IAEA Nuclear Energy Series

No. NF-G-2.1 (Rev. 1)

Basic Principles

Objectives

Guides

Technical Reports

Quality and Reliability Aspects in Nuclear Power Reactor Fuel Engineering

IAEA
International Atomic Energy Agency

IAEA NUCLEAR ENERGY SERIES PUBLICATIONS

STRUCTURE OF THE IAEA NUCLEAR ENERGY SERIES

Under the terms of Articles III.A.3 and VIII.C of its Statute, the IAEA is authorized to "foster the exchange of scientific and technical information on the peaceful uses of atomic energy". The publications in the **IAEA Nuclear Energy Series** present good practices and advances in technology, as well as practical examples and experience in the areas of nuclear reactors, the nuclear fuel cycle, radioactive waste management and decommissioning, and on general issues relevant to nuclear energy. The **IAEA Nuclear Energy Series** is structured into four levels:

(1) The **Nuclear Energy Basic Principles** publication describes the rationale and vision for the peaceful uses of nuclear energy.
(2) **Nuclear Energy Series Objectives** publications describe what needs to be considered and the specific goals to be achieved in the subject areas at different stages of implementation.
(3) **Nuclear Energy Series Guides and Methodologies** provide high level guidance or methods on how to achieve the objectives related to the various topics and areas involving the peaceful uses of nuclear energy.
(4) **Nuclear Energy Series Technical Reports** provide additional, more detailed information on activities relating to topics explored in the **IAEA Nuclear Energy Series**.

Each publication undergoes internal peer review and is made available to Member States for comment prior to publication.

The IAEA Nuclear Energy Series publications are coded as follows: **NG** – nuclear energy general; **NR** – nuclear reactors (formerly **NP–** nuclear power); **NF** – nuclear fuel cycle; **NW** – radioactive waste management and decommissioning. In addition, the publications are available in English on the IAEA web site:

www.iaea.org/publications

For further information, please contact the IAEA at Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of the IAEA Nuclear Energy Series publications are invited to inform the IAEA of their experience for the purpose of ensuring that they continue to meet user needs. Information may be provided via the IAEA web site, by post, or by email to Official.Mail@iaea.org.

QUALITY AND RELIABILITY ASPECTS IN NUCLEAR POWER REACTOR FUEL ENGINEERING

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ANTIGUA AND BARBUDA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHAMAS
BAHRAIN
BANGLADESH
BARBADOS
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA, PLURINATIONAL STATE OF
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BRUNEI DARUSSALAM
BULGARIA
BURKINA FASO
BURUNDI
CABO VERDE
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
COMOROS
CONGO
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DJIBOUTI
DOMINICA
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ESWATINI
ETHIOPIA
FIJI
FINLAND
FRANCE
GABON
GAMBIA
GEORGIA
GERMANY
GHANA
GREECE
GRENADA
GUATEMALA
GUINEA
GUYANA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LAO PEOPLE'S DEMOCRATIC REPUBLIC
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORTH MACEDONIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PAPUA NEW GUINEA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
RWANDA
SAINT KITTS AND NEVIS
SAINT LUCIA
SAINT VINCENT AND THE GRENADINES
SAMOA
SAN MARINO
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
TOGO
TONGA
TRINIDAD AND TOBAGO
TUNISIA
TÜRKİYE
TURKMENISTAN
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VANUATU
VENEZUELA, BOLIVARIAN REPUBLIC OF
VIET NAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

IAEA NUCLEAR ENERGY SERIES No. NF-G-2.1 (Rev. 1)

QUALITY AND RELIABILITY ASPECTS IN NUCLEAR POWER REACTOR FUEL ENGINEERING

GUIDANCE AND BEST PRACTICES TO IMPROVE NUCLEAR FUEL RELIABILITY AND PERFORMANCE IN WATER COOLED REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2024

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 26007 22529
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
www.iaea.org/publications

Printed by the IAEA in Austria
February 2024
STI/PUB/2072

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Quality and reliability aspects in nuclear power reactor fuel engineering : guidance / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2024. | Series: IAEA nuclear energy series, ISSN 1995–7807 ; no. NF-G-2.1 (rev. 1) | Includes bibliographical references.
Identifiers: IAEAL 23-01632 | ISBN 978–92–0–149923–3 (paperback : alk. paper) | ISBN 978–92–0–150023–6 (epub) | ISBN 978–92–0–150123–3 (pdf)
Subjects: LCSH: Nuclear fuels. | Nuclear fuel elements — Design and construction. | Nuclear fuel rods. | Water cooled reactors — Quality control.
Classification: UDC 621.039.54 | STI/PUB/2072

FOREWORD

The IAEA's statutory role is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world". Among other functions, the IAEA is authorized to "foster the exchange of scientific and technical information on peaceful uses of atomic energy". One way this is achieved is through a range of technical publications including the IAEA Nuclear Energy Series.

The IAEA Nuclear Energy Series comprises publications designed to further the use of nuclear technologies in support of sustainable development, to advance nuclear science and technology, catalyse innovation and build capacity to support the existing and expanded use of nuclear power and nuclear science applications. The publications include information covering all policy, technological and management aspects of the definition and implementation of activities involving the peaceful use of nuclear technology. While the guidance provided in IAEA Nuclear Energy Series publications does not constitute Member States' consensus, it has undergone internal peer review and been made available to Member States for comment prior to publication.

The IAEA safety standards establish fundamental principles, requirements and recommendations to ensure nuclear safety and serve as a global reference for protecting people and the environment from harmful effects of ionizing radiation.

When IAEA Nuclear Energy Series publications address safety, it is ensured that the IAEA safety standards are referred to as the current boundary conditions for the application of nuclear technology.

The IAEA Nuclear Energy Series provides a vision for the peaceful use of atomic energy and comprises various tiers of publications in a hierarchical structure, including Nuclear Energy Basic Principles, Objectives, Guides and supporting Technical Reports. The IAEA Nuclear Energy Series is consistent with and complementary to the IAEA Safety Standards Series. Nuclear Energy Basic Principles is the highest level publication in the IAEA Nuclear Energy Series and presents eight basic principles on which nuclear energy systems should be based to fulfil nuclear energy's potential to help meet growing global energy needs. The Nuclear Energy Objectives comprise the second level publications and describe what needs to be considered and the specific goals to be achieved at different stages of implementation. The Nuclear Fuel Cycle Objectives publication, which governs this Guide, sets out the objectives that need to be achieved in the area of the nuclear fuel cycle to ensure that the Nuclear Energy Basic Principles are satisfied. Fuel engineering and performance is one of topics dealt with in the Nuclear Fuel Cycle Objectives publication. This Guide provides recommendations to address the objective set for the fuel engineering and performance area.

To decrease costs and increase competitiveness, nuclear utilities seek to operate their nuclear power plants under more challenging and flexible operational conditions, with longer fuel cycles and higher burnups, which require modifications to fuel designs and materials. For decades, the IAEA has supported Member States in addressing topical issues that may be encountered under such demanding conditions for nuclear fuel via technical meetings, coordinated research programmes, the publication of Technical Reports and other means. Fuel reliability and performance can be secured by considering such topical issues during the entire lifetime of fuel, from its design, manufacture and operation through to its storage after discharge from the reactor. In this sense, there was a consensus within the nuclear fuel community that guidance needed to be developed to provide comprehensive recommendations on how to improve the reliability and performance of nuclear fuel in water cooled reactors.

This publication is a revision of IAEA Nuclear Energy Series No. NF-G-2.1, Quality and Reliability Aspects in Nuclear Power Reactor Fuel Engineering, which it replaces. This publication, NF-G-2.1 (Rev. 1), mainly reflects updated information on fuel reliability and performance issues, to maintain consistency with the recommendations presented in the new IAEA safety standards on the design of the reactor core (IAEA Safety Standards Series No. SSG-52), and to apply to newly developed advanced technology fuels. This revision provides guidance on fuel design changes, fuel manufacturing, qualification, in-reactor operation and on-site services to achieve excellence in fuel reliability and performance.

The IAEA wishes to thank N. Waeckel (France), J. Zhang (Belgium) and J. Judah (Canada) for their contributions to the drafting of this publication. The IAEA officer responsible for this publication was K.S. Sim of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The more sophisticated technologies become, the more important reliability is to guaranteeing the properties and operational characteristics of these technologies. A nuclear reactor is generally characterized by challenging operational conditions, which include extreme conditions within the reactor core, where high temperatures and coolant flow, and where corrosive media and mechanical stresses are combined with intensive irradiation of the fuel rods, the fuel assemblies and the reactor core internals. All of these operational aspects can lead to the degradation of the material properties, which reduces the margins for all applicable plant states,[1] and ultimately to the failure of fuel and other core components.

The operational cost of such failures is usually high, not to mention the possible consequences of the events for nuclear plant safety. Therefore, careful attention is paid to the selection of materials used for the fuel and in-core components of the nuclear reactor as well as to their design, manufacture and qualification testing. The goal is to ensure their reliability and performance with adequate margins during their operational lifetime.

When applied to nuclear fuel engineering, the concepts of reliability, performance[2] and quality[3] are interconnected, even though sometimes the terms are used separately by fuel manufacturers (stressing ‘quality’ to minimize costly non-conformance manufacturing wastes) and fuel operators (stressing ‘reliability and performance’ to avoid costly failures during operation).

Nuclear power belongs to a highly competitive power industry that aspires to better commercial nuclear power plant performance within defined

[1] The term ‘all applicable plant states’ is defined in IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants [1], as follows:
“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’”.

[2] ‘Fuel reliability’ characterizes the ability of the fuel to behave accordingly to the fuel design requirements. ‘Fuel performance’ characterizes the ability of the fuel to show acceptable behaviour in all applicable plant states. In addition, ‘fuel robustness’ characterizes the ability of the fuel to withstand not only the above mentioned design requirements but also ‘unexpected’ operational situations.

[3] ‘Quality’ usually refers to the quality assurance and control of the processes in the various stages of fuel development (e.g. design, manufacturing, operation).

margins. Nuclear power development reflects the evolving compromise between technoeconomic incentives and safety requirements. Hence, both technical and safety aspects are to be considered, along with managerial approaches aiming to implement this philosophy in a practical and effective way.

Whereas the separate technical and safety aspects of fuel are dealt with in various publications, there seems to be a shortage of holistic guidance on integrated approaches to enhancing fuel reliability and performance together with quality.

Fuel designers, vendors and utilities have their own quality management systems; nevertheless, within the current globalization of the fuel market and the growing concerns about nuclear safety and the security of the fuel supply, harmonization of national practices and sharing best experiences will provide an effective way to ensure the high reliability and performance of nuclear fuel in the reactor.

1.2. OBJECTIVE

This publication describes how to achieve the nuclear fuel cycle objective on the basis of the benefits set for the area of nuclear fuel engineering and performance. Ref. [2] defines the objective of said benefits as follows:

> "*Fuel materials and designs are developed, and fabrication technologies are implemented to provide nuclear energy with benefits that outweigh the associated costs and risks, to achieve reliable and economical power generation, improved safety and reduced environmental impact*."

Poor fuel reliability can cause fuel failures, generating undesirable releases to the environment and additional waste management to deal with, eventually resulting in high associated costs for remedial actions together with additional costs for 'failed core' operation and maintenance. Poor fuel performance can lead to uncompetitive operational conditions for a nuclear power plant — for example, flexible operational conditions need to be restricted, or fuel assemblies need to be discharged before the anticipated end of cycle.

Therefore, this publication is intended to provide technical guidance to ensure excellence in the reliability and performance of nuclear fuel in water cooled reactors.

1.3. SCOPE

In line with the objective described in Section 1.2, the following scope is considered:

(a) Review of in-reactor performance issues affecting fuel reliability and proposed mitigation measures;
(b) Guidance on the qualification of design and design changes for nuclear fuel and reactor core components to ensure high performance and safe operation under all applicable plant states;
(c) Guidance on maintaining adequate margins to ensure the high performance and safe operation of nuclear fuel under all applicable plant states (e.g. to allow for operating condition changes, fuel design evolutions, mixed cores in the reactor, etc.);
(d) Good practices for the plant operators to optimize core loadings and irradiation conditions to avoid failures and improve overall fuel reliability and performance;
(e) Guidance on maintaining a high and constant level of quality during the manufacturing processes for nuclear fuel and reactor components;
(f) Good practices for on-site services and plant operation to improve fuel reliability (e.g. timely poolside inspections or damaged fuel assembly repairs, etc.).

This publication is applicable to all types of water reactors, including pressurized water reactors (PWRs), boiling water reactors (BWRs) and pressurized heavy water reactors (PHWRs), unless otherwise specified.

All guidance statements in this publication are mainly applicable to ‘operational states’, including normal operation and anticipated operational occurrences (AOOs), unless otherwise specified.

1.4. STRUCTURE

This guide includes six sections as well as related information in three appendices and four annexes.

Section 1 provides background information and defines the objective and the corresponding scope to achieve it.

Section 2 provides a brief description of the nuclear fuels to which the guidance statements in this publication are applicable. The major features of the fuel designs used in different water cooled reactor types are described, including fuel assembly structures, the materials used in the fuel assemblies and accident tolerant and advanced technology fuels (ATFs).

Section 3 describes in-reactor fuel degradation and failure mechanisms, where fuel performance issues together with mitigation actions are described. On-site fuel inspection and services such as failed fuel detection and repair are also addressed. Examples of fuel inspection and service techniques are described in Annex I.

Section 4 provides guidance to ensure that any fuel design change proposed by a fuel vendor is going to improve fuel reliability and performance. The methodology for margin quantification and management is described, as are the design and safety limits considered for fuel rod thermomechanical design, fuel rod safety evaluation and fuel assembly mechanical design verification. Fuel thermohydraulic design and neutronic design, as well as fuel reliability assessment during reload design, are also explained. As supplementary information for Section 4, fuel design and safety limits for operational states and accident conditions are described in Appendix I, and their verification is discussed in Appendix II.

Section 5 provides guidance for applying quality management to fuel design and manufacture in order to improve fuel reliability and performance. Quality management requirements applicable to fuel design activities and fuel manufacturing activities are described. The requirements for software quality assurance are also described. As supplementary information for Section 5, fuel manufacturing steps and respective quality control are addressed in Appendix III. Annexes II–IV provide examples and information related to the fuel manufacturing process and fuel product controls.

Section 6 describes the nuclear industry's practice in improving fuel reliability and performance in plant operation.

