>(h) Low susceptibility to stress corrosion cracking;<br>(i) Adequate resistance to hydrogen assisted cracking and hydride related cracking in normal operation and for fuel storage” [3]. | Applicable with due consideration of the difference in cladding material properties (mechanical and thermal properties, nuclear properties, the reaction with water, etc.), including their burnup effect. | Some phenomena (mainly relating to (b), (d)–(f), (h) and (i)) are material dependent. The effect of the difference of material needs to be considered and confirmed.<br>In particular, the susceptibility of FeCrAl material to stress corrosion cracking needs to be considered and confirmed.<br>Also the susceptibility of FeCrAl cladding to the embrittlement at low temperature (e.g. normal operation temperature) by Cr rich α’ (alpha prime) precipitation within the cladding needs to be included. |

TABLE 11. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO ADVANCED STEELS (FeCrAl) (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to FeCrAl | Suggested modification to SSG-52 [3] to enhance applicability to FeCrAl |
|---|---|---|---|
| 3.22 | "The maximum reactivity worth of the reactivity control devices (e.g. control rods and/or chemical and volume control systems) should be limited, or interlock systems should be provided, so that any resultant power variations do not exceed specified limits for relevant reactivity insertion transients and accidents, such as the following:<br>(a) Control rod ejection;<br>(b) Control rod drop;<br>(c) Boron dilution;<br>(d) Uncontrolled withdrawal of control banks.<br>Such reactivity limits should be determined via safety analyses to ensure that the fuel design limits described in paras 3.65–3.76 are not exceeded. These analyses should be performed for all fuel types in the core (e.g. $UO_2$ or mixed oxide fuel) or a representative core with appropriate margins, and for all allowable operating conditions and fuel burnup values" [3]. | Applicable as far as the fuel design limits described in paras 3.65–3.76 are not exceeded, considering the difference in cladding material and/or nuclear properties. | To add: 'these analyses should be performed for all fuel types in the core (e.g. $UO_2$ or mixed oxide fuel, with either zirconium alloy or FeCrAl cladding)'. |
| 3.30 | "The design should ensure that the minimum ratio of operating power to critical power (i.e. a minimum critical heat flux ratio, a minimum departure from the nucleate boiling ratio, a minimum critical channel power ratio or a minimum critical power ratio) takes into account that critical heat flux correlations have been developed from representative tests performed at steady state conditions" [3]. | Applicable with due consideration of the difference in cladding material and surface conditions on thermal hydraulic properties. | Fuel assembly CHF tests need to be performed with FeCrAl cladding to quantify the margins. |
| 3.40 | "In operational states, the design should ensure that the peak fuel temperature is lower than the fuel melting temperature by an adequate margin to prevent melting of the fuel, when appropriate provisions and uncertainties are considered. For design basis accidents (e.g. reactivity initiated accidents) and for design extension conditions without significant fuel degradation, incipient fuel melting can be allowed (e.g. fuel centreline melting limited to a small fraction of fuel pellet volume). The design and safety assessments should take into account the effects of fuel burnup on the fuel rod and fuel assembly properties (see Annex I for supplementary information)" [3]. | Applicable with due consideration of the difference in cladding material properties. | In operational states, limits on the peak fuel temperature are established to prevent fuel melt, it needs to be confirmed the applicability of limits established for uncoated zirconium alloy to FeCrAl cladding.<br>In particular, the reduction of melting temperature of FeCrAl cladding compared to zirconium alloy cladding needs to be addressed. |

TABLE 11. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO ADVANCED STEELS (FeCrAl) (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to FeCrAl | Suggested modification to SSG-52 [3] to enhance applicability to FeCrAl |
|---|---|---|---|
| 3.41 | "Straining of the cladding is caused by overpressure of internal gases in the fuel rod or by gaseous swelling of the fuel or thermal expansion of the fuel as a consequence of fuel burnup or local power increases. The design should ensure that cladding stresses and strains are limited. Limits for cladding stress, accumulated cladding strain, and cladding corrosion and hydriding should be specified for all applicable plant states and should be applied throughout the reloading cycle" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The effect of the difference of cladding material between FeCrAl and zirconium alloy on fuel design and phenomena (e.g. due to corrosion, hydriding, etc.) needs to be considered. |
| 3.51 | "Hydrogen pick-up correlations should be determined as a function of the corrosion of the cladding in normal operation for each cladding type, so that appropriate fuel design limits, such as for reactivity initiated accidents and loss of coolant accidents, can be expressed as a function of the pre-transient hydrogen content of the cladding (see Annex I for supplementary information)" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The extents of hydrogen pick-up and the embrittlement by hydrogen pick-up are dependent on cladding material, and such properties different from zirconium alloy need to be considered. |
| 3.54 | "The design analyses should take into account the degradation of the heat transfer from fuel rods due to the formation of deposits on the surface of the cladding via corrosion products coming from the reactor coolant system or other chemical changes. For pressurized water reactors, in the event that boron is trapped in the crud layer, its potential impact on the neutronic performance of the core should be assessed and addressed in the core design analyses" [3]. | Applicable with due consideration of the difference in the formation of deposits on cladding surface. | The deposition behaviour of corrosion products on FeCrAl cladding surface needs to be confirmed. If the amount of deposits is not negligible, the effect of the deposits formed on cladding surface needs to be confirmed from a thermal and/or neutronic viewpoint. |
| 3.59 | "The design should ensure that no fuel cladding failure takes place due to pellet–cladding mechanical interaction in normal operation and anticipated operational occurrences (see Annex I for supplementary information). The design of the fuel rods and plant specific guidelines for power changes in normal operation and anticipated operational occurrences should ensure that excessive pellet–cladding mechanical interaction is prevented" [3]. | Applicable with due consideration of the difference in material properties. | The effects of cladding mechanical properties (e.g. creep rates), which are different from those of zirconium alloy cladding, need to be confirmed.<br>The change in gap width during irradiation, which is different from that of the cladding with zirconium alloys needs to be considered and confirmed. |

TABLE 11. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO ADVANCED STEELS (FeCrAl) (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to FeCrAl | Suggested modification to SSG-52 [3] to enhance applicability to FeCrAl |
|---|---|---|---|
| 3.60 | “In design basis accidents that lead to rapid power transients (e.g. a reactivity initiated accident), the fuel cladding can fail due to excessive pellet–cladding mechanical interaction combined with cladding embrittlement due to in-reactor hydriding at high burnup levels. Fuel failures corresponding to this failure mode should be considered in safety analysis” [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | Cladding embrittlement due to in-reactor hydriding is material dependent, and FeCrAl cladding is not affected by embrittlement through the reaction with hydrogen and oxygen. The effect of the difference of material needs to be considered. The thinner wall thickness and lower creep rates of FeCrAl cladding vs. zirconium alloy cladding need to be considered during safety analysis of PCMI. |
| 3.63 | “The power-ramp failure threshold should be established, if applicable, in test reactors by means of power-ramp tests for each type of fuel or cladding. The data collected should cover the entire burnup range (see Annex I for supplementary information)” [3]. | Applicable. | The change in the power-ramp failure threshold of FeCrAl cladding by PCI-SCC phenomena needs to be studied with sufficient power-ramp tests. Analytical predictions need to be verified by experimental studies. |
| 3.66 | “For normal operation and for anticipated operational occurrences, the design of fuel rods should address at least the following limitations throughout the whole reloading cycle:<br>(a) No melting occurs in any location within the fuel pellets;<br>(b) No cladding overheating occurs (e.g. no departure from nucleate boiling for pressurized water reactors, critical power ratio below limits for boiling water reactors and no dryout condition for pressurized heavy water reactors);<br>(c) Fuel cladding does not collapse (light water reactor fuel only);<br>(d) The internal pressure of the fuel rods does not increase to the extent that cladding deformations caused by it would negatively affect the heat transfer between the fuel pellets and the coolant (i.e. there is no reopening of the fuel pellet–cladding gap by cladding lift-off);<br>(e) Fuel cladding corrosion and hydriding do not exceed specified limits;<br>(f) Cladding stress and strain remain below specified limits;<br>(g) Reduction of the cladding wall thickness (e.g. through wear or erosion) does not exceed specified limits” [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The impact of different material properties, corrosion behaviour, and cladding design needs to be duly addressed.<br>Some phenomena (mainly relating to (c), (e)–(g)) are material dependent. Cladding stress and strain are dependent on fuel cladding design. Hence, the effect of the difference of material and design needs to be considered. In terms of (d), the effects of material properties of FeCrAl cladding, e.g. on fuel temperature, need to be confirmed. |

TABLE 11. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO ADVANCED STEELS (FeCrAl) (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to FeCrAl | Suggested modification to SSG-52 [3] to enhance applicability to FeCrAl |
|---|---|---|---|
| 3.72 | “The ability to cool the core should not be endangered in the event of the following:<br>(a) Excessive ballooning or bursting of the fuel rods (e.g. in a loss of coolant accident);<br>(b) Significant deformation of fuel assembly components or reactor internals (e.g. in a seismic event);<br>(c) Flow blockage or other consequences of fuel dispersal and fuel coolant interaction as a result of fuel cladding failure (e.g. in a reactivity initiated accident).<br>The design of fuel rods should also be adequate to prevent undesired consequences of reactivity initiated accidents that may cause damage to the reactor coolant pressure boundary or damage that impairs the capability to cool the core. This is generally ensured by means of limits on the maximum fuel enthalpy and on the allowable increase in fuel enthalpy” [3]. | Applicable with due consideration of the difference in material properties (e.g. mechanical properties at high temperature). | The difference in mechanical properties (e.g. mechanical properties at high temperature) needs to be considered and confirmed. |

TABLE 11. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO ADVANCED STEELS (FeCrAl) (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to FeCrAl | Suggested modification to SSG-52 [3] to enhance applicability to FeCrAl |
|---|---|---|---|
| 3.73 | “To ensure that the structural integrity of the fuel rods is preserved, the following design limits should be defined and justified:<br>(a) The peak cladding temperature in accident conditions should not exceed a level at which cladding oxidation causes excessive cladding embrittlement or accelerates uncontrollably. In addition, for light water reactors, effects on the peak cladding temperature due to fuel fragmentation and its axial relocation within the ballooned area of the fuel rod should be assessed as appropriate. Possible effects of the dispersal of fuel particles on doses to workers and on core coolability should also be addressed.<br>(b) The total oxidation of the cladding should remain below limits such that the cladding can still withstand accident induced loadings (e.g. in the quenching phase of a loss of coolant accident). Such limits should be determined by experiments that take into account pre-transient in-reactor cladding oxidation and transient oxidation (outer side oxidation and possibly inner side oxidation), pre-transient and transient hydrogen absorption, as well as chemical interactions between the fuel pellets and cladding material.<br>(c) The allowable enthalpy rise for reactivity initiated accidents should be limited to values that take into account initial fuel rod conditions (e.g. pre-transient hydrogen content of the cladding and fuel burnup).<br>(d) If applicable, fuel centreline melting should be limited to a small fraction of fuel pellet volume.<br>(e) Fuel rods should be designed to withstand loadings resulting from post-transient fuel assembly handling, storage and transport to a reprocessing or disposal facility” [3]. | Applicable with due consideration of the difference in material properties (e.g. mechanical properties at high temperature). | The impact of different material properties, corrosion behaviour, cladding design, and oxidation behaviour at high temperature needs to be addressed.<br>In terms of items (a)–(c) and (e) of this para., some limits need to be changed considering the cladding design and the properties of cladding material and their modification with burnup (e.g. embrittlement with hydrogen pickup, cladding corrosion and hydriding). An unstable oxidation of FeCrAl cladding at a fast LOCA heat up condition needs to be taken into account.<br>A new performance metric would be needed to include the phenomena of wall thickness thinning, considering that the corrosion process of FeCrAl material is a competing reaction of the formation of an oxide layer. In this formulation, the effects of cladding heat up rate on unstable oxidation may need to be considered.<br>The reaction between cladding and its neighbouring material (e.g. fuel pellets) at high temperature needs to be taken into account to determine the acceptable limit. |