2. FUEL DESIGNS FOR DIFFERENT WATER COOLED REACTOR TYPES

2.1. GENERAL DESCRIPTION OF FUEL ASSEMBLY DESIGNS

There are a wide variety of different fuel assembly types for water cooled reactors. Updates on fuel assembly design, including fuel pellet and cladding design, are periodically published in Nuclear Engineering International [3].

2.1.1. Light water reactor fuel assemblies

The fuel rod array for BWRs has evolved from 7 × 7 or 8 × 8 to either 9 × 9, 10 × 10 or 11 × 11 square configuration designs. The driving force for this trend was to reduce the peak linear heat rating (LHR) of fuel rods, which can help mitigate several fuel performance issues, including lowered fission gas release and enhanced pellet–cladding interaction (PCI). To increase utility competitiveness, the average LHRs of fuel assemblies with an increased number of subdivisions (e.g. 9 × 9 and 10 × 10 fuel assemblies) have successively been increased, while maintaining peak LHRs that are almost comparable to those of the old designs (e.g. 7 × 7 and 8 × 8 fuel assemblies).

The fuel rod array for PWRs has also evolved, with increased subdivisions within the fuel assemblies (e.g. from 14 × 14 or 15 × 15 to 16 × 16, 17 × 17 or 18 × 18 designs). To accomplish this, the designers also needed to modify the reactor design (e.g. upper internals) to be compatible with the new fuel assembly designs, noting that PWRs are not as flexible as BWRs in terms of core internals and control rod management.

In PWRs, fuel assemblies are positioned using bottom and top fittings, maintaining the lateral clearance between spacer grids to permit proper fuel assembly handling during outages. The control rods, consisting of rod cluster control assemblies (RCCAs), move into guide thimbles (or guide tubes). These guide thimbles are components of the integral assembly structure.

In all BWRs the fuel assemblies are enclosed in fuel channels, between which the control rod blades are inserted.

Irrespective of the many possible different shapes, sizes and configurations, fuel assembly designs need to ensure compatibility with the reactor core boundary conditions and the fuel handling system to:

— Maintain proper positioning of the fuel rods under all applicable plant states;
— Permit safe and rapid fuel assembly handling before and after irradiation.

Figures 1 and 2 show typical PWR and BWR fuel assemblies, respectively. In addition, the different fuel assembly components are shown and the material selections for these components are provided. The reason for the difference in structural material selection is that the most inexpensive material is generally chosen for a specific component, provided that it yields the lowest cost to produce the component while ensuring adequate performance during all applicable plant states.

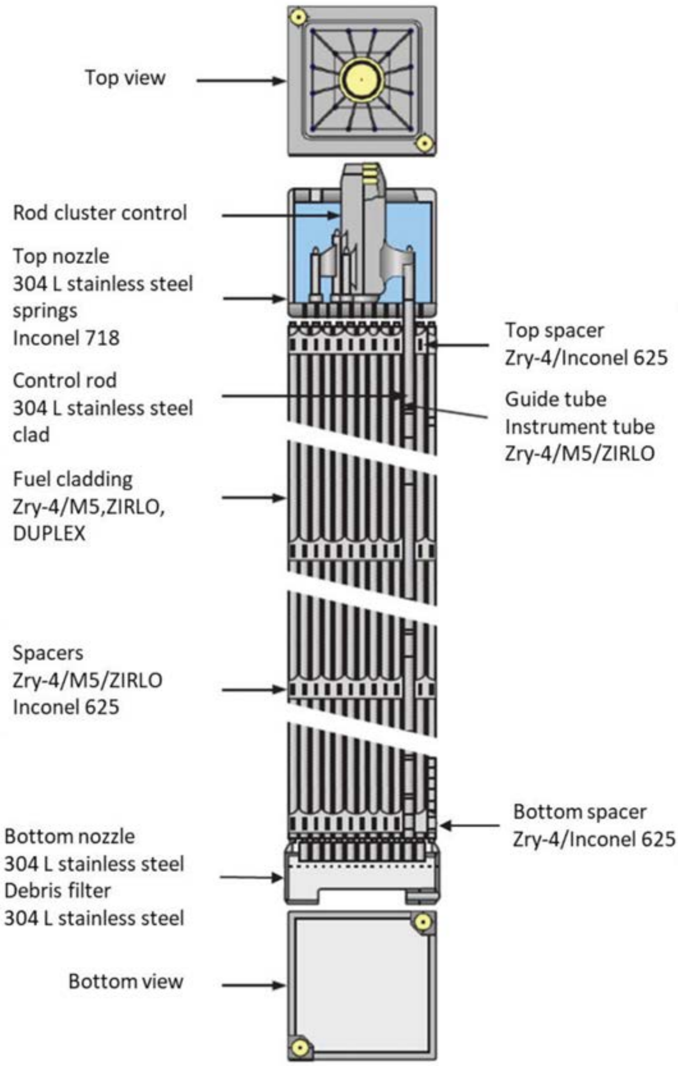


FIG. 1. Typical PWR fuel assembly. (Courtesy of ANT International.)

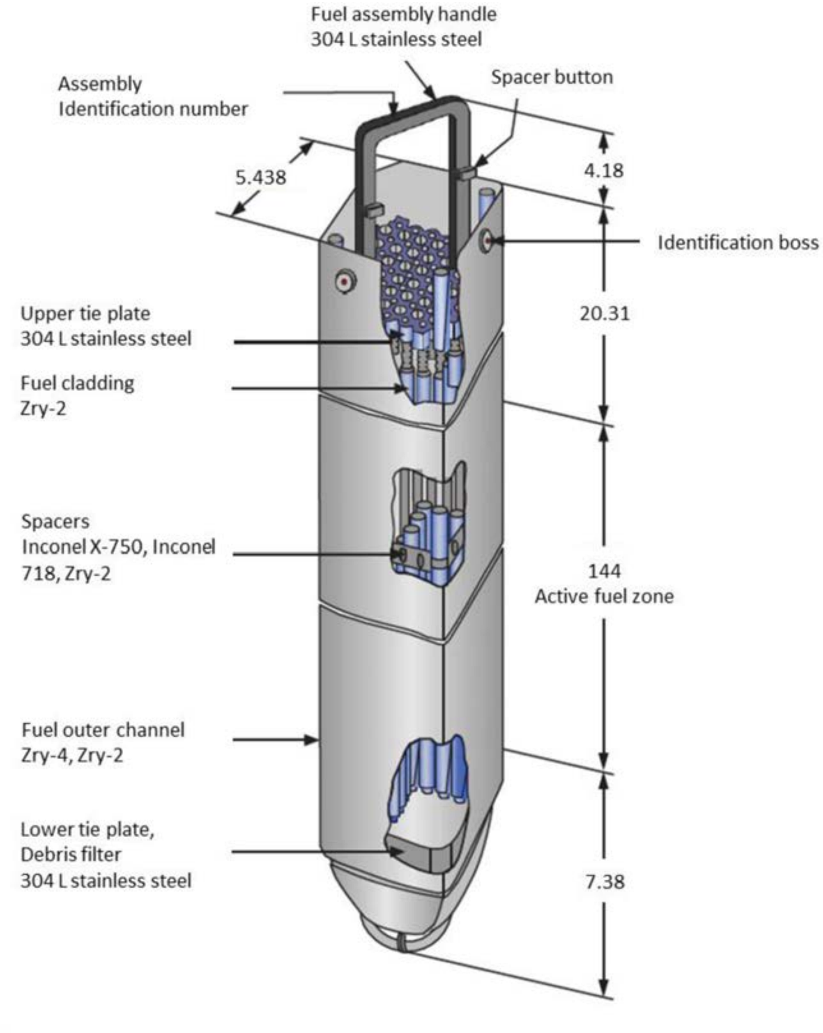


FIG. 2. Typical BWR fuel assembly. (Courtesy of ANT International.)

2.1.2. Water water energy reactor fuel assemblies

Two types of water water energy reactor (WWER)[4] core designs are in operation, WWER-440 and WWER-1000/1200.

The WWER-1000 reactor core consists of 163 fuel assemblies, incorporating the control protection system absorber rods (i.e. RCCAs). The fuel assemblies are spaced in a plane by fixing the fuel assembly end fittings in the protective tube unit and in the core barrel bottom inside the reactor internals. Fuel assembly rising is prevented and vibration is reduced by means of elastic compression of spring loaded fuel assembly top nozzles by the reactor cover via the protective tube unit. The WWER-1000 fuel assembly consists of the following components:

— Skeleton;
— Fuel rod bundle;
— Top nozzle;
— Bottom nozzle.

General views of two basic designs for the WWER-1000 fuel assembly are provided in Fig. 3.

2.1.3. Pressurized water reactor fuel bundles

The fuel bundle is an assembly of fuel rods. In most current PHWR fuel designs (which are categorized into two groups [4] — horizontal channel-type PHWRs and vertical channel-type PHWRs[5]) there are 37 fuel rods. Figure 4 shows a fuel bundle from a Canadian deuterium–uranium (CANDU) reactor.

---

[4] Also known as 'water cooled, water moderated energy reactor', a PWR of Soviet/Russian design.

[5] Operating PHWRs are categorized into two groups in Technical Review of Acceptance Criteria for Pressurized Heavy Water Reactor Fuel (IAEA-TECDOC-1926) [4]; see footnotes 2–4 of that publication:

> "Horizontal channel-type PHWRs include CANDUs (or Canadian PHWRs), Indian PHWRs and Pakistan's KANUPP reactor, and are equipped with horizontal channels each of which contains multiple fuel bundles. Vertical channel-type PHWRs include Argentina's Atucha reactors and are equipped with vertical channels each of which contains a single fuel bundle … CANDUs currently in operation worldwide are as follows: CANDU 6 (Point Lepreau (Canada), Wolsong (Korea), Qinshan (China), Cernavoda (Romania) and Embalse (Argentina)), and the Pickering/Bruce/Darlington nuclear reactors in Canada … Indian operating PHWRs are similar in concept to the CANDUs. In the early 1960s, the RAPS-1 design was the result of a Canada/India collaboration."

In this context, the term 'CANDUs' is used interchangeably with 'PHWRs'.

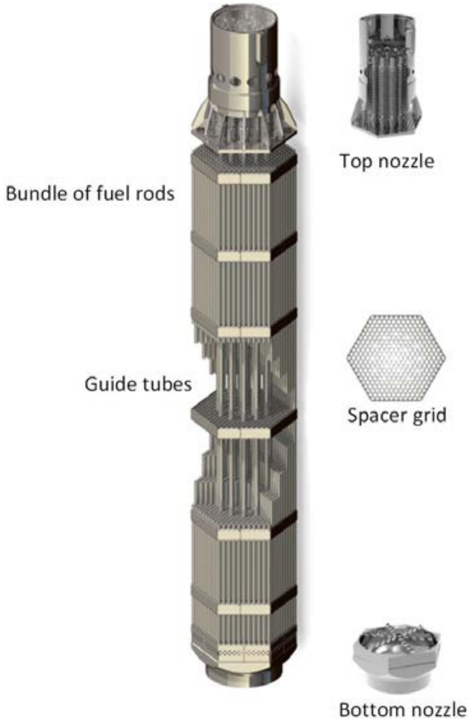


FIG. 3. Example of the WWER-1000 fuel assembly design. (Courtesy of TVEL.)

FIG. 4. A CANDU fuel bundle (CANDU 6 design) consisting of 37 fuel rods. (Courtesy of Atomic Energy of Canada Ltd.)

The fuel rods are joined together by resistance welding to end plates at both ends. The 37 rods in the fuel bundle are arranged in concentric rings of 6, 12 and 18 fuel rods, with a centre rod. Each rod consists of a stack of sintered, natural uranium UO_2 pellets in a thin Zircaloy-4 cladding with end plugs welded at both the ends. The inside surface of the cladding is coated with a thin layer of graphite.

Two design variants of the 37-element fuel bundles are now in use: the reference 37-element fuel bundle (designated 37R) and the modified 37-element fuel bundle (designated 37M). In the original design, 37R, all 37 fuel rods have equal diameters. The 37M variant has a reduced diameter central rod, providing this bundle with a higher critical heat flux and higher operating safety margins. There are also two bundle length variants for each of these. The 'long' fuel bundles are 12.7 mm (0.5 inches) longer than the standard length fuel bundles, which have a nominal length of 495.3 mm. The long fuel bundles were developed to address issues related to reactor ageing and safety concerns associated with postulated reactivity transients during some accident scenarios and some reactor fuelling processes in the Bruce and Darlington reactors.

The CANDU fuel bundles used in the Pickering reactors contain only 28 rods in each bundle. The outer dimensions (diameter and length) of these fuel bundles are the same as for 37-element fuel bundles, but the individual rods of the 28-element fuel bundles have larger diameters and slightly thicker fuel claddings. The rods of a 28-element bundle are arranged in 3 concentric rings of 4, 8 and 16 rods.

The fuel rods are prevented from direct fuel cladding to cladding contact by interelement spacing through split spacers consisting of Zircaloy-4. These spacers are attached to the outside surface of the cladding using beryllium brazing. The spacers are attached to each of the neighbouring rods such that the two spacers are in contact with each other in an opposite skewed angle position. This arrangement reduces the tendency of fuel rods to become interlocked.

Bearing pads are attached to the rods of the outer ring of the bundle at three axial planes at the middle and near the bundle ends. There are two designs for bearing pad configuration. For CANDU 6 and Pickering fuel bundles there are three planes of bearing pads, as shown in Fig. 4. However, on the Bruce and Darlington fuel bundles, the outer bearing pads are placed alternately at 'inboard' and 'outboard' positions, providing five planes of bearing pads on each bundle, with three bearing pads on each outer element. These bearing pads are made of Zircaloy-4 and have a curvature matching the inside surface of the pressure tubes. The end plates are made of Zircaloy-4 and have a web structure to allow the coolant to flow through the open cross-sectional area of the end plate into the subchannels of the fuel bundle.

2.1.4. Small modular reactor fuel assemblies

There is growing interest from IAEA Member States in the deployment of small modular reactors (SMRs). Compared with large commercial nuclear power plants, SMRs exhibit various benefits, such as lower capital investment, faster building processes based on modularity, enhanced safety features, less demanding siting requirements and interesting operational conditions, such as a partial shutdown of the plant to reload one module at a time while still producing electricity.

Today, more than 70 SMR designs and concepts are available worldwide [5]. These SMR designs and concepts range from scaled down existing nuclear reactor designs (e.g. water cooled SMRs) to entirely new SMR concepts based on generation IV designs. Most of these SMR designs and concepts are at various stages of development. Some are claimed to be near term deployable. Therefore, SMR fuels also come with various designs and concepts. Ultimately, most SMR fuels are intended to operate at higher burnup and with longer fuel cycles.