TABLE 11. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO ADVANCED STEELS (FeCrAl) (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to FeCrAl | Suggested modification to SSG-52 [3] to enhance applicability to FeCrAl |
|---|---|---|---|
| 3.155 | "When fuel assemblies of different types are loaded into the core (a 'mixed core'), all fuel assembly types should meet the fuel design limits for all applicable plant states. An assessment should be performed for the initial loading and subsequent reloading of mixed cores. It should include the dimensional, mechanical and thermohydraulic response of the various fuel types (e.g. in terms of pressure drop characteristics through the fuel assembly or assemblies and flow rate), the compatibility of each fuel assembly with the neutronic and thermohydraulic characteristics of the original core and with the related safety analyses. The critical heat flux or critical power correlation used in the core monitoring system should be valid for all fuel assembly types present in the mixed core" [3]. | Applicable. | The "fuel assemblies of different types" or "all fuel assembly types" needs to include ATFs. |
| 4.9 | "In-reactor testing of design features through irradiation of fuel rods or fuel assemblies in materials test reactors or through irradiation of lead use fuel assemblies in power reactors should be used to justify the specified maximum burnup limit for a new design. The following phenomena may be tested in this manner:<br>(a) Fuel and burnable absorber rod growth;<br>(b) Fuel rod bowing;<br>(c) Fuel rod, spacer grid and fuel channel (if present) oxidation and hydride levels;<br>(d) Fuel rod fretting and spacer (for pressurized heavy water reactors) fretting;<br>(e) Fuel assembly growth;<br>(f) Fuel assembly bowing;<br>(g) Fuel channel (for boiling water reactors) wear and distortion;<br>(h) Fuel rod ridging (i.e. pellet–cladding interaction);<br>(i) Fuel rod integrity;<br>(j) Hold-down spring relaxation (for pressurized water reactors);<br>(k) Spacer grid spring relaxation (for light water reactors);<br>(l) Control rod and guide tube wear (for pressurized water reactors)" [3]. | Applicable. | After the safety analysis for new cladding material is conducted and the integrity of fuel rods (and the assembly) during such testing is confirmed, in-reactor testing needs to be conducted step by step. The data and knowledge of irradiation behaviour of new cladding material need to be obtained in order to conduct safety analysis. |

## II.3. DOPED $UO_2$

The results of the detailed evaluation of the applicability of recommendations of SSG-52 [3] to doped $UO_2$ are provided in Table 12.

TABLE 12. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO DOPED $UO_2$

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to doped $UO_2$ | Suggested modification to SSG-52 [3] to enhance applicability to doped $UO_2$ |
|---|---|---|---|
| 1.5 | “This Safety Guide is intended mainly for NPPs that use natural and enriched $UO_2$ fuels and plutonium-blended $UO_2$ fuel (mixed oxide fuel) with zirconium alloy cladding. Unless otherwise specified, all recommendations apply to these fuel types” [3]. | Applicable with minor modification. | To add ‘(including advanced technology fuels, such as $UO_2$ doped fuel)’ |
| 2.8 | “The design of fuel rods and fuel assemblies should provide features that will facilitate future waste management (including reprocessing when applicable). The physical condition of discharged fuel assemblies from the reactor core will influence the design of the storage and disposal systems for the used fuel” [3]. | Applicable with due consideration of the effect of dopants on the reprocessing. | $UO_2$ doped fuel may not facilitate reprocessing, as opposed to design objectives stated in this paragraph.<br>For example, some forms of chromium may generate corrosion problems for reprocessing plants. Therefore, these issues need to be given appropriate consideration. |
| 3.38 | “Annex II describes important aspects that are typically taken into account in the design of fuel rods and fuel assemblies, and of reactivity control assemblies, neutron source assemblies and hydraulic plug assemblies, including the irradiation and environmental conditions” [3]. | Applicable. | The behaviour of the fuel is modified considering the changes of the material properties but there is no new phenomenon identified with $UO_2$ doped fuel technology. |
| 3.39 | “The design should ensure that fuel rods and fuel assemblies are reliable throughout their lifetime, including during manufacturing, transport, handling, in-core operation, storage and disposal, where applicable. Key contributors to fuel reliability should be addressed” [3]. | Applicable with due consideration of the difference in material properties. | The difference in material properties needs to be considered. |

TABLE 12. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO DOPED $UO_2$ (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to doped $UO_2$ | Suggested modification to SSG-52 [3] to enhance applicability to doped $UO_2$ |
|---|---|---|---|
| 3.40 | "In operational states, the design should ensure that the peak fuel temperature is lower than the fuel melting temperature by an adequate margin to prevent melting of the fuel, when appropriate provisions and uncertainties are considered. For design basis accidents (e.g. reactivity initiated accidents) and for design extension conditions without significant fuel degradation, incipient fuel melting can be allowed (e.g. fuel centreline melting limited to a small fraction of fuel pellet volume). The design and safety assessments should take into account the effects of fuel burnup on the fuel rod and fuel assembly properties (see Annex I for supplementary information)" [3]. | Applicable with due consideration of the difference in material properties. | Fuel melting temperature is still considered as a possible issue for the rod behaviour. |
| 3.41 | "Straining of the cladding is caused by overpressure of internal gases in the fuel rod or by gaseous swelling of the fuel or thermal expansion of the fuel as a consequence of fuel burnup or local power increases. The design should ensure that cladding stresses and strains are limited. Limits for cladding stress, accumulated cladding strain, and cladding corrosion and hydriding should be specified for all applicable plant states and should be applied throughout the reloading cycle" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | Pellet expansion and swelling of $UO_2$ doped fuel and its effect on the cladding might differ from $UO_2$. The strain and stresses on the cladding will have to be assessed in order to make proper predictions (fuel performance codes) and compute the margin for each design limit. |
| 3.42 | "For accident conditions, cladding deformation should be evaluated to determine the potential for cladding failure (e.g. burst or rupture) and any resulting release of fission products from the fuel" [3]. | Applicable with due consideration of the difference in material properties. | The difference in material properties needs to be considered. |
| 3.47 | "The design should include analyses to ensure that straining of the fuel cladding due to mechanical loads (e.g. coolant pressure, seismic loads) meets fuel design limits. The analyses should take into account radial gap closure kinetics, which depend on various parameters such as fuel densification, fuel swelling, fuel pellet cracking, fragmentation and its radial relocation within the fuel rod after a power change, cladding creep behaviour at low stress, initial internal pressure of the fuel rods, release of fission gases to the free volumes, and operating parameters including power history and coolant pressure" [3]. | Applicable with due consideration of the difference in material properties. | Same remark as for para. 3.41. |
| 3.48 | "Stress corrosion cracking induced by pellet–cladding interaction in the presence of corrosive fission products should be prevented (see Annex I for supplementary information)" [3]. | Applicable with due consideration of the difference in material properties. | Same remark as for para. 3.41. |

TABLE 12. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO DOPED $UO_2$ (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to doped $UO_2$ | Suggested modification to SSG-52 [3] to enhance applicability to doped $UO_2$ |
|---|---|---|---|
| 3.59 | "The design should ensure that no fuel cladding failure takes place due to pellet–cladding mechanical interaction in normal operation and anticipated operational occurrences (see Annex I for supplementary information). The design of the fuel rods and plant specific guidelines for power changes in normal operation and anticipated operational occurrences should ensure that excessive pellet–cladding mechanical interaction is prevented" [3]. | Applicable with due consideration of the difference in material properties. | It is important to characterize the behaviour of $UO_2$ doped fuel with respect to PCI. Paragraph 3.63 calls for such a characterization. |
| 3.63 | "The power-ramp failure threshold should be established, if applicable, in test reactors by means of power-ramp tests for each type of fuel or cladding. The data collected should cover the entire burnup range (see Annex I for supplementary information)" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The difference in material properties (including their burnup effect) needs to be considered. |
| 3.66 | "For normal operation and for anticipated operational occurrences, the design of fuel rods should address at least the following limitations throughout the whole reloading cycle:<br>(a) No melting occurs in any location within the fuel pellets;<br>(b) No cladding overheating occurs (e.g. no departure from nucleate boiling for pressurized water reactors, critical power ratio below limits for boiling water reactors and no dryout condition for pressurized heavy water reactors);<br>(c) Fuel cladding does not collapse (light water reactor fuel only);<br>(d) The internal pressure of the fuel rods does not increase to the extent that cladding deformations caused by it would negatively affect the heat transfer between the fuel pellets and the coolant (i.e. there is no reopening of the fuel pellet–cladding gap by cladding lift-off);<br>(e) Fuel cladding corrosion and hydriding do not exceed specified limits;<br>(f) Cladding stress and strain remain below specified limits;<br>(g) Reduction of the cladding wall thickness (e.g. through wear or erosion) does not exceed specified limits" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The difference in material properties (including their burnup effect) needs to be considered. |
| 3.68 | "To prevent fuel cladding failure caused by pellet–cladding mechanical interaction, possibly assisted by stress corrosion cracking, appropriate operating limits on power changes and power-ramp rates of change should be determined such that the power-ramp failure thresholds are not exceeded" [3]. | Applicable with due consideration of the difference in material properties. | $UO_2$ doped fuel technology needs to be assessed in terms of PCI. |

TABLE 12. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO DOPED $UO_2$ (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to doped $UO_2$ | Suggested modification to SSG-52 [3] to enhance applicability to doped $UO_2$ |
|---|---|---|---|
| 3.71 | "For design basis accidents and design extension conditions without significant fuel degradation, the following should be ensured:<br>(a) For accident sequences in which some fuel rod failures cannot reasonably be avoided, the number of fuel rod failures should not exceed a small percentage of the total number of fuel rods in the reactor core to minimize the radiological consequences of the accident under consideration.<br>(b) In determining the total number of fuel rod failures, all known potential failure mechanisms should be evaluated. Chemical reactions, including oxidation and hydriding, cladding ballooning or collapse of the cladding, or damage to the cladding caused by an increase in the fuel enthalpy, are some of the failure mechanisms that should be considered.<br>(c) Limits applied in assessing the risk for loss of cladding integrity should be based on experimental studies. In determining the limits, chemical, physical, hydraulic and mechanical factors affecting the failure mechanisms, as well as the dimensional tolerances of the fuel rods, should be comprehensively and conservatively evaluated. When fuel failure mechanisms and fuel failure limits are burnup dependent, irradiation effects on cladding and fuel properties should be considered in the experimental studies and should be incorporated into the analyses to ensure that the application of the experimental results is comprehensive.<br>(d) Fuel failure is considered to occur if the radial average enthalpy of a fuel rod at any axial location, calculated with validated tools, exceeds a certain value to be determined based on representative experimental results by appropriately adjusting test conditions to represent in-reactor conditions (test parameters to take into account include the coolant temperature, coolant pressure, coolant flow rate, reactivity insertion kinetics and fuel rod internal pressure). Since the mechanical resistance of the cladding changes with irradiation and may vary from one cladding type to another" [3]. | Applicable with due consideration of the difference in material properties. | The difference in material properties needs to be considered. |

TABLE 12. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO DOPED $UO_2$ (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to doped $UO_2$ | Suggested modification to SSG-52 [3] to enhance applicability to doped $UO_2$ |
|---|---|---|---|
| 3.73 | "To ensure that the structural integrity of the fuel rods is preserved, the following design limits should be defined and justified:<br>(a) The peak cladding temperature in accident conditions should not exceed a level at which cladding oxidation causes excessive cladding embrittlement or accelerates uncontrollably. In addition, for light water reactors, effects on the peak cladding temperature due to fuel fragmentation and its axial relocation within the ballooned area of the fuel rod should be assessed as appropriate. Possible effects of the dispersal of fuel particles on doses to workers and on core coolability should also be addressed.<br>(b) The total oxidation of the cladding should remain below limits such that the cladding can still withstand accident induced loadings (e.g. in the quenching phase of a loss of coolant accident). Such limits should be determined by experiments that take into account pre-transient in-reactor cladding oxidation and transient oxidation (outer side oxidation and possibly inner side oxidation), pre-transient and transient hydrogen absorption, as well as chemical interactions between the fuel pellets and cladding material.<br>(c) The allowable enthalpy rise for reactivity initiated accidents should be limited to values that take into account initial fuel rod conditions (e.g. pre-transient hydrogen content of the cladding and fuel burnup).<br>(d) If applicable, fuel centreline melting should be limited to a small fraction of fuel pellet volume.<br>(e) Fuel rods should be designed to withstand loadings resulting from post-transient fuel assembly handling, storage and transport to a reprocessing or disposal facility" [3]. | Applicable with due consideration of the difference in material properties | It may be necessary to consider the reaction between fuel pellet and cladding at high temperature to determine the limit.<br>The allowable enthalpy rise for reactivity initiated accidents needs to be characterized or assessed according to existing results. |