Water cooled SMRs are soon to be deployed in many countries. The fuels used in water cooled SMRs are generally selected from the existing fuel designs to benefit from extensive in-reactor experience. For example, most light water SMRs (land based) consider a conventional 17 × 17 assembly design but with shorter lengths. The fuel for the unique operating commercial SMRs of the Russian floating nuclear power plant has been designed using the vast experience of designing and operating icebreaker cores. Experience gained from the conventional fuels used in commercial power reactors can provide great support in reducing the development costs for light water SMR fuel (e.g. minimizing the number of experiments required for fuel qualification, the verification and validation of specific calculation tools, and the overall licensing process). However, SMR fuel assemblies need to be qualified for their specific operating conditions, which may differ from those of typical light water reactors (LWRs), such as severe power manoeuvring operation due to mechanical control systems as a result of boron free operation modes in some SMRs. Further, shorter fuel assemblies make them more robust from a mechanical point of view but more challenging in terms of applied axial thermal gradients.

Some SMRs concepts do not consider on-site refuelling to improve siting flexibility. Many SMR developers foresee the implementation of advanced fuel cycles in the longer term, when increasing the number of nuclear deployments in order to:

- Close the nuclear fuel cycle and optimize usage of radioactive material and waste management;

— Make a transition from ^{235}U fuelling to fuelling with transuranic and ^{233}U from ^{238}U and ^{232}Th, respectively.

As ATF concepts begin to be deployed in LWRs (refer to Section 2.3 for details), these concepts are also anticipated to be deployed in light water SMRs.

2.2. MATERIALS USED IN CURRENT FUEL ASSEMBLY DESIGNS

The materials used for the fuel assembly components are Zr alloys, Inconel[6] (precipitation hardened Inconel X-750, Inconel 718, solution treated Inconel 625) and stainless steel (SS 304L). Zr alloys are used predominantly for reactor core components because of their low thermal neutron cross-sections. Spring materials are those with low stress relaxation rates, such as Inconel X-750 or Inconel 718. These Ni base alloys are generally heat treated for optimal precipitation hardening. To lower the parasitic neutron absorption for grids and spacers, the strips are made of Zircaloy-2 and -4, while the spring itself is made of either Inconel X-750 or Inconel 718 to ensure adequate fuel rod support during its entire irradiation. In some fuel designs, the top and bottom PWR grid is made entirely of Inconel X-750 or Inconel 718. This is possible because the neutron flux is much lower at the top and bottom parts of the core, resulting in a very small loss of thermal neutrons owing to parasitic material absorption. The low neutron flux at the top and bottom parts of the core is also why the much cheaper material SS 304 L can be used for fuel and reactor core components at these elevations instead of Zr alloys, for example. In newer BWR designs the spacers are made entirely of Inconel X-750, using the minimum thicknesses possible.

Table 1 [6] presents the chemical compositions of stainless steel and nickel base alloys used in LWR fuel assemblies.

The main Zr based alloys commercially used for fuel assemblies and reactor core components include the following:

— Zircaloy-4 (PWR fuel cladding and structural materials, BWR channels, CANDU fuel cladding and assembly components) and Zircaloy-2 (BWR cladding and channels);
— Binary Zr–1Nb alloys: E110 (WWER and RBMK (a Russian graphite moderated LWR) cladding and fuel assembly structural material) and (M5 Zr based alloy cladding and fuel assembly structural material);
— Binary Zr–2.5Nb alloy (pressure tubes for PHWRs and RBMKs);

---

[6] Inconel, ZIRLO, Optimized ZIRLO, AXIOM, M5 and Q12 are registered trademarks.

TABLE 1. CHEMICAL COMPOSITION OF VARIOUS STAINLESS STEEL AND NICKEL BASE ALLOYS

Material	Concentration (wt%)								
	Fe	Ni	Cr	Mn	Si	Mo	Ti	Nb	Al
AISI 304	Bal. [a]	10	19	≤ 2	≤ 0.75	n.a. [b]	n.a.	n.a.	n.a.
DIN 1.4541	Bal.	11	18	≤ 2	≤ 0.75	n.a.	0.4	n.a.	n.a.
Inconel X-750	7	Bal.	15	≤ 1	≤ 1	n.a.	2.6	1	0.7
Inconel 718	17	Bal.	19	0.5	0.75	3	0.7	5	0.6
Inconel 625	12.5	Bal.	22	0.3	0.1	8.8	0.3	3.9	0.2

[a] Bal.: balanced

[b] n.a.: not applicable

Data taken from Ref. [6].

— Other Zr based alloys with additions of iron, niobium and tin (E635 and E110M for WWERs and ZIRLO or Optimized ZIRLO for PWRs) are used.

The chemical compositions of these alloys are presented in Table 2. This table characterizes preferable alloy compositions that are within the limits specified by patents or the American Society for Testing and Materials (ASTM) standards. This may be seen by comparing the data in Table 2 and the ASTM B-353 Standard (first version 1990, most recent version 2007 [7]) describing the chemical composition of Zircaloy-2 and Zircaloy-4 alloys (see Table 3).

Data on zirconium alloy cladding tubes are presented in the ASTM-811 Standard [8]. The composition and structure of these alloys are subject to constant modification because of changing requirements for materials intended for increasingly onerous in-pile operating conditions.

In some designs, the entire grid is made of Zr alloys (i.e. the grid, the dimples and the springs), requiring specific spring designs to ensure proper fuel rod support during the fuel assembly lifetime. Zircaloy-2, Zircaloy-4 and E110 alloys have been the main materials used for fuel claddings and other fuel assembly structural components for many years. They were subject to continuous optimizations and were partially replaced by more radiation and corrosion resistant alloys (e.g. E635 and E110M in the Russian Federation, M5 and Q12

TABLE 2. CHEMICAL COMPOSITION OF COMMERCIAL ZIRCONIUM ALLOYS

Alloy	Concentration (wt%)					
	Sn	Fe	Ni	Nb	Cr	O
Zircaloy-2	1.2–1.7	0.07–0.20	0.03–0.08	n.a.[a]	0.05–0.15	0.09–0.16
Zircaloy-4	1.2–1.7	0.18–0.24	n.a.	n.a.	0.07–0.13	0.09–0.16
E110	n.a.	n.a.	n.a.	0.9–1.1	n.a.	0.06
E110M	n.a.	0.07–0.15	n.a.	0.9–1.1	n.a.	0.1–0.15
M5	n.a.	n.a.	n.a.	0.9–1.2	n.a.	0.125
E125	n.a.	n.a.	n.a.	2.4–2.7	n.a.	0.05
Zr–2.5Nb	n.a.	n.a.	n.a.	2.4–2.8	n.a.	0.125
E635	1.1–1.3	0.3–0.45	n.a.	0.69–1.10	n.a.	0.08
Q12	0.5	0.1	n.a.	1.0	n.a.	n.a.
ZIRLO	1.0–1.1	0.09–0.10	n.a.	1.0–1.2	n.a.	0.125
Optimized ZIRLO	0.7	0.1	n.a.	1.0	n.a.	0.12
AXIOM	0.3	0.05	n.a.	0.8	n.a.	n.a.

[a] n.a.: not applicable.

in France, ZIRLO, Optimized ZIRLO and AXIOM in the United States of America). Note that advanced alloys have also been developed in other countries, for example, N36 and N45 in China, MDA and M-MDA in Japan and HANA in the Republic of Korea.

For CANDU fuel bundles, beryllium is used as a brazing material for the attachment of bearing and spacer pads to fuel elements. This use of beryllium represents a conventional hazard for the persons working in the fuel manufacturing plants.

TABLE 3. COMPOSITION RANGE OF STANDARD ZIRCONIUM ALLOYS, ASTM STANDARD B353-07

ASTM ref.	Common name	Concentration (wt%)					
		Sn	Fe	Cr	Ni	Nb	O
R60802	Zircaloy-2	1.20–1.70	0.07–0.20	0.05–0.15	0.03–0.08	n.a.[a]	TBS[b]
R60804	Zircaloy-4	1.20–1.70	0.18–0.24	0.07–0.13	n.a.	n.a.	TBS
R60901	Zr–2.5Nb	n.a.	n.a.	n.a.	n.a.	2.40–2.80	0.09–0.15
R60904	Zr–2.5Nb	n.a.	n.a.	n.a.	n.a.	2.50–2.80	TBS

[a] n.a.: not applicable.
[b] TBS: to be specified in the purchase order.

Also note that to improve fuel resistance to power ramps, a graphite coating has been implemented on the inner surface of the fuel cladding for CANDUs, and a Zr liner for BWRs.

2.3. NEW TYPES OF FUEL

New fuel developments, including ATFs, are driven mainly by the need for additional safety and operational margins, lower fuel cycle costs and innovative reactors.

Accident tolerant fuels, also called advanced technology fuels, can be defined as fuels that:

— Have the potential to enhance safety in the case of a severe accident in a reactor core for a longer time than the current UO_2–zirconium alloy fuel system (increased coping time);
— Maintain or improve fuel performance during normal operation and operational transients;
— Remain compatible with all aspects of the nuclear fuel cycle (transport, storage, possible use within a closed fuel cycle).

Among the multiple variants of ATFs proposed by fuel suppliers (see Fig. 5), two main categories of fuel are identified with respect to their technology

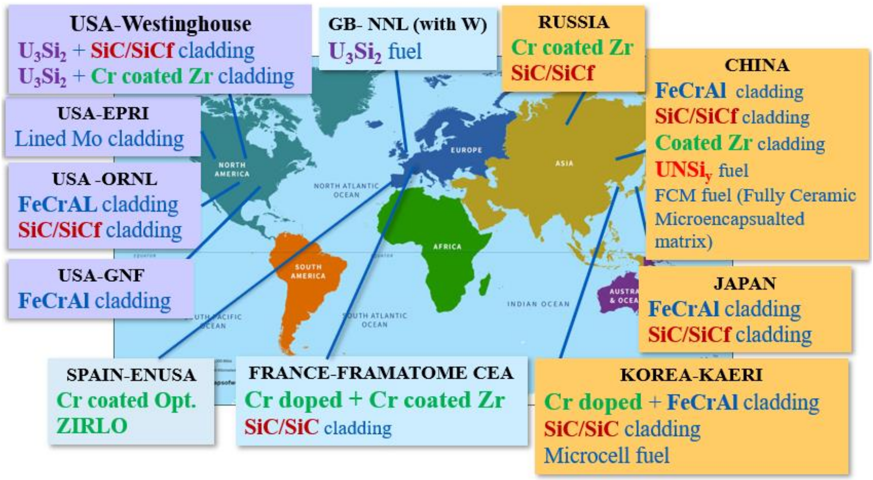


FIG. 5. The multiple variants of ATFs proposed by various fuel suppliers. (Courtesy of N. Waeckel.)

readiness level, which includes the time and effort required to develop, qualify and license the concepts:

(a) Short term 'evolutionary' concepts (e.g. coated Zr based claddings, stainless steel cladding (e.g. FeCrAl), high density fuel (e.g. U_3Si_2), doped fuel pellets (e.g. Cr doped);
(b) Longer term 'revolutionary' concepts, such as refractory claddings (e.g. lined Mo), silicon carbide claddings (e.g. SiC/SiC), microcells fuel, microencapsulated fuel pellet concepts and uranium nitride fuel.

Any new fuel concept needs to be compliant with current design, safety, operational and economic requirements. The overall fuel cycle (including reprocessing) also needs to be considered, particularly for concepts that differ significantly from current technologies.

To address these requirements, it is critical to establish the desired performance attributes of the new fuel concepts to guide their deployment.

The key requirements for advanced fuels [9] are related to in-reactor fuel performance, cladding performance and compatibility with all of the system constraints, including the following:

(a) In-reactor operations — an ATF needs to maintain or improve fuel behaviour in normal operation and maintain or extend plant operating cycles, reactor power output and reactor control. In particular, the innovative fuel concept needs to exhibit similar or improved performance with respect to stress corrosion cracking (SCC) or pellet–cladding interaction (PCI) to allow extensive load following (e.g. PCI resistance of SiC cladding concepts should be thoroughly demonstrated).

(b) Dose consequences — in cases of undesired cladding leakage (e.g. leaks induced by grid to rod fretting or foreign material fretting), the ATF concept needs to not interact adversely with the coolant, which would lead to detrimental fuel rod degradation, premature plant shutdown and unacceptable dose consequences (e.g. the uranium nitride fuel concept's reaction with coolant is a major issue that needs to be thoroughly addressed).

(c) Safety — an ATF needs to meet or exceed current fuel performance for all applicable plant states.

(d) Compatibility — ATF concepts need to comply with the existing fuel handling and storage systems and co-resident fuel in the reactor core.

(e) Front end of the nuclear cycle — ATF concepts need to comply with the regulations for both fuel fabrication facilities and operating plants (e.g. that might be of concern for high assay low enrichment uranium (HALEU) fuel concepts).

(f) Back end of the nuclear fuel cycle — ATF concepts need to not degrade the transport, storage (wet or dry) and repository performance of the fuel. Further, for fuel that is going to be reprocessed, the compatibility of the ATF concept with reprocessing techniques and protocols needs to be demonstrated.

It is noted that none of the proposed concepts are ideal; they all exhibit minor or major drawbacks in addition to their identified enhanced performance. Before selecting an ATF to be irradiated in its commercial power reactor, the utility requires a detailed assessment of the key attributes of the candidates.

Figure 6 shows an example of qualitative comparison of the key attributes of three types of fuel pellets. The survey shows that UO_2 is a good compromise. The utility is reluctant to implement fuel pellets that exhibit poor coolant compatibility.