TABLE 12. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO DOPED $UO_2$ (cont.)

| SSG-52 [3] para. | SSG-52 [3] content | Applicability to doped $UO_2$ | Suggested modification to SSG-52 [3] to enhance applicability to doped $UO_2$ |
|---|---|---|---|
| 3.166 | “Design limits are determined, based on the concept of defence in depth, to meet safety requirements for all applicable plant states. The fuel design limits described in paras 3.65–3.76 should be extended to ensure that the fuel rods and fuel assemblies remain intact (when applicable) or do not degrade further (in the case of leaking fuel rods) in the back end of the nuclear fuel cycle after the assemblies are discharged from the core. The back end of the fuel cycle includes handling, transport, storage, reprocessing and disposal. The following fuel performance parameters are among those that may have an impact on the post-irradiation behaviour of the fuel rods and the fuel assemblies:<br>(a) Internal pressure of fuel rods at the end of life<br>Even though fuel rods can withstand some extent of overpressurization exceeding the normal coolant pressure without failure in normal operation, the handling of such highly pressurized used fuel rods might not be acceptable when coolant counter-pressure is diminished (e.g. in spent fuel storage facilities). This is particularly relevant for mixed oxide fuels that remain at a higher temperature for a longer period of time and continue to release helium gas from the fuel material.<br>(b) Massive cladding hydriding and cladding mechanical properties<br>Localized hydriding (e.g. due to spalling of the corrosion layer or due to axial pellet–pellet gaps) might not take place during normal operation or be of consequence in accident conditions, but such a condition may lead to delayed hydride cracking of zirconium alloy cladding in post-irradiation handling or storage, or undesired failures in the event of a transport accident” [3]. | Applicable with due consideration of the difference in material properties. | 3.166(a): $UO_2$ doped fuel will in theory have a benefit on the rod internal pressure nevertheless the internal pressure of the rods is still a parameter to consider in the frame of creep and also hydride reorientation.<br><br>3.166(e): This disposition is already taking into account the fact that new type of fuel needs to be compatible with the back end cycle. |

## II.4. SIC-SIC COMPOSITE

The results of the detailed evaluation of the applicability of recommendations of SSG-52 [3] to SiC-SiC composite are provided in Table 13.

TABLE 13. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO SIC-SIC COMPOSITE

| SSG-52 [3] para. | SSG-52 [3] Content | Applicability to SiC-SiC composite | Additional phenomena needing consideration or suggested modification to SSG-52 [3] to enhance applicability to SiC-SiC composite |
|---|---|---|---|
| 1.6 | "For innovative fuel materials, such as uranium nitride fuel or inert matrix fuel, or cladding materials other than zirconium alloys, this Safety Guide can be applied with judgement" [3]. | Applicable with minor modification. | To add 'cladding materials such as SiC-SiC composite'. |
| 2.26 | "Safety analysis for the reactor core should be performed to verify that fuel design limits are not exceeded in all applicable plant states. For accident conditions, the effect of fuel behaviour on core cooling should be included in the safety analysis (e.g. ballooning and rupture of the cladding, exothermic metal–water reactions, distortions of fuel rods and fuel assemblies). The effects of hydrogen accumulation (as a result of a metal–water reaction between the zirconium alloy cladding and water at high temperature) on the boundary of the reactor coolant system should be evaluated" [3]. | Applicable with due consideration of the difference in material properties (mechanical and thermal properties, the reaction with water, etc.). | New phenomena specific to SiC (e.g. the dissolution of silicon in water) need to be considered and added. |
| 3.5 | "Cladding materials should be selected with consideration of the following properties (examples of cladding materials are provided in Annex I):<br>(a) Low absorption cross-section for thermal neutrons;<br>(b) High resistance to irradiation conditions;<br>(c) High thermal conductivity and high melting point;<br>(d) High corrosion resistance and low hydrogen pick-up;<br>(e) Low oxidation and low hydriding in high temperature conditions;<br>(f) Adequate resistance to breakaway oxidation at high integrated-time temperature conditions;<br>(g) Adequate mechanical properties (e.g. high strength, high ductility, low creep rate in normal operation, high relaxation rate in transients);<br>(h) Low susceptibility to stress corrosion cracking;<br>(i) Adequate resistance to hydrogen assisted cracking and hydride related cracking in normal operation and for fuel storage" [3]. | Applicable with due consideration of the difference in cladding material properties (mechanical and thermal properties, nuclear properties, the reaction with water, etc.), including their burnup effect. However, large changes in existing safety criteria are expected based on the behaviour (e.g. corrosion in high temperature water, the stress-strain and fatigue mechanisms) of SiC different from that of zirconium alloys. | Some phenomena (mainly relating to (b), (d)–(f), (h) and (i)) are material dependent. The effect of the difference of material, e.g. chemical compatibility (i.e. hydrothermal corrosion) of SiC with the coolant at about 300 °C, no formation of an oxide layer on the cladding surface and the absence of hydrogen uptake, insusceptibility to corrosive fission products that would cause stress corrosion cracking, low pseudo ductility, relatively poor thermal conductivity under neutron irradiation, irradiation growth and swelling, etc., need to be considered.<br>Some items are neither suitable nor applicable for SiC-SiC composite cladding owing to its intrinsic properties that are different from those of zirconium alloy. |

TABLE 13. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO SIC-SIC COMPOSITE (cont.)

| SSG-52 [3] para. | SSG-52 [3] Content | Applicability to SiC-SiC composite | Additional phenomena needing consideration or suggested modification to SSG-52 [3] to enhance applicability to SiC-SiC composite |
|---|---|---|---|
| 3.22 | “The maximum reactivity worth of the reactivity control devices (e.g. control rods and/or chemical and volume control systems) should be limited, or interlock systems should be provided, so that any resultant power variations do not exceed specified limits for relevant reactivity insertion transients and accidents, such as the following:<br>(a) Control rod ejection;<br>(b) Control rod drop;<br>(c) Boron dilution;<br>(d) Uncontrolled withdrawal of control banks.<br>Such reactivity limits should be determined via safety analyses to ensure that the fuel design limits described in paras 3.65–3.76 are not exceeded. These analyses should be performed for all fuel types in the core (e.g. $UO_2$ or mixed oxide fuel) or a representative core with appropriate margins, and for all allowable operating conditions and fuel burnup values” [3]. | Applicable as far as the fuel design limits described in paras 3.65–3.76 are not exceeded, considering the difference in cladding material and/or nuclear properties. | To add ‘these analyses should be performed for all fuel types in the core (e.g. $UO_2$ or mixed oxide fuel, with zirconium alloy or SiC-SiC composite cladding)’. |
| 3.30 | “The design should ensure that the minimum ratio of operating power to critical power (i.e. a minimum critical heat flux ratio, a minimum departure from the nucleate boiling ratio, a minimum critical channel power ratio or a minimum critical power ratio) takes into account that critical heat flux correlations have been developed from representative tests performed at steady state conditions” [3]. | Applicable with due consideration of the difference in cladding material and surface conditions on thermal hydraulic properties. | Fuel assembly CHF tests need to be performed with SiC-SiC composite cladding to quantify the margins. |
| 3.40 | “In operational states, the design should ensure that the peak fuel temperature is lower than the fuel melting temperature by an adequate margin to prevent melting of the fuel, when appropriate provisions and uncertainties are considered. For design basis accidents (e.g. reactivity initiated accidents) and for design extension conditions without significant fuel degradation, incipient fuel melting can be allowed (e.g. fuel centreline melting limited to a small fraction of fuel pellet volume). The design and safety assessments should take into account the effects of fuel burnup on the fuel rod and fuel assembly properties (see Annex I for supplementary information)” [3]. | Applicable with due consideration of the difference in cladding material properties. | In operational states, limits on the peak fuel temperature are established to prevent fuel melt. the applicability of limits established for uncoated zirconium alloy to SiC-SiC composite cladding, and the effects of material properties of SiC-SiC cladding, such as thermal conductivity on fuel temperature, need to be confirmed. |

TABLE 13. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO SIC-SIC COMPOSITE (cont.)

| SSG-52 [3] para. | SSG-52 [3] Content | Applicability to SiC-SiC composite | Additional phenomena needing consideration or suggested modification to SSG-52 [3] to enhance applicability to SiC-SiC composite |
|---|---|---|---|
| 3.41 | "Straining of the cladding is caused by overpressure of internal gases in the fuel rod or by gaseous swelling of the fuel or thermal expansion of the fuel as a consequence of fuel burnup or local power increases. The design should ensure that cladding stresses and strains are limited. Limits for cladding stress, accumulated cladding strain, and cladding corrosion and hydriding should be specified for all applicable plant states and should be applied throughout the reloading cycle" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The effect of the difference of cladding material on some phenomena (relating to e.g. thermal conductivity, mechanical properties) needs to be considered.<br>To add: 'the leaktightness of cladding should be maintained even in consideration of thermal and/or mechanical stress generated or expected during irradiation'. |
| 3.51 | "Hydrogen pick-up correlations should be determined as a function of the corrosion of the cladding in normal operation for each cladding type, so that appropriate fuel design limits, such as for reactivity initiated accidents and loss of coolant accidents, can be expressed as a function of the pre-transient hydrogen content of the cladding (see Annex I for supplementary information)" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The extents of hydrogen pick-up and the embrittlement by hydrogen pick-up are dependent on cladding material. The hydrogen pick-up properties of SiC-SiC composite that are different from zirconium alloy need to be addressed.<br>Considering the absence of the hydrogen uptake caused by corrosion, this para. could be either eliminated, or the following wording included: 'for each cladding type excluding SiC-SiC composite'. |
| 3.54 | "The design analyses should take into account the degradation of the heat transfer from fuel rods due to the formation of deposits on the surface of the cladding via corrosion products coming from the reactor coolant system or other chemical changes. For pressurized water reactors, in the event that boron is trapped in the crud layer, its potential impact on the neutronic performance of the core should be assessed and addressed in the core design analyses" [3]. | Applicable with due consideration of the difference in the formation of deposits on cladding surface. | The deposition behaviour of corrosion products on cladding surface needs to be confirmed. If the amount of deposits is not negligible, the effect of the deposits formed on cladding surface needs to be confirmed from thermal and/or neutronic viewpoint. |

TABLE 13. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO SIC-SIC COMPOSITE (cont.)