Regarding accident tolerant claddings, in addition to their performance at high temperature in steam, the key attributes of the various cladding concepts need to be assessed, for example:

— PCI resistance (this might be challenging for SiC and Mo cladding tubes);
— Risk of eutectic between layers (for coated or lined claddings);
— Tritium retention (SiC claddings are not leak tight and require high quality or performance liners to ensure leak tightness);

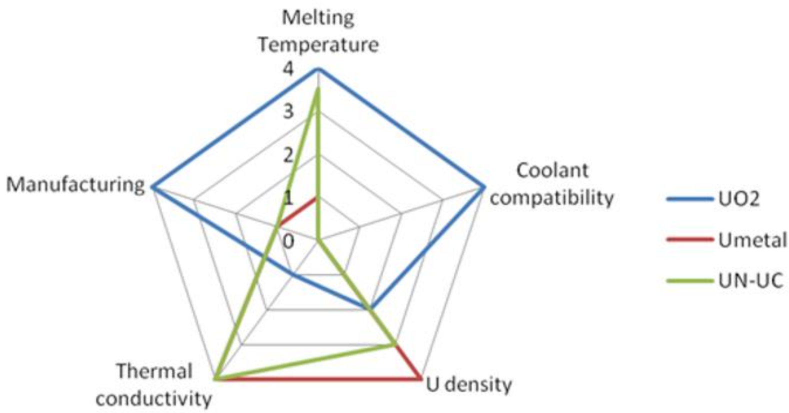


FIG. 6. Example of comparison of key attributes for three types of fuel. (Reproduced from Ref. [9] with permission from the NEA.)

— Behaviour under irradiation (coatings may crack and lose some of their efficiency);
— Neutronic penalty, which needs to be compensated elsewhere (a serious challenge for stainless steels and Mo, which may lead to higher enrichments, shorter irradiation cycles and lower discharge burnups);
— Compatibility with the coolant under irradiation (very challenging for uranium nitride and uranium silicide fuels in case of cladding failure, whatever the root cause) and for SiC (soluble in water and swelling), etc.

To compensate for some of the drawbacks listed above, the developers propose cladding with coatings or liners, sometimes on both sides of the cladding tube. The difficulty at this point is demonstrating that these mitigating features do not become the weak link of the ATF candidates because they may not play their protective role fully when they are damaged (e.g. eutectic at low temperature, spallation, scratches).

It is likely that some of the breakthrough technologies will require adapted specified acceptable fuel design limits to complete the safety demonstration. For instance, a 'damage index' (or something similar) needs to be defined for SiC claddings to account for the microcracks that could appear in the material during normal operation.

3. IN-REACTOR FUEL DEGRADATION, FAILURE MECHANISMS AND ON-SITE FUEL INSPECTIONS

3.1. FUEL PERFORMANCE ISSUES AND MITIGATION ACTIONS

The nuclear fuel present in the reactor core is renewed regularly, but, unlike the early days of the nuclear era, it can now stay in the reactor for up to six years, experiencing aggressive irradiation conditions. The fuel pellet and the cladding of the fuel rods are considered to be physical barriers in the sense of defence in depth. As a result, it is crucial to know as accurately as possible how irradiation influences the overall fuel rod behaviour under operational states (i.e. normal operation and AOOs). This process needs to include the long-term storage conditions, as well.

Table 4 summarizes the main in-reactor performance issues and proposes the corresponding mitigation measures.

In the reactor, the microstructure and the properties of the materials constituting the fuel are constantly evolving under the effect of irradiation. To model the fuel behaviour implies addressing multiphysics multiscale phenomena and mechanisms (i.e. neutronics, thermomechanics, thermohydraulics, thermochemistry). To deconvolute the various physical phenomena occurring within the fuel requires specific investigation strategies, including separate effect tests and in-pile integral tests. The objective is to turn the experimental observations into models and then implement those models in calculation tools capable of simulating the nuclear fuel behaviour in all kinds of operational and hypothetical accident conditions. Ultimately, these qualified calculation tools are used in application methodologies to assess the actual safety margins during all reactor operation states, including spent fuel pool situations.

The fuel is designed to withstand the aggressive irradiation conditions it will experience in the reactor pressure vessel: high fluid temperature (>300°C), high coolant pressure (1.55×10^7 Pa), high coolant velocity (>4 m/s), high neutronic flux and a specific chemical environment.

TABLE 4. FUEL PERFORMANCE ISSUES IN BOILING WATER REACTORS (BWRs), PRESSURIZED WATER REACTORS (PWRs), WATER WATER ENERGY REACTORS (WWERs) AND PRESSURIZED HEAVY WATER REACTORS (PHWRs) DURING NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (AOOs)

Performance issue	Short description	Mitigation actions
Manufacturing defects	Incipient cracks generated during cladding manufacture can propagate during operation (due to (PCMI[a])) Defective end plug welds can lead to a leak during normal operation Primary hydriding due to excess moisture in fuel pellets and due to moisture or organic contamination on the clad inner surface can embrittle the cladding Excessive gaps between fuel rods and spacer grid supports can lead to GTRF[b] failures Fuel assembly endogenous debris (SCC[c] failure of grid straps, cladding shavings during rod insertion) can lead to debris induced fretting failure Cladding shavings during rod insertion can lead to reduced grid to rod hold down force (‘gall ball’ formation) that, in turn, can cause GTRF due to flow induced vibration Defective pellets (chipped or missing surfaces) may result in PCI[d] failures at low power	Sound qualification of critical manufacturing and inspection processes Manufacturing quality surveillance Fuel assembly final inspection and cleaning Implement adequate investigation techniques to identify manufacturing defects (incipient cracks, cladding shavings), geometrical deviances, etc. Develop fuel assembly and reactor core component materials that are insensitive to SCC Automation of critical processes, including techniques such as artificial vision, digitalization and machine learning to minimize human errors

TABLE 4. FUEL PERFORMANCE ISSUES IN BOILING WATER REACTORS (BWRs), PRESSURIZED WATER REACTORS (PWRs), WATER WATER ENERGY REACTORS (WWERs) AND PRESSURIZED HEAVY WATER REACTORS (PHWRs) DURING NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (AOOs) (cont.)

Performance issue	Short description	Mitigation actions
	The following are specific to CANDU[e] reactors: — Inattention to the maintenance of internal diametral clearances can lead to pellet interaction with fuel cladding and to internal cladding scraping during fuel loading — Inattention to the maintenance of internal axial clearances can lead to stresses on the endcap to fuel cladding weld and weld failure, most likely due to hydride/deuteride accumulation and weld embrittlement	
Excessive corrosion and hydriding	Accelerated corrosion can result in cladding perforation: corrosion acceleration can be generated by water chemistry impacts such as crud deposition or thermohydraulic impacts leading to DNB[f]. The latter is also related to the excessive bowing of fuel rods or fuel assemblies	Manufacturing surveillance Cladding and guide tube design and material development Operational guidelines Regular poolside examinations
	Associated with excessive corrosion, some alloys may exhibit undesired hydriding, which can result in lower ductility of fuel cladding and excessive growth of guide tubes	For CANDU reactors, attention to these issues during off-normal primary heat transport system chemistry excursions because of either unexpected operational excursions or planned chemistry transients due to activities such as reactor refurbishment

TABLE 4. FUEL PERFORMANCE ISSUES IN BOILING WATER REACTORS (BWRs), PRESSURIZED WATER REACTORS (PWRs), WATER WATER ENERGY REACTORS (WWERs) AND PRESSURIZED HEAVY WATER REACTORS (PHWRs) DURING NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (AOOs) (cont.)

Performance issue	Short description	Mitigation actions
Grid to rod fretting	Cladding wear can take place due to relative motion between the fuel rod and the grid	Performing full scale hydraulic tests to assess changes that can affect resistance to GTRF. Hydraulic tests need to account for fuel assembly, fuel rod and grid strap vibration, in a full range of flow conditions for the plant (including mixed core effects). Tests also need to account for the evolution of grid to rod force/gap during operation and the evolution of the cladding (corrosion) Performing GTRF on-site inspections after changes. For major changes, introduction of lead test assemblies is considered Qualifying the fuel rod insertion process during manufacture, controlling the potential for grid spring damage and clad scratching
Stress corrosion cracking of structural components	SCC of different structural components (hold down springs, screws, joints) can lead to the mechanical fracture of some parts and can affect the fuel assembly functionality conditions	Control of manufacturing parameters for the components: time/temperature of thermal treatments Control of heat input during welding Control of spent fuel pool chemistry

TABLE 4. FUEL PERFORMANCE ISSUES IN BOILING WATER REACTORS (BWRs), PRESSURIZED WATER REACTORS (PWRs), WATER WATER ENERGY REACTORS (WWERs) AND PRESSURIZED HEAVY WATER REACTORS (PHWRs) DURING NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (AOOs) (cont.)

Performance issue	Short description	Mitigation actions
Excessive spacer grid growth or excessive fuel assembly bowing	Fuel assembly bowing results from excessive hold down spring forces and excessive guide tube growth from corrosion/hydriding and irradiation	Fuel assembly and fuel assembly components designed to make them as robust as possible
	Fuel assembly bowing can impair control rod insertion (safety issue) and degrade thermal margins (i.e. DNB) Fuel assembly bowing and excessive spacer growth can also hinder loading/ unloading and handling of the fuel assemblies during outages	Improved core reloads design to minimize fuel assembly bowing
Fuel channel bowing	In BWRs, fuel channel bowing may result in difficulties in inserting the control rods (safety issue) and/or to smaller thermal margins (loss of coolant accident and dryout/DNB). Failures may occur because local power is too high for the coolant flow, causing the cladding to be overheated. Overheating causes (local) corrosion penetration of the cladding	Proper fuel assembly and core reload designs
	In CANDU reactors, fuel channel bowing (sagging) can be caused by reactor ageing and can reduce clearances between fuel channels and internal reactivity mechanisms, such as liquid poison injection shutdown nozzles. The effectiveness of negative reactivity insertion may require reassessment. Verification by analysis that fuel operating conditions (fuel operating powers and fuelling power ramps) remain acceptable, might also be required	Retubing of sagged pressure tubes in CANDU reactors

TABLE 4. FUEL PERFORMANCE ISSUES IN BOILING WATER REACTORS (BWRs), PRESSURIZED WATER REACTORS (PWRs), WATER WATER ENERGY REACTORS (WWERs) AND PRESSURIZED HEAVY WATER REACTORS (PHWRs) DURING NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (AOOs) (cont.)

Performance issue	Short description	Mitigation actions
Fuel channel diametral increase	CANDU pressure tube diametral increases due to irradiation induced creep can change fuel channel coolant flows and reduce fuel bundle critical heat flux	The 37Mg was designed with a smaller diameter central rod to increase the critical heat flux of the bundle
PCI–SCC	The PCI iodine assisted SCC phenomenon may result in fuel rod failures during power increases. This failure mechanism is more frequent in BWRs but could also occur in PWRs and PHWRs. Three parameters need to occur simultaneously to induce PCI–SCC: high enough cladding tensile stresses (induced by the power ramp), proper concentration of corrosive fission products and sufficient time for the first two parameters to initiate and propagate a through the wall crack	Manufacturing surveillance (to avoid missing pellet surface defects, smaller pellet–clad gaps than the recommended value, etc.) Design of the reactor core with qualified fuel performance codes (validation based on in-pile power ramp tests series) Operating limitations Power monitoring during operation
Excessive rod internal pressure	Excessive in-reactor rod internal pressure is not a real issue during normal operation but can be during accident conditions and transport and storage conditions. It may result in detrimental hydride reorientations during the drying process or excessive cladding creep-out during the dry storage phase	Manufacturing surveillance to ensure the recommended initial free volume is implemented. Design fuel rods with validated fuel performance codes Advanced pellet design with reduced in-reactor fission gas release

TABLE 4. FUEL PERFORMANCE ISSUES IN BOILING WATER REACTORS (BWRs), PRESSURIZED WATER REACTORS (PWRs), WATER WATER ENERGY REACTORS (WWERs) AND PRESSURIZED HEAVY WATER REACTORS (PHWRs) DURING NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (AOOs) (cont.)

Performance issue	Short description	Mitigation actions
Excessive cladding temperature	Upon the occurrence of DNB, local cladding temperature may increase significantly Crud on the fuel rods or overly high fuel duty may result in excessive cladding surface temperature, which in turn causes an excessive corrosion rate	Operating limits on the pressure, temperature and flow rate, as well as the overpower/ overtemperature protection Operational guidelines (chemistry and LHR)

[a] PCMI: pellet–cladding mechanical interaction.
[b] GTRF: grid to rod fretting.
[c] SCC: stress corrosion cracking.
[d] PCI: pellet–cladding interaction.
[e] CANDU: Canadian deuterium–uranium.
[f] DNB: departure from nucleate boiling.
[g] 37M: modified 37-fuel element bundle.

There are various potential impacts of this specific environment on fuel behaviour under normal conditions or in post-irradiation situations (e.g. see Refs [10–16]). Examples are as follows:

(a) Manufacturing defects may lead to unreliable in-reactor fuel behaviours (e.g. the internal hydriding of fuel rods due to moisture in the pellets, pollution of end plugs welds leading to leaks, grid spring fretting leading to debris, fuel cladding scratches resulting from fuel rod insertion in the fuel assembly skeleton, etc.).
(b) Excessive corrosion/hydriding of the Zr alloy cladding can lead to its embrittlement and to a potential risk of failure during in-reactor transients, either due to excessive power or to lack of cooling, or under storage or shipment conditions. The excessive cladding corrosion may result from various causes: inappropriate water chemistry parameters, which make the coolant too aggressive or lead to crud deposits on the fuel rods (which in turn degrades the cooling of the cladding), high duty operational conditions

(i.e. local high void fraction enhancing cladding temperatures) or insufficient optimization of the chemical formula of the cladding.

(c) Fuel assembly or fuel rod induced vibration can cause grid to rod fretting (GTRF) that leads to cladding wear (in CANDU fuel bundles spacer pad wear) and to a potential risk of local loss of integrity of the fuel rod during normal operation.

(d) Stress corrosion cracking (SCC) can affect various structural components and can lead to their failure, eventually undermining the compliance of the fuel assembly with its functional requirements. For instance, mechanical fracture of the screws maintaining the hold down spring system to the top nozzles in a PWR can cause the loss of the required hold down force and can generate debris, causing additional damage (fretting wear, incomplete insertion of RCCAs, etc.).

(e) Foreign debris causing fretting can become trapped in the grid cells.

(f) Endogenous debris fretting materials can come from the fuel assembly itself:

   (i) During normal operation, irradiation of nickel based alloys (e.g. used in spacers and grids) will result in irradiation induced microstructural changes, which in turn may result in SCC, provided that the stresses in the material are large enough (e.g. see Ref. [10]). Such a mechanism may produce endogenous debris, which in turn can generate debris fretting wear of the cladding. The tendency for SCC may be related to both design and manufacturing [16].

   (ii) During the fuel assembly manufacturing process, the insertion of fuel rods within the fuel assembly skeleton may generate thin shavings along the fuel rod cladding that may become trapped in the grid cells, and these shavings may undergo vibration during reactor operation, leading to a debris fretting phenomenon. Further, these shavings may form a clot or ‘gall ball’ under the spring, which will increase the as-built spring deflection and its relaxation at power, reducing the grid to rod force and causing GTRF.

(g) In-reactor dimensional changes of Zr alloy components may occur:

   (i) Fuel assembly distortion in PWRs can lead to incomplete control rod insertion issues, neutronic penalties, local thermohydraulic modifications and post-irradiation handling issues.

   (ii) Fuel assembly or fuel rod excessive irradiation (together with corrosion and hydriding) induced growth can lead to a situation where the available gaps are insufficient to accommodate such dimensional changes, which can cause excessive fuel assembly or fuel rod bow, and excessive stresses in other fuel or reactor vessel structural parts.

   (iii) Excessive irradiation (together with corrosion and hydriding) induced grid growth can lead to severe fuel assembly handling limitations.

(iv) Distortion of BWR channels can restrict control blade movement, altering the clearance between neighbouring channels. Channel bow is induced mainly by fast fluence gradient, which for the most part is managed during the core design process, and also by shadow corrosion, which was observed in the early 2000s with longer cycle lengths and is known to be driven by differential hydrogen concentrations in the Zr alloy.

(v) In CANDU reactors, pressure tube diametral increases due to irradiation induced creep can change fuel channel coolant flows and reduce fuel bundle critical heat flux. Sagging of pressure tubes in older reactors can influence core design configuration (spacing between horizontal fuel channels and reactivity mechanisms — a reactor safety issue), and potentially affect on-power fuel loading reliability.

(h) Mixed cores that include various fuel design features require specific care to avoid damaging incompatibilities between fuel assemblies, particularly in the case of unchanneled fuel designs (e.g. mechanical interaction between fuel assemblies with different grid designs and different grid elevations, thermohydraulic interaction of fuel assemblies exhibiting different pressure drops and hold down force margins, excessive vibration due to cross flow, loss of departure from nucleate boiling (DNB) margins, etc.).

(i) Excessive fuel rod internal pressure due to the fission gas releases during irradiation (and fuel rod free volume reduction) can lead to a pellet–clad gap reopening, but experience has shown that it is not an in-reactor issue within design and operating envelopes but can become a concern in SFP or under dry cask storage conditions if the cladding is highly corroded and the hydrides are radially oriented.

(j) Pellet–cladding interaction assisted by stress corrosion cracking (PCI–SCC) can generate local cladding failures in cases of excessive power transients. Unless the cladding is highly corroded or hydrided with radially oriented hydrides (if the stress field is high enough), pellet–cladding mechanical interaction (PCMI) rarely results in cladding failure.

Specific consideration needs to be given to these known fuel degradation mechanisms for new fuel concepts such as ATFs (see Section 2.3). To ensure proper reliability of ATFs, various additional degradation mechanisms need to be considered and mitigated (e.g. spallation of the coating under normal or accident conditions of the coated claddings, fibre microcracks in SiC based cladding, etc.).

As a matter of fact, limited numbers of leaking rods can be managed safely[7] by the operator but may also be costly[8]. To ensure that the fuel behaves adequately under normal and off-normal conditions, it is important to do the following:

— Develop appropriate tools (i.e. relevant experimental protocols and validated calculation codes) to assess the various fuel designs proposed by the vendors and select the best ones.
— Carefully review international and domestic in-reactor experience for each type of fuel design.
— Implement any fuel design change, or any change in the system (i.e. power uprates, modifications in coolant flow and temperatures or water chemistry), under a well conceived surveillance programme. If the changes are significant, the relevance of irradiating lead test rods or lead test assemblies needs to be considered.
— Consider the use of different types of fuel designs to balance the risk of experiencing a detrimental ‘common failure mode effect’ versus the risk of experiencing deleterious unexpected incompatibility between fuel designs.

Once the fuel is leaking, it is important from a safety, operational and economic point of view that the primary failure does not degenerate into an uncontrolled situation. Some examples are provided below:

— Axial propagation of PCI–SCC cracks (this phenomenon is unlikely in PWRs but can happen in BWRs).
— Secondary degradation of cladding away from the primary defect; a highly corroded/hydride zone appears a few metres above or below the primary leak and can result in a guillotine type fuel rod failure and loss of fuel pellet fragments. Monitoring the evolution of the fission gas release in the coolant, stopping load following operations and shutting down the reactor can usually prevent severe secondary hydriding.
— Fuel–coolant interaction between water sensitive fuel pellets and coolant; uranium nitride, for instance, is among the proposed ATF concepts that can react with water.

---

[7] Provided that the primary circuit activity (which is monitored continuously) is kept below specific required radiochemical limits.

[8] If the operator needs to shut down the plant before the end of the cycle (e.g. when the coolant activity level becomes higher than the authorized limits), some incomplete burnt fuel needs to be discharged, representing a lack of power generation and a significant financial loss. Further, detecting the leaking fuel rod within the fuel assembly is not an easy process and may take considerable time. Searching for the root cause and implementing a solution to fix the problem might also be costly.

In general, using appropriate poolside examination techniques to measure cladding corrosion evolution, fuel rod and assembly dimensional changes and the distribution of fuel assembly distortion within the core, or using specific cleaning tools to remove the crud layers, are good practices to avoid negative surprises in the following fuel cycle and to ensure efficient fuel management strategies.

3.2. PRE-LOADING INSPECTIONS AND VERIFICATION

3.2.1. Fresh fuel

Newly manufactured fuel assemblies are inspected carefully at the fuel manufacturing plant to detect potential manufacturing flaws and the presence of foreign materials and to ensure compliance with utility design specifications. For CANDU reactor fuel bundles, the inspection also includes passage through a ‘bent tube gauge’ to ensure that the bundle will pass freely through a fuel channel that might have sagged due to age.

These fresh fuel inspections are undertaken by the manufacturer and verified by dedicated utility supply chain staff. The fuel assemblies are then packed into shipping casks (shipping boxes for CANDU reactors) and shipped to the operating station.

At the reactor site, all fuel assemblies are inspected again by the operation staff as they are unloaded from the shipping casks. All assemblies are inspected to detect potential manufacturing flaws and handling physical damages and presence of foreign materials within the fuel assembly structure.

Upon receipt, fresh fuel assemblies are stored in dry storage areas that need to be protected against internal flooding to maintain subcriticality margins [17]. The effects of firefighting chemical agents on subcriticality need to be accounted for.

Fresh fuel containing fissionable material coming from reprocessing emits a significant amount of radiation. These reprocessed fuel assemblies need to be handled with additional shielding to limit the exposure of operating personnel.

If the enrichments of the fresh fuel assemblies differ, they need to be stored carefully in appropriate storage racks that are designed for the fuel assembly with the corresponding highest enrichment value or for the most reactive fuel assembly [17].

At CANDU reactor sites, the circumference of each bundle is also measured using a ring tube gauge immediately prior to loading into the new fuel transfer mechanism and subsequent loading into the fuelling machine. This check ensures that the individual fuel rods of the bundle have not become interlocked during transportation and that passage of the bundle through the fuel channel will not be hindered. The manufacturing serial number of each bundle is crosschecked against the shipment record and then entered into the fuel handling database to allow the tracking of each bundle throughout its residence in-reactor.

3.2.2. Irradiated fuel

The core reload pattern is designed ahead of time and usually reviewed by the regulator or through an internal control system set by the fuel vendor or the operator. During LWR outages, irradiated and fresh fuel assemblies are reshuffled in the reactor core cavity. The operator needs to ensure that each fuel assembly is positioned at the right location (i.e. according to the original reload plan).

In cases of significant distortion of some fuel assemblies during irradiation, the reload pattern needs to be adapted accordingly to avoid unwanted interactions between fuel assemblies and to facilitate damage free handling operations.

In all cases, to prevent loss of fuel reliability, fuel assemblies need to be handled in such a way that damage to the fuel assembly structure is avoided.

In CANDU reactors, most irradiated fuel is discharged directly to the irradiated (spent) fuel bays. However, in some circumstances, such as during the pre-equilibrium operation of fresh cores, low burnup irradiated fuels can be reinserted (‘recycled’) into the core from the fuelling machine. This recycle fuelling of fresh cores allows for greater flexibility in flux shaping and the maintenance of safety parameters and promotes optimal fuel utilization. The loading history of the reused fuels is tracked and recorded as appropriate. Recycle fuelling is not permitted for fuel bundles discharged from fuel channels that are suspected of containing a defective fuel bundle. Recycle fuelling is not permitted using fuel bundles that have left the fuelling machine and have been discharged to the fuel bay.

3.3. ON-SITE FUEL INSPECTION AND SERVICES

On-site inspection techniques and devices are continuously being developed to help utilities achieve safe and economical reactor operation through better knowledge of the actual states of the fuel assemblies during their lifetime.[9] These techniques include visual inspection, corrosion characterization and measurement for fuel assemblies, fuel assembly components, fuel channels, reactor control clusters, connectors, etc.

Fuel services aim to support the operator from the introduction of the first lead test fuel assembly to the fuel core surveillance during the entire fuel

[9] Fuel inspection and services include assembly repair for further reuse or depository.

life cycle, including last minute core reload design modification and disposal preparation phases.

These regular inspections also provide valuable in-reactor operating experience, which in turn will be used for the validation of the fuel performance calculation codes used for safety analysis.

Distinctive LWR fuel service technologies that represent the state of the art include the following, for example:

(a) The automated sipping technique (for BWRs, PWRs and WWERs, in-core or mast sipping) to detect defective fuel assemblies rapidly and with a high degree of reliability.
(b) The inspection devices (e.g. a multi-inspection device for PWRs, equipped with a remote controlled camera for BWRs) to carry out fuel assembly or RCCA visual inspections or measurement campaigns (including fuel rod and fuel assembly length, grid width, overall fuel assembly distortion) in a minimum period, without impairing outage critical paths.
(c) The corrosion inspection devices (usually based on eddy currents systems) to monitor in-reactor cladding corrosion.
(d) The crud cleaning devices (usually based on ultrasonic testing techniques) to reduce the impact of crud on the reloaded fuel assemblies.
(e) The gamma scanning devices to characterize fuel assembly and fuel rod radiation (to validate neutronic calculation codes and optimize spent fuel management).
(f) The failed fuel rod detection devices to identify defective fuel rods within a fuel assembly (e.g. with ultrasonic testing devices or with single rod sipping devices).
(g) The fuel assembly repair devices for PWRs and BWRs to remove defective fuel rods rapidly and safely and replace them with dummy rods, either within the original fuel assembly skeleton or within a new one. These tools are also well suited to examining single fuel rods (e.g. for a root cause analysis).
(h) The RCCA inspection devices: the goal is to check the structural integrity of RCCAs (for PWRs) by using the eddy current technique to identify wear marks and absorber swelling.

When repairing the defective fuel assemblies, the radiation level in the primary circuit needs to be kept as low as possible.

Checking the structural integrity of RCCAs on a regular basis improves reactor safety. Using state of the art technologies allows for fewer on-site staff and improves the radiation protection of the plant workers. Examples of fuel inspection and service techniques are described in Annex I.

4. FUEL DESIGN CHANGES TO IMPROVE RELIABILITY AND PERFORMANCE

4.1. OVERALL APPROACH

Fuel design changes are initially assessed and categorized as either major or minor. Minor changes are those that remain within the conditions of the qualified fuel design. Changes beyond this envelope are major changes that require further investigation to characterize their potential impact.

Fuel design changes to improve reliability and performance are verified via a design verification process that includes checking against operational experiences, design analyses using computer codes, out-reactor tests for thermohydraulic characteristics, assessing mechanical response and material properties, on-site inspections and irradiation tests at material test reactors, and independent design reviews by experts who have not participated in the fuel design and testing. In the design analyses, individual changes — although their individual effects on reactor core performance may be insignificant — are all considered together for their interactions with interfacing systems (e.g. core neutronic and thermohydraulic designs, reactor core components, reactor control systems).

The fuel design changes are validated via a design qualification process that includes lead test rods, lead test assemblies or demonstration irradiations. The demonstration irradiation can be initiated with a small number of fuel rods or assemblies with design changes that are loaded at specific locations in the core while considering the thermohydraulic, neutronic or thermomechanical aspects (refer to Section 5.4.3). The demonstration irradiation continues with a gradually increasing number of fuel assemblies with design changes to go from a transition core to a full core. The effect of the mixed core on fuel reliability and performance requires specific attention [12]. These operating experiences can be gathered from other plants with similar characteristics, thereby reducing the qualification process.

As stated in Paragraph 2.7 of SSG-52 [1][10], “The safety assessments shall be commenced at an early point in the design process, with iterations between design activities and confirmatory analytical activities, and shall increase in scope and level of detail as the design programme progresses”. As such, all design changes are assured via confirmation that sufficient margins exist to the

[10] The quoted statement was taken from the statement in Paragraph 4.17 of Safety Standards Series No. SSR-2/1 (Rev.1), Safety of Nuclear Power Plants: Design, as noted in Paragraph 2.7 of SSG-52 [1].

limits of the fuel failures (i.e. safety or design margin) and to the limit of reactor operating conditions (i.e. operating margin). An appropriate balance needs to be established between design and operating margins.

Good practices for margin quantification and management methodology are described in Section 4.2. Margin quantification requires verification of the fuel design and safety limits by design and safety evaluations, which are described in Section 4.3.

4.2. DESIGN AND OPERATING MARGINS QUANTIFICATION AND MANAGEMENT

4.2.1. Definition of design and operating margins

Figure 7 is a practical definition of limits and margins based on the definitions by the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) [18] and the Institute of Nuclear Power Operation (INPO) [19], as well as various interpretations in IAEA publications [20–22] and utilities' practices [23–24].

In deterministic fuel design and safety analysis, the acceptance criteria are represented by one or more physical parameters and are categorized into two levels: high level (radiological) criteria and detailed (derived) technical criteria, according to IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [25]. The radiological criteria are not covered by this publication (detailed (derived) technical criteria are described in the footnote below).[11]

For each given physical parameter, a safety limit (or criterion) is defined with consideration of the uncertainties in the evaluation of failure points (e.g. from test data) and provisions for the unknown phenomena in the approach used to establish the safety limit. These uncertainties and provisions, also known

[11] In SSG-2 (Rev. 1) [25], technical acceptance criteria are placed into three categories:

(a) "Safety criteria: Criteria that relate either directly to the radiological consequences of operational states or accident conditions, or to the integrity of barriers against releases of radioactive material, with due consideration given to maintaining the safety functions.