| SSG-52 [3] para. | SSG-52 [3] Content | Applicability to SiC-SiC composite | Additional phenomena needing consideration or suggested modification to SSG-52 [3] to enhance applicability to SiC-SiC composite |
|---|---|---|---|
| 3.59 | "The design should ensure that no fuel cladding failure takes place due to pellet–cladding mechanical interaction in normal operation and anticipated operational occurrences (see Annex I for supplementary information). The design of the fuel rods and plant specific guidelines for power changes in normal operation and anticipated operational occurrences should ensure that excessive pellet–cladding mechanical interaction is prevented" [3]. | Applicable with due consideration of the difference in material properties. | The occurrence of PCMI needs to be confirmed by experiments and/or analysis. The change in gap width during irradiation is expected to be different from zirconium alloys and needs to be taken into account at the design phase, needs to be considered in this para. |
| 3.60 | "In design basis accidents that lead to rapid power transients (e.g. a reactivity initiated accident), the fuel cladding can fail due to excessive pellet–cladding mechanical interaction combined with cladding embrittlement due to in-reactor hydriding at high burnup levels. Fuel failures corresponding to this failure mode should be considered in safety analysis" [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | Cladding embrittlement due to in-reactor hydriding is material dependent. The effect of the difference of cladding material needs to be considered.<br>Due to the absence of hydrogen uptake of SiC, this para. relating to fuel failure mode by hydrogen induced embrittlement could be eliminated. Instead, the failure mode by low pseudo ductility (especially SiC-SiC composites) of cladding needs to be considered and included (e.g. '… combined with low pseudo ductility of cladding with e.g. SiC-SiC composite.', instead of '…combined with cladding embrittlement due to in-reactor hydriding at high burnup levels.') |
| 3.63 | "The power-ramp failure threshold should be established, if applicable, in test reactors by means of power-ramp tests for each type of fuel or cladding. The data collected should cover the entire burnup range (see Annex I for supplementary information)" [3]. | Applicable. | The power-ramp failure threshold of SiC-SiC composite cladding by PCI-SCC or other phenomena need to be studied with sufficient power-ramp tests. Analytical predictions also need to be verified by experimental studies. |

TABLE 13. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO SIC-SIC COMPOSITE (cont.)

| SSG-52 [3] para. | SSG-52 [3] Content | Applicability to SiC-SiC composite | Additional phenomena needing consideration or suggested modification to SSG-52 [3] to enhance applicability to SiC-SiC composite |
|---|---|---|---|
| 3.66 | “For normal operation and for anticipated operational occurrences, the design of fuel rods should address at least the following limitations throughout the whole reloading cycle:<br>(a) No melting occurs in any location within the fuel pellets;<br>(b) No cladding overheating occurs (e.g. no departure from nucleate boiling for pressurized water reactors, critical power ratio below limits for boiling water reactors and no dryout condition for pressurized heavy water reactors);<br>(c) Fuel cladding does not collapse (light water reactor fuel only);<br>(d) The internal pressure of the fuel rods does not increase to the extent that cladding deformations caused by it would negatively affect the heat transfer between the fuel pellets and the coolant (i.e. there is no reopening of the fuel pellet–cladding gap by cladding lift-off);<br>(e) Fuel cladding corrosion and hydriding do not exceed specified limits;<br>(f) Cladding stress and strain remain below specified limits;<br>(g) Reduction of the cladding wall thickness (e.g. through wear or erosion) does not exceed specified limits” [3]. | Applicable with due consideration of the difference in material properties including their burnup effect. | The impact of material properties, chemical compatibility with water coolant, cladding design, and maintenance of leaktightness of cladding under irradiation needs to be duly addressed.<br>The effect of the difference of cladding material needs to be considered because some phenomena (mainly relating to (c) (e)–(g)) are material dependent.<br>Regarding (d) and (f), the formation of microcracks and/or pores during irradiation needs to be considered in relation to the maintenance of leaktightness of fuel rods. In the case of SiC, it may be necessary to consider the effect of (g) by material dissolution into coolant water. |
| 3.72 | “The ability to cool the core should not be endangered in the event of the following:<br>(a) Excessive ballooning or bursting of the fuel rods (e.g. in a loss of coolant accident);<br>(b) Significant deformation of fuel assembly components or reactor internals (e.g. in a seismic event);<br>(c) Flow blockage or other consequences of fuel dispersal and fuel coolant interaction as a result of fuel cladding failure (e.g. in a reactivity initiated accident).<br>The design of fuel rods should also be adequate to prevent undesired consequences of reactivity initiated accidents that may cause damage to the reactor coolant pressure boundary or damage that impairs the capability to cool the core. This is generally ensured by means of limits on the maximum fuel enthalpy and on the allowable increase in fuel enthalpy” [3]. | Applicable with due consideration of the difference in material properties (e.g. mechanical properties at high temperature). | The difference in material properties (e.g. mechanical properties at high temperature) needs to be considered and confirmed. |

TABLE 13. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO SIC-SIC COMPOSITE (cont.)

| SSG-52 [3] para. | SSG-52 [3] Content | Applicability to SiC-SiC composite | Additional phenomena needing consideration or suggested modification to SSG-52 [3] to enhance applicability to SiC-SiC composite |
|---|---|---|---|
| 3.73 | "To ensure that the structural integrity of the fuel rods is preserved, the following design limits should be defined and justified:<br>(a) The peak cladding temperature in accident conditions should not exceed a level at which cladding oxidation causes excessive cladding embrittlement or accelerates uncontrollably. In addition, for light water reactors, effects on the peak cladding temperature due to fuel fragmentation and its axial relocation within the ballooned area of the fuel rod should be assessed as appropriate. Possible effects of the dispersal of fuel particles on doses to workers and on core coolability should also be addressed.<br>(b) The total oxidation of the cladding should remain below limits such that the cladding can still withstand accident induced loadings (e.g. in the quenching phase of a loss of coolant accident). Such limits should be determined by experiments that take into account pre-transient in-reactor cladding oxidation and transient oxidation (outer side oxidation and possibly inner side oxidation), pre-transient and transient hydrogen absorption, as well as chemical interactions between the fuel pellets and cladding material.<br>(c) The allowable enthalpy rise for reactivity initiated accidents should be limited to values that take into account initial fuel rod conditions (e.g. pre-transient hydrogen content of the cladding and fuel burnup).<br>(d) If applicable, fuel centreline melting should be limited to a small fraction of fuel pellet volume.<br>(e) (e) Fuel rods should be designed to withstand loadings resulting from post-transient fuel assembly handling, storage and transport to a reprocessing or disposal facility" [3]. | Applicable with due consideration of the difference in material properties (e.g. mechanical properties at high temperature). | The impact of material properties and cladding design needs to be duly addressed.<br>The design limits relating to (a)–(e) would significantly change with consideration of e.g. high resistance of SiC against melting, oxidation in high temperature steam, and rupture at elevated temperatures during LOCA, the absence of hydrogen uptake of SiC during normal operation, low pseudo ductility (especially SiC-SiC composites) of cladding, etc.<br>The reaction between fuel pellet and cladding at high temperature needs to be considered to determine the acceptable limit. |