(b) Design criteria: Design limits for individual structures, systems and components, which are part of the design basis as important preconditions for meeting safety criteria…

(c) Operational criteria: Rules to be followed by the operator during normal operation and anticipated operational occurrences, which provide preconditions for meeting the design criteria and ultimately the safety criteria."

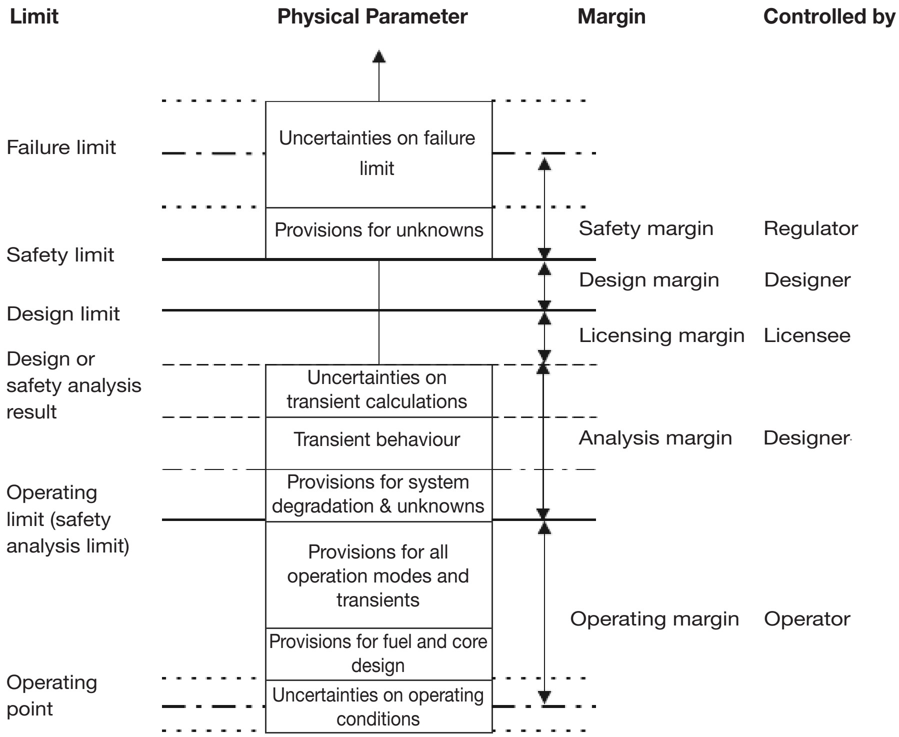


FIG. 7. Definition of limits and margins in fuel design and safety analysis. (Courtesy of J. Zhang from TRACTEBEL.)

as safety margins, are controlled or are imposed by the regulators to ensure an adequate margin for the physical failure limit.

Most of the time, several decoupled conservative safety limits are defined due to the very large uncertainties on failure limits, such as the peak cladding temperature (PCT) and equivalent cladding reacted (ECR) limits for the cladding embrittlement failure during design basis accidents (DBAs).

The design margin may be maintained between the safety limit and the design limit (or criterion). The design margin is controlled and managed by the designer (fuel vendors or utilities) and is defined to ensure that, if the design limit is preserved, the safety limit is met in all applicable plant states. The design limit (or criterion) can be proposed by the designers (fuel vendors or utilities). In practice, for normal operation and AOOs, the design limit is more restrictive than the safety limit, while for DBAs, the design limit is usually identical to the safety limit. A good example of the design limit is the limit on the departure from nucleate boiling ratio (DNBR) for normal operation and AOOs, which is used to prevent fuel cladding failures due to overheating, although the DNB phenomenon is not a failure mechanism.

The licensing margin is defined as the difference between the design limit and the design or safety analysis result (e.g. calculated maximum peak cladding temperature or minimum DNBR) for the related physical parameter during the analysed transient. It is controlled by the designers or operators to maintain a balance between the analysis margins and the operational margins. This margin can be used by the operator to face unexpected events (such as last minute core reload changes, fuel assembly bow, etc.).

The analysis margin consists of provisions (or conservatisms) on the degradation of the plant safety related systems (including the protection systems and the safeguard systems) involved in the analysed transient and unknown phenomena in the analysis (e.g. unanalysed scenarios), transient behaviours and uncertainties. It is controlled by the designer with the agreement of the operators and the regulator. This margin can be used by the designer to justify plant modifications. Margin reassessment is required.

Finally, the operating margin needs to be considered in the safety analyses to ensure that plant operation is flexible and reliable. It includes provisions for all operational modes and transients, equipment degradation (unavailability or aging issues) and the physical data characterizing the reference core and the reference fuel that can be (re)allocated to justify different core and fuel designs or mixed cores. The operating limits (or safety analysis limits) are the bounding values of the physical parameters characterizing the operating condition at the beginning of the transient, including all possible normal operating conditions and transients, as well as the associated uncertainties. The operating margin and operating limits are controlled and managed by the operators. This margin can be used by the operator to justify flexible operation or unexpected operational events, unexpected impairment of some safeguards system (e.g. degraded flow rate of a safety injection pump, degraded set point of protection systems), or degraded performance of the plant system (e.g. steam generator tube plugging ratio).

Distinguishing the roles played by the different fuel assembly components is helpful for better quantification of the available margins each stakeholder can use while performing fuel design and safety evaluations.

4.2.2. Quantification of margins

The quantification of margins depends on advances in the following fields:

— Use of best estimate calculation codes to simulate physical phenomena;

— Understanding of the definition of the design limits based on relevant experiments and analytical analysis;
— Quantification of calculation codes' uncertainties, based on a wide range of experimental data;
— Assessment of plant operating conditions through realistic modelling of the core and fuel behaviours;
— Use of plant surveillance and on-site fuel inspection to verify the accuracy of the design predictions for a set of critical parameters;
— Analysis of plant operating experience and occurrences to provide an empirical basis for the quantification of the margin to failure;
— Removal of unnecessary conservatisms by using best estimate plus uncertainty analysis (BEPU) design methodology.

According to SSG-2 (Rev. 1) [25], different options are considered to perform deterministic safety analyses. These are shown in Table 5 [25], according to the different levels of conservatism associated with the computer codes that have been used, the assumptions on the availability of the reactor core protection and safeguards systems, and the initial and boundary conditions used in the analysis.

Historically, fuel designs were based on semiempirical and conservative approaches. Following the significant improvements of the available analysis tools, the BEPU approach is increasingly used to conduct fuel design or assess fuel design changes.

TABLE 5. EVOLUTION OF DETERMINISTIC SAFETY ANALYSIS APPROACH

Option	Computer code type	Assumptions about systems availability	Type of initial and boundary conditions
Conservative	Conservative	Conservative	Conservative
Combined	Best estimate	Conservative	Conservative
BEPU	Best estimate	Conservative	Best estimate Partly most unfavourable conditions
Realistic	Best estimate	Best estimate	Best estimate

The BEPU approach considers the uncertainties on operating conditions and on analysis conditions, including the following:

- Overall code or individual modelling uncertainties (basic equation and closure laws);
- Representation (nodalization) uncertainties, numerical inadequacies;
- Uncertainties due to the scaling issues;
- User effects, computer/compiler effects;
- Input data uncertainties for the analysis of an individual event.

The BEPU methodology is the appropriate approach to quantify the available margins, as the uncertainties are better quantified, but it requires high level verification, validation and uncertainty quantification tools for the computer codes that are used, and considerable effort to develop the right methodology. For industry applications, a graded BEPU approach needs to be applied to quantify the margins efficiently and cost effectively in the safety analysis for DBA conditions to ensure compliance with the safety acceptance criteria and design extension conditions to reduce the potential cliff edge effects [26].

A good balance between safety, design and operating margins should be considered in the fuel design and safety evaluation process.

4.2.3. Management of margins

Effective management of margins by operators is important to improve fuel reliability and performance during operation. The margin management process recommended by INPO comprises the following key steps [19]:

- Understanding margins;
- Identifying margin loss;
- Evaluating consequences of margin loss;
- Prioritizing modifications resulting in margin loss;
- Recovering margins;
- Identifying roles and responsibilities;
- Defining processes;
- Performing periodic assessment and communications.

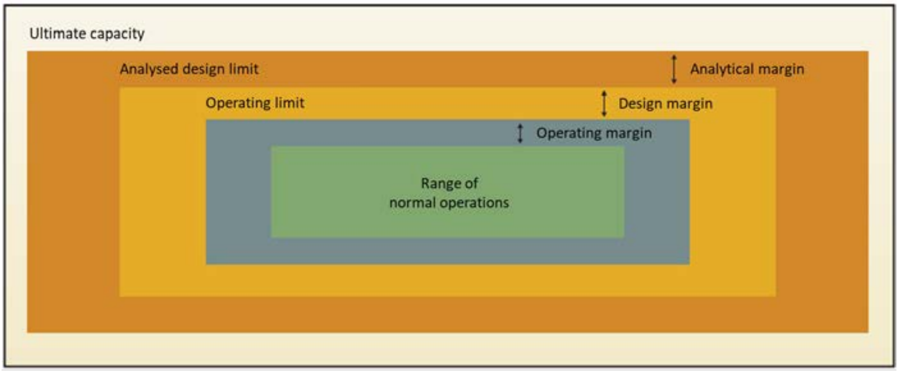


FIG. 8. The margin model used in the margin management process. (Reproduced from Ref. [19] with permission from INPO.)

The margin management process is based on the following margin model[12] proposed by INPO [19] and is presented in Fig. 8.

Margin management ensures that the fuel is operated as designed and licensed in accordance with the following requirements:

— The fuel designer needs to provide reliable, robust and high performance products with adequate design, operating and analytical margins to meet safety, flexibility and economic requirements.
— The core designer needs to provide flexible and cost efficient cycle specific reload core designs that are within analysed design limits.
— The plant operator needs to anticipate and manage the operation of the fuel within operating limits to ensure safety, reliability and performance.

The definition and determination of margins can be found in:

— The updated final safety analysis report (SAR);
— The technical specifications and bases;
— The engineering calculation reports;

[12] The margin model in Fig. 8 is consistent with that in Fig. 7, but it is somewhat simplified to be easily utilized by operators, while the margin model in Fig. 7 is a better fit for designers. Note that:
— The design margin in Fig. 8 is equivalent to the analysis margin plus licensing margin in Fig. 7.
— The analytical margin in Fig. 8 is equivalent to the safety and design margin in Fig. 7.

— The design basis documentation.

The margin database is established by the core designer or plant operator by:

— Documenting the operating and design limits and margins;
— Identifying margins when new plant configurations are implemented;
— Ensuring that the database is up to date.

To monitor changes in margin, they need to be prioritized by:

— Being aware of operating and design requirements;
— Knowing the actual conditions of the plant;
— Identifying the critical design or safety parameters;
— Assessing the potential impact of operations;
— Evaluating the aggregated impact of margin changes;
— Assessing the associated consequences for plant safety and reliability.

In case of insufficient design and operating margins due to fuel or plant modifications, these modifications need to be prioritized commensurate with their potential effects on plant safety and reliability. This prioritization is typically based on cross-disciplinary reviews. The management should review and approve the prioritization list and provide adequate support. If the consequences of margin loss are acceptable, a comprehensive rationale needs to be documented to close the issue.

If the consequences of margin loss are unacceptable, the following actions are required:

— Development of an action plan;
— Evaluation of the potential consequences of exceeding the limits and associated costs;
— Assessment of costs and benefits associated with margin recovery.

Some of the ways to recover margin loss are as follows:

— Recovery by using analytical approaches, for example, by changing assumptions or methods to remove conservatisms in analysis, recalculating the safety parameters with more rigorous or physical models to reduce uncertainties, justifying margins with alternative provisions, or trading off provisions on a less limiting parameter with provisions on a more limiting one;

- Recovery by changing design, for example, changing fuel design characteristics or modifying plant operating conditions;
- Recovery by changing maintenance and operating procedures, for example, by corrective or preventative maintenance, testing to identify actual performance, or operating the plant differently.

Roles and responsibilities should be identified in the margin management process:

(a) Maintenance should identify and document unexpected or excessive fuel degradation that was not anticipated.
(b) Operations should identify operator workarounds, procedures that are too restrictive, or adverse changes in the structures, systems and components or barriers.
(c) Engineering owns and is responsible for the process and should take the following actions:
   (i) Verify, validate and maintain assumptions.
   (ii) Identify and document margin changes associated with core and fuel designs and safety analyses.
   (iii) Determine new operating limits for equipment.
   (iv) Evaluate the margin change issues.
   (v) Champion resolution of the issues.

4.3. GOOD PRACTICES FOR FUEL DESIGN CHANGE VERIFICATION

Based on operating experience and the results of relevant research and development programmes, fuel design or design change verification is realized quantitatively by verifying the specific fuel design and safety limits for all applicable plant states in fuel design and safety analysis.

The fuel design and safety analysis process to meet the above safety guide recommendations consists of the following:

(a) Definition and determination of fuel design and safety limits;
(b) Fuel design code verification and validation, and uncertainty quantification;
(c) Verification of fuel design and safety limits.

This section provides good practice guidelines for the above processes.

4.3.1. Definition and determination of fuel design and safety limits

In SSG-52 [1], Paragraph 3.65 states that, "Fuel design limits should be established based on all physical, chemical and mechanical phenomena that affect the performance of fuel rods and fuel assemblies for all applicable plant states." The term 'design limits' in SSG-52 [1] is interpreted to be equivalent to the commonly used term 'design criteria' (see footnote 13) and thus these two terms are used interchangeably in the context of this publication. The terms 'safety limits' and 'safety criteria' are also used interchangeably.

Fuel design and safety limits are to ensure that the nuclear fuel performs satisfactorily in the reactor core throughout its design lifetime and for all applicable plant states. Fuel suppliers use validated fuel performance calculations codes, in-pile and out-of-pile experiments, and in-reactor performance data to demonstrate compliance with these limits.

The fuel design and safety limits to be verified in the design and safety analysis can be summarized as follows:

— For operational states including normal operation and AOOs, the probability of failure of the fuel cladding, resulting from the DNB or from any other failure mechanisms, needs to be insignificant.
— For accident conditions including DBAs and design extension conditions without significant fuel degradation (the latter is outside the scope of this publication), the fuel failure needs to be limited for each type of accident to ensure a coolable geometry. Energetic dispersal of fuel needs to be prevented to avoid reactivity initiated accidents (RIAs).

Details for fuel design and safety limits, as well as their applicability to ATF, are described in Appendix I.

4.3.2. Code verification, validation and uncertainty quantification

Requirement 18 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [27], states that "**Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation.**"

Paragraph 5.1 of SSG-2 (Rev. 1) [25] states that:

> "The models and methods used in the computer codes for deterministic safety analysis should be appropriate and adequate for the purpose. The extent of the validation and verification necessary and the means for achieving it should depend on the type of application and the purpose of the analysis."

In SSG-52 [1], Paragraph 3.64 states that:

> "Fuel performance analysis codes can be used to analyse and interpret the data from power-ramp tests and to determine a failure threshold. The parameter used to define this threshold is usually the maximum cladding stress but the strain energy density can also be used. These same fuel performance analysis codes can be used to assess risk factors that cause this type of stress corrosion cracking of fuel rods in the reactor core and to define adequate guidelines to avoid it."

Paragraph 3.146 of SSG-52 [1] states that:

> "Multidimensional and multiscale physics codes and system thermohydraulic codes should preferentially be used for realistic analysis of the reactor core for all applicable plant states. Uncertainties should be adequately incorporated into the analyses…[13]".

To verify fuel design and safety limits, computer codes need to be used to predict the neutronic, thermohydraulic and thermomechanical behaviours of fuel rods and fuel assemblies during all applicable plant states. These computer codes need to be verified and validated to ensure that they are suitable for the intended application and need to be approved for use. The uncertainties of the key models need to be quantified based on experimental and operational data.

This section focuses on the technical aspects of the code development, verification and validation, and uncertainty quantification. The quality assurance aspect is described in Section 5.7.

4.3.2.1. Types of computer codes

To simulate the fuel behaviour during all applicable plant states, four categories of computer codes are generally needed:

(a) Reactor physics codes;
(b) System and subchannel thermohydraulics codes;
(c) Fuel performance codes;
(d) Structural mechanical codes and models.

[13] See SSG-2 (Rev. 1) [25] for details.

Traditionally, computer codes in these categories were used separately in a sequential decoupled manner, while the current trend is to use them in a coupled multiphysics code package or environment.

4.3.2.2. Code verification

Code verification is the process of examining whether computer codes or simulation tools accurately represent mathematical models that have been applied and the associated solution methods. It can be further divided into numerical algorithm verification (i.e. solution verification) and software quality assurance (or code verification) [28].

Solution verification focuses on the assessment of the mathematical correctness of the simulation models applied and their solutions by comparison with known solutions (i.e. What is the numerical error?).

Code verification focuses on identifying and removing coding errors in the source code and numerical algorithms and improving the software using software quality assurance practices. It aims to answer the question of whether there is any bug in the code. Typical methods and good practices used to verify computer programs can be found in Refs [29, 30].

4.3.2.3. Code validation

Code validation is the process of examining how accurately computer codes or simulation models represent reality. It focuses on the assessment of the prediction accuracy of the code or simulation model through comparison with experimental data [28]. It aims to answer the question of how well code calculation outputs represent physical reality.

The main objectives of validation are to achieve the following:

— Assess the adequacy and applicability of the models employed in the codes;
— Demonstrate the capabilities and limits of the codes;
— Determine the accuracy of the codes.

These objectives of validation are achieved typically by comparing code predictions with the following:

— Experimental data;
— Plant commissioning or operating data, where available;
— Solutions to standard or benchmark problems;
— Closed mathematical solutions;
— Results of another validated code.

The typical steps and good practices applied to computer code validation can be found in Refs [29–31].

The quantification of the accuracy of the simulation model can be expressed by the accuracy of the quantities of interest, which needs to be related to the consistency between the predicted quantity of interest value and the experimentally measured quantity of interest value. It should consider all sources of uncertainties, including the quantification of model input uncertainties by inverse methods [32, 33].

It is noted that comparisons of code predictions with ‘solutions to standard or benchmark problems’ (see item above) that also include closed mathematical solutions is acceptable for validation purposes, but they need to be followed by other types of comparisons.

In addition to the validation performed by the code developer, it is also important for the code user to perform independent validation [31]. Most widely used computer codes have gone through international code assessment and independent validation procedures, with system thermohydraulic codes having received the most attention [34]. Other types of codes have also been validated, but to a lesser degree [35–37]. Extensive code validation experiment matrices are available [38–41]. Many experiments contained in these matrices have been used in the validation of some codes [34].

Validation exercises are sometimes conducted by participating in international standard problems or benchmarks [42–46]. In these benchmarks, selected experimental data have been carefully reviewed and then integrated into the International Fuel Performance Experiments database at the NEA [47] before being used by the participants to compare the various fuel performance code simulations. Parallel efforts have also been made at the NEA on PCMI [48–50], loss of coolant accidents (LOCAs) [51, 52] and RIAs [53–57].

All of these international benchmarks allow for better validation of the fuel performance codes that are used for fuel design and fuel safety evaluations.

4.3.2.4. Uncertainty quantification

Uncertainty quantification can be defined as the process of characterizing all relevant uncertainties in a model and quantifying their effect on a figure of merit. It aims to answer the question of how large the uncertainty in the calculation result is.

Computer code uncertainty mainly results from the following:

— Model input uncertainty that includes the model parameters of closure laws as well as geometry, boundary and initial conditions;

— Model form uncertainty that includes simplifications in all modelling assumptions or approaches (conceptualizations, abstractions, approximations, mathematical formulations) employed in a mathematical model to model an actual physical phenomenon, and a lack of knowledge regarding that phenomenon;
— Numerical approximation errors, such as space–time discretization or iterative convergence errors.

From a methodological point of view, uncertainty quantification can be split into two main topics that are dealt with using different approaches. The first approach assumes that all input uncertainties are known (e.g. by expert judgement or inverse uncertainty quantification). It is intended to estimate the impact of input uncertainties on output uncertainties and is referred to as 'uncertainty analysis'. It is based on 'input uncertainty forward propagation' [58]. The second approach focuses on input uncertainty quantification. It is intended to derive model input uncertainty by taking account of the inverse propagation of the difference between code prediction and experiment [59].

Input uncertainty forward propagation has been widely used in the BEPU methodology for thermohydraulic safety analysis since the 1990s [60, 61] and gradually extended to fuel design and multiphysics safety analysis [62–64].

Inverse quantification of model input uncertainties is essential for the BEPU methodologies. A general framework called SAPIUM[14] has been developed to provide good practice guidance for quantification of thermohydraulic code model input uncertainty [33].

A comparison of available input uncertainty quantification methods was made in order to provide good guidance for an appropriate selection of input uncertainty quantification methods [65]. Similar methodologies have also been developed for quantification of model input uncertainty for fuel rod codes and multiphysics codes (see [66], for example).

4.3.3. Fuel design and safety evaluation methods

4.3.3.1. General requirements

Fuel design bases need to address fuel damage or failure mechanisms and provide acceptable levels of important performance parameters within which such fuel damage or failures are prevented.

The fuel performance for all applicable plant states needs to be evaluated to determine whether all design and safety limits are met. Fuel assembly

[14] Systematic Approach for Input Uncertainty Quantification Methodology

components are reviewed both as separate components and as integrated bodies (i.e. fuel rods and fuel assemblies).

Fuel design evolutions, new operating envelopes (e.g. maximum rod burnup and power) and changes to fuel component materials require new fuel design and safety evaluations to verify that the existing design and safety limits and computer codes as well as the methods used for such evaluations remain applicable for the new design in all applicable plant states. If the established design and safety limits do not apply, new limits need to be established based on appropriate data and theoretical considerations.

Fuel design and safety evaluation needs to account for operational experience, direct comparisons with experimental data, analyses (using fuel performance codes) and other information. The codes used need to be verified and validated, audited and approved by the safety authorities. An example of the specifications on the scope and methods of fuel design and safety evaluation can be found in Ref. [67]. Appropriate provisions need to be made to treat mixed core effects [68].

4.3.3.2. Generation of neutronic design input data

The design and safety evaluation needs to consider a reference in-core fuel management (ICFM) strategy that covers current and anticipated future reload designs. The bounding design power histories are provided by the fuel vendors (or design and analysis organizations for CANDU reactors), using their methodology and based on a reference equilibrium cycle that is representative of the current cycles and foreseen future cycles such that no cycle specific fuel rod design verification is needed.

During the cycle specific reload safety evaluation, only the bounding power histories are verified, and specific fuel rod design verification can be performed only exceptionally if the bounding design power histories cannot be verified.

In the case of major modifications to the ICFM strategy, such as an extended cycle length (e.g. up to 18 months or 24 months), an increased fuel rod burnup limit (e.g. above 62 GWd/t or 75 GWd/t) or significantly increased enrichment (above 5%), a new fuel design and safety evaluation is required for a new reference equilibrium cycle that is representative of the foreseen new cycles in order to demonstrate good fuel rod behaviours with the new ICFM strategy.

The so defined reference equilibrium cycle for fuel design needs to be calculated with approved three dimensional neutronic models. These calculations provide fuel rod power histories for baseload operation. The rod power histories used in the fuel rod design verification are to be based on a bounding fuel rod or rod duty that envelops the entire irradiation in the core. These bounding design power histories define the maximum steady state power for several thermomechanical design criteria (e.g. rod internal pressure, corrosion).

For neutronic transient analysis, the selected transients, such as uncontrolled boron dilution, excessive load increase and RCCA withdrawal at power, are simulated in consideration of the nuclear steam supply system and core characteristics (fuel management). Each transient is simulated at several points in the cycle, including the beginning, middle and end of each irradiation cycle. The attained maximum transient powers are provided for verification of the fuel rod design criteria under AOO transients (e.g. fuel and cladding temperatures, cladding stress and strain).

Core design inputs provided for the fuel rod design verification and safety evaluation need to include at least the following data:

- Design parameters;
- Bounding design power histories (for normal operation);
- Axial power distributions;
- Radial power distributions;
- Fast flux/fluence;
- AOO transient powers (maximum local power observed in the core and/or maximum local power variation during an AOO transient, as a function of local burnup);
- Core daily fuelling strategy and channel fuelling rates (for CANDU reactors).

4.3.3.3. Verification of fuel design and safety limits

Verification of fuel design and safety limits is to be performed in the fuel design and safety evaluation, covering the following scope:

- Fuel rod thermomechanical design (for operational conditions);
- Fuel rod safety evaluation (for accident conditions);
- Fuel assembly mechanical design verification;
- Thermohydraulic design verification;
- Neutronic design verification.

The primary function of the fuel rod is the generation and transfer of heat to the reactor coolant. In the course of heat generation via fission reactions, fission products are produced in the fuel. Another function of the fuel rod is to confine fission products within the fuel rod.

To achieve these functions, the fuel rod design needs to maintain its structural integrity for plant operational states (normal operation and AOOs).

Fuel cooling and control rod insertion need to remain possible during and after bounding DBA events (i.e. RIAs and LOCAs).

The detailed fuel rod design evaluation considers pellet diameter and density, cladding composition, cladding diameter, cladding thickness, pellet to cladding diametral gap, plenum size, filling gas composition and rod pressurization level as input parameters. The fuel design evaluation process needs to consider the effects of the specific core cavity geometry, the local power variations, coolant temperature, pressure, chemistry and flow variations occurring during operational states. The design also considers the irradiation effects on fuel rod material properties.

To ensure fuel rod integrity during normal operation and AOOs, the design needs to be such that fuel damage due to excessive gas pressure, excessive fuel temperatures, excessive loads and excessive stresses and strains in the cladding (e.g. PCI–SCC) is precluded. This is achieved by verifying that all fuel rod design criteria are satisfied.

To ensure fuel rod integrity during accident conditions (e.g. RIAs and LOCAs), the design needs to preclude significant fuel failures due to excessive cladding deformation, burst, high temperature oxidation and embrittlement, fuel melting and dispersal. This is achieved by verifying that all design and safety criteria are satisfied.

The fuel rod design evaluation needs to be complemented by mechanical, thermohydraulic and neutronic evaluations of the fuel assembly to ensure fuel assembly integrity during operational and accident conditions.

Fuel design and safety evaluation is to be performed using one or a combination of the following methods:

(a) Conservative (or bounding) approach — bounding uncertainties on the initial parameters, on the boundary conditions and on the key models of the calculation codes need to be accounted for.
(b) Statistical approach — uncertainties on the relevant initial parameters, on the boundary conditions and on the key models of the calculation codes need to be considered through a statistical combination of their variabilities in parametric sensitivity studies (e.g. root of mean square).
(c) BEPU approach — uncertainties on the initial parameters, boundary conditions and key models used in the calculation codes are combined statistically, using the input uncertainty propagation methods.

Details on the verification of fuel design and safety limits are described in Appendix II.

4.3.3.4. *Cycle specific fuel design verification and safety evaluation*

As stated earlier, the ‘generic’ fuel design verification and safety evaluation are performed on the basis of the bounding design power histories for a reference equilibrium cycle. This means that sufficient margins are provided in the generic fuel design verification and safety evaluation such that they will remain valid and applicable for ‘specific’ cycles that are covered by the reference equilibrium cycle.

During the cycle specific reload safety evaluation, the verification is limited to the bounding power histories. If the latter cannot be verified, specific fuel design verification and safety evaluation are performed, but such an approach should remain exceptional.

In the case of insufficient margins or a specific issue, the vendor or the operator performs a cycle specific verification regarding certain key fuel design and safety criteria (e.g. rod internal pressure, PCI, corrosion). This complementary verification has a beneficial effect on fuel reliability.

4.3.3.5. *Loading pattern fuel reliability risk assessment*

To assess the effect of loading pattern design on fuel reliability, appropriate tools need to be developed to address specific issues [69–71], such as the following:

— Axial offset anomaly;
— Axial xenon stability;
— Crud induced localized corrosion and crud induced power shift;
— Damaged fuel or debris failure;
— Pellet–cladding mechanical interaction due to missed pellet surfaces;
— Potential for PCI–SCC;
— GTRF wear;
— Fuel assembly bow;
— Incomplete control rod insertion, etc.

In CANDU reactors, initial fuel loading of a new or refurbished reactor is followed by the ‘pre-equilibrium’ period, which lasts many months and during which the reactor is not fuelled. The anticipated power transients in this initial core, the ‘plutonium peak’, are managed by inclusion in the initial fuel load of depleted fuel bundles in selected core locations and by the introduction of a small number of anticipated reference core nominal flux shapes into the reactor regulating system computer program. There are no fuelling power ramps during this pre-equilibrium period, and the fuel bundles are not at risk due to PCI–SCC type failures. This pre-equilibrium period is followed by a very long period of equilibrium operation, during which the reactor is fuelled almost daily for the balance of the reactor's operating life. A small number of fuel channels are selected daily by the fuelling engineer for the introduction of a few unirradiated 'fresh' fuel bundles and the simultaneous discharge of irradiated fuel bundles.