TABLE 13. DETAILED EVALUATION OF THE APPLICABILITY OF RECOMMENDATIONS OF IAEA SSG-52 [3] TO SIC-SIC COMPOSITE (cont.)

| SSG-52 [3] para. | SSG-52 [3] Content | Applicability to SiC-SiC composite | Additional phenomena needing consideration or suggested modification to SSG-52 [3] to enhance applicability to SiC-SiC composite |
|---|---|---|---|
| 3.155 | “When fuel assemblies of different types are loaded into the core (a ‘mixed core’), all fuel assembly types should meet the fuel design limits for all applicable plant states. An assessment should be performed for the initial loading and subsequent reloading of mixed cores. It should include the dimensional, mechanical and thermohydraulic response of the various fuel types (e.g. in terms of pressure drop characteristics through the fuel assembly or assemblies and flow rate), the compatibility of each fuel assembly with the neutronic and thermohydraulic characteristics of the original core and with the related safety analyses. The critical heat flux or critical power correlation used in the core monitoring system should be valid for all fuel assembly types present in the mixed core” [3]. | Applicable. | The "fuel assemblies of different types" or "all fuel assembly types" need to include ATFs. |
| 4.9 | “In-reactor testing of design features through irradiation of fuel rods or fuel assemblies in materials test reactors or through irradiation of lead use fuel assemblies in power reactors should be used to justify the specified maximum burnup limit for a new design. The following phenomena may be tested in this manner:<br>(a) Fuel and burnable absorber rod growth;<br>(b) Fuel rod bowing;<br>(c) Fuel rod, spacer grid and fuel channel (if present) oxidation and hydride levels;<br>(d) Fuel rod fretting and spacer (for pressurized heavy water reactors) fretting;<br>(e) Fuel assembly growth;<br>(f) Fuel assembly bowing;<br>(g) Fuel channel (for boiling water reactors) wear and distortion;<br>(h) Fuel rod ridging (i.e. pellet–cladding interaction);<br>(i) Fuel rod integrity;<br>(j) Hold-down spring relaxation (for pressurized water reactors);<br>(k) Spacer grid spring relaxation (for light water reactors);<br>(l) Control rod and guide tube wear (for pressurized water reactors)” [3]. | Applicable. | After the safety analysis for new cladding material is conducted and the integrity of fuel rods (and assemblies) during such testing is confirmed, in-reactor testing needs to be conducted step by step. The data and knowledge of irradiation behaviour of new cladding material should be obtained in order to conduct safety analysis. |

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# Annex

## A-1. MODIFIED 37-ELEMENT FUEL BUNDLE FOR THE RECOVERY OF ERODED OPERATIONAL MARGIN IN AGED CANDU REACTORS IN CANADA

In horizontal channel type pressurized heavy water cooled reactors (i.e., CANDUs), large overpower trip margins are established at beginning-of-life. Over time, as the reactor ages, a loss of margin to dryout in the sagged channel is encountered due to various phenomena (including the pressure tube diametral creep, increase in the reactor inlet header temperature, increased hydraulic resistance of feeders), resulting in a steady reduction of the overpower trip margins. If no action is taken, the reactor may need to be operated at a reduced power to ensure enough operating margin to trip. Local power peaks (end flux peaking, refuelling ripples) will even worsen the situation.

Several advanced fuel concepts have been considered to enhance the CHF of the fuel bundle. Among these, the Canadian fuel community has focused on developing an alternative fuel concept (called the modified 37-element fuel bundle) that can replace the current reference 37-element fuel bundle in the aged reactor. The concept of the modified 37-element fuel bundle facilitates increasing the flow of the coolant to bundle interior by increasing the central subdivisions of the fuel bundle. Since dryout takes place in the central region of the reference 37-element fuel bundle, dryout power increases with an increased flow of the coolant and thus CHF of the bundle increases.

The dryout power of the modified 37-element fuel bundle was measured to increase by 3–5 % [A-1] compared to the reference 37-element bundle, which can be utilized to recover some loss of thermal margin in the aged reactor. This modified 37-element fuel bundle concept has been implemented in the Darlington and Bruce Nuclear Generation Stations, following extensive qualification activities of the bundle [A-2] and a subsequent regulatory review by the Canadian Nuclear Safety Commission. Although there was a concern that the modified 37-element bundle could increase the void reactivity during a LOCA, safety assessments confirmed that such increase could be marginal.

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## LIST OF ABBREVIATIONS

| | |
|---|---|
| AOO | Anticipated Operational Occurrence |
| ASN | Autorité de Sureté Nucléaire, France |
| ATF | Accident Tolerant Fuels, Advanced Technology Fuel |
| BWR | Boiling Water Reactor |
| CANDU | Canadian Deuterium Uranium reactor |
| CHF | Critical Heat Flux |
| CNSC | Canadian Nuclear Safety Commission, Canada |
| CRUD | Chalk River Unidentified Deposits |
| DBA | Design Basis Accident |
| DEC | Design Extension Condition |
| DEC A | Design Extension Conditions without significant fuel degradation |
| DNB | Departure from Nucleate Boiling |
| LOCA | Loss of Coolant Accident |
| LTA | Lead Test Assembly |
| LTR | Lead Test Rod |
| LWR | Light Water Reactor |
| NEA | OECD Nuclear Energy Agency |
| NRA | Nuclear Regulation Authority, Japan |
| NPP | Nuclear Power Plant |
| NUREG | U.S. Nuclear Regulatory Commission technical report designation |
| ODS | Oxide Dispersion Strengthened alloys |
| PCI-SCC | Pellet-Cladding Interaction induced Stress Corrosion Cracking |
| PCMI | Pellet-Cladding Mechanical Interaction |
| PWR | Pressurized Water Reactor |
| RGP | Relevant Good Practice |
| RIA | Reactivity Insertion Accident |

| | |
|---|---|
| SCC | Stress Corrosion Cracking |
| SSCs | Structures, Systems and Components |
| U.S. NRC | United States Nuclear Regulatory Commission |
| WGFS | OECD NEA Working Group on Fuel Safety |

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