5. QUALITY MANAGEMENT OF FUEL DESIGN AND MANUFACTURING

5.1. PREFACE

The objective of quality management is to confirm and ensure that fuel design complies with the safety and operational requirements that are documented in the design specifications and that fuel products are manufactured according to the approved fuel design specifications. The design process and related analyses are described in Section 4. Successive activities to manufacture fuel rod and assembly products are described in Appendix III.

5.2. GENERAL CONCEPTS OF QUALITY, QUALITY CONTROL, QUALITY ASSURANCE AND MANAGEMENT SYSTEM

Quality is defined as the "degree to which a set of inherent characteristics of an object fulfils requirements" according to International Organization for Standardization (ISO) [72].

The concept of quality as underpinning safety and reliability has been evolving for a long time, as illustrated in Fig. 9, and has been adopted to the design and manufacture of the structures, systems and components of the nuclear power plants.

Figure 9 shows the evolution of quality requirements from the 1970s to today, starting from quality control (QC) through quality assurance (QA) and quality management system (QMS) to the integrated management system.

QC is defined as "Part of quality management intended to verify that *structures, systems and components* correspond to predetermined *requirements*" [73]. Therefore, QC focuses on verifying or demonstrating that the specified requirements have been achieved (conformity to requirements) via activities such as sampling, measuring, inspecting, testing, recording, witnessing and auditing.

FIG. 9. Evolution of concepts of quality. (Courtesy of I. Gorokhov from OKB GIDROPRESS.)

QA is defined as "The function of a *management system* that provides confidence that specified *requirements* will be fulfilled" [73]. Therefore, QA focuses on providing confidence that requirements are achieved via systematic processes, including audits, training, management of human resources and management of documentation and records, changes, procedures, working methods, supply chain, etc. Quality assurance requires planning before any work takes place and implementation of the above mentioned processes.

Management system or quality management system (QMS) is defined as "A set of interrelated or interacting elements (*system*) for establishing policies and objectives and enabling the objectives to be achieved in an efficient and effective manner" [73]. Therefore, the QMS describes the activities, responsibilities, processes and procedures so that an organization can achieve its objectives in a coherent manner.

Integrated management system is defined as "A single coherent *management system* for *facilities and activities* in which all the component parts of an organization are integrated to enable the organization's objectives to be achieved" [73]. The integrated management system describes the processes, procedures, instructions and planned actions necessary to provide adequate confidence that all these requirements are satisfied. The integrated management system ensures that other considerations, such as health, environmental, security and economic requirements, are consistent with safety requirements (see Fig. 10).

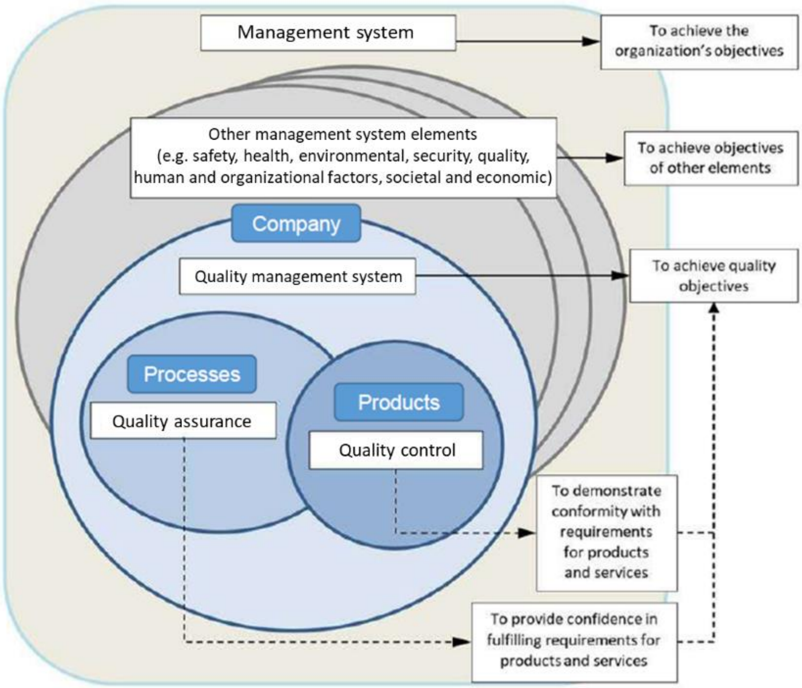


FIG. 10. A high level illustration of quality control, quality assurance and the management system of nuclear facilities. (Reproduced from Ref. [74] with permission from the IAEA.)

5.3. QUALITY MANAGEMENT SYSTEM DOCUMENTATION APPLICABLE TO NUCLEAR FACILITIES AND ACTIVITIES

Several types of QMS documentation are prepared to provide strategic direction and implementation guides for the function (as shown in Fig. 11). QMS documentation is prepared to be consistent with requirements of international standards (e.g. the IAEA's GSR Part 2 [75], ISO 9001:2015 [76]), national legal and regulatory requirements and customers' requirements.

Quality policy is defined by the senior management of each organization [74]. The quality policy needs to be aligned with the organization's vision and missions. The quality policy provides the quality objectives of the organization. The quality policy can be included in the quality manual or in the quality assurance programme documents. The policy needs to be reviewed periodically and communicated to all employees to ensure that all employees are

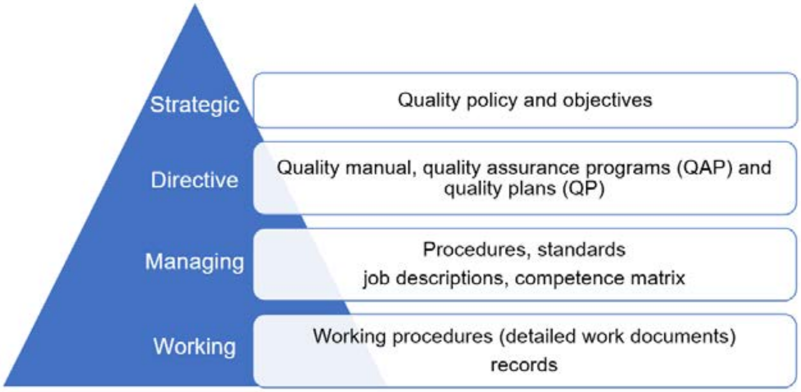


FIG. 11. Quality management system (QMS) documentation hierarchy. (Courtesy of I. Gorokhov from OKB GIDROPRESS.)

aligned on concrete and consistent quality objectives to satisfy customers and other stakeholders.

The quality manual or quality assurance programme defines organizational arrangements and technical measures relative to quality assurance. A quality plan describes how an organization, typically a vendor or supplier, provides the intended processes, products or services with reference to the quality manual or to the procedure documents. A quality plan is also used for monitoring conformity with specified requirements (which is also known as the inspection and testing plan).

Important requirements of the quality manual that are applicable to fuel design and manufacturing are described in Sections 5.4–5.7. Procedures describe how to carry out activities or processes. They include detailed work instructions dedicated to elementary tasks. Irrespective of the level of detail of the procedures, the following aspects need to be considered as applicable:

— Defining the needs of the organization and of its customers and suppliers;
— Describing the processes related to the required activities (e.g. using flow charts);
— Establishing tasks and responsibilities with the organization (what, who, why, when, where, how);
— Describing the control of processes;
— Defining the resources required to accomplish the activities (in terms of personnel, training, equipment, materials);
— Defining the documentation associated with the required activities;
— Defining the input and output of the processes;
— Defining the measurements to be taken.

5.4. GENERAL REQUIREMENTS APPLICABLE TO FUEL DESIGN AND MANUFACTURING

5.4.1. Organizational activities

The governance of and the organizations within the fuel design and manufacturing company need to be clearly defined on several points:

(a) All the activities of the company need to be described by processes that clearly underline the added value of each contributor in the end products.
(b) The responsibilities need to be clearly defined for each process. In particular, the decision-making process needs to be clearly defined among the organizations in cases of non-routine situations. In particular, the company needs to ensure, in accordance with State regulations, that each product is controlled in an independent manner by the quality team to avoid any nonconformance in the quality of the product. Quality teams need to be empowered by the company to be able to stop production if necessary.
(c) The governance of the company needs to make sure that the decision making process is organized at the right level and that the information flow, bottom up and top down, is precise and consistent.
(d) Processes, organization and governance need to be adapted as needed to the situation, the customers' requirements and the business priorities.

The QMS needs to be built on the basis of feedback from experience — incident collection and non-conformance management. Preventive actions based on risk analysis are performed to maintain high quality. Each event needs to be analysed to identify weak points in the quality systems and to put corrective actions in place to prevent reoccurrence of the event. Risk analysis is performed on a regular basis in all domains. The aim of this analysis is to prevent incidents by means of defence in depth actions and to provide security of supply for customers.

Note that, in some organizations, fuel design is a function provided by the fuel manufacturer, while in others, such as CANDU reactors, both fuel design and licensing are the responsibility of the operating utility. QMS for fuel design applies to either manufacturers or operating utilities.

5.4.2. Human resources management

Since the high quality of end products depends on all the employees in the organization, each of them needs to have a clear understanding of the importance and underlying purposes of their contributions.

Human performance and behaviour are keys to reaching and maintaining a high level of quality. Good performance necessitates highly trained, qualified and motivated personnel.

5.4.2.1. Training and education

All employees need to be trained so that they are eventually qualified to accomplish their tasks independently. Training needs to contain theoretical and practical components. The organization needs to ensure that workers maintain their qualification for each task.

The ability to manage expertise in each relevant domain in the organization (e.g. engineering design, modelling, welding and sintering workstations) is a key success factor for high quality end products. This can be achieved only through the cooperation of all of the departments and individuals involved.

5.4.2.2. Maintenance and human factors

The key role of management is to involve all of the employees and the subcontractors in the global mindset regarding quality. Therefore, the management needs to pay attention to the following three points:

(a) The organization needs to guarantee excellence in production and will need to anticipate and provide guildelines for future changes.
(b) A safety culture needs to be implemented and maintained by the organization.
(c) The strategy and the objectives of the company need to be shared among all employees.

The motivation of the employees is the most important guarantor for high quality in a company. Therefore, the management system should provide the necessary means for generating and especially maintaining the motivation of employees, which includes a no-blame culture.

5.4.2.3. Training and certification of inspection personnel

It should be noted that the specific activities in nuclear fuel quality control are interrelated. The requirement for highly qualified and trained personnel in QA/QC is well justified. It is extremely important that these people are motivated and dedicated to their tasks. They also need appropriate experience regarding the process steps they are monitoring and the control methods they are applying.

5.4.3. Documentation management

5.4.3.1. Documentation policy

Documenting and recording data is a challenge in nuclear fuel technology processes. There is a large amount of data that needs to be handled and stored and be easily accessible for a long period. For example, to evaluate fuel reliability and performance, it is extremely important to pay attention to the database architecture, regarding collection, maintenance and retrieval of manufacturing data. Traceability is important for root cause analysis in cases of non-conformance.

Care needs to be taken in establishing data collection systems to ensure that the product lifetime is consistent with the data storage lifetime. As such, the storage of quality control data required for long lived reactor components should be longer than for shorter lived fuel assemblies. However, if post-irradiation examinations of the fuel are performed, fuel manufacturing data need to remain accessible accordingly. Storage time and storage conditions for data are to be specified clearly for each group of products and should be consistent with the lifetime of the products.

Most documents today are created electronically. To expedite their handling, the fuel supplier requires a system of electronic inspection and approval. Distribution of documents to the customer in electronic form would also shorten the total time needed for their approval by the customer and the licensing authority.

5.4.3.2. Identification and coding of products

The following features are proposed:

(a) A route or travel card is attached to the product and travels along with the product, and all of the processing details are entered at each workstation. It contains the following data:
   (i) Product identity;
   (ii) Fabrication route;
   (iii) Stages of fabrication and inspection status;
   (iv) Deviations observed at various stages.
(b) A quality control label, indicating product lot number, stage and status of inspection and quantity. Alternatively, the quality control details and comments can be included.
(c) Fuel assembly components, which should be engraved with numbers wherever possible.

(d) Colour coding of the containers and handling equipment.

5.4.3.3. Handling storage and delivery

The quality of the manufactured products depends on the way they are handled. It is necessary to identify and prevent sources of contamination, defects and damage due to handling at each stage of production. Handling of raw materials, tools and consumables is of importance. All parts need to be stored properly in containers or on shelves for further processing. To prevent damage or deterioration, special handling and transport equipment for fuel tubes, parts, bundles, etc. are designed and used.

Necessary documentation includes:

— Fuel supplier’s quality certificate confirming that the fuel has been manufactured in accordance with approved specifications;
— Traceability files for each component of any product (i.e. manufacturing records).

If not delivered together with the fuel assemblies, at least the following documents (or their copies) need to be readily available for inspection by the utility at the fuel supplier’s premises:

— Material certificates for all component materials, including welding constitutive materials;
— Manufacturing certificates for fuel assembly components, subassemblies and completed fuel assemblies;
— Documents showing the results of inspections made in accordance with the inspection plans.

5.4.4. Non-conformance management

Non-conformance is a failure to meet one or more of the existing requirements specified in the quality manual or associated documents. If it affects safety and operation staff, non-conformance needs to be resolved based on comprehensive root cause analyses and corrective actions. Implementation of corrective actions needs to be followed up for continuous quality improvements.

IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [77], provides guidance on processes for non-conformances and corrective actions.

5.4.5. Assessments and audits

A quality assurance audit is an independent process to ensure that design and products meet expected quality levels. GS-G-3.1, Application of the Management System for Facilities and Activities [77], provides guidance on quality assurance audits. Audits are performed on processes, products or QMS.

All levels of management are required to carry out self-assessments on a regular basis to evaluate the performance of work and improvement to safety culture. The assessments need to be complemented by independent assessments undertaken by an organizational unit charged with this task, or by other independent parties external to the area or work being assessed. Senior management is required to review the results of assessments, make and record decisions and take the follow-up actions deemed appropriate.

5.4.6. Additional requirements — customer surveillance of fuel assembly suppliers

Nuclear fuel is a critical component involved in the safety of the reactor core; its reliability is essential to ensure efficient operation of the nuclear power plant. Consequently, a customer is responsible for ensuring that all safety requirements are fulfilled in accordance with the legal requirements of the State where the products are delivered and that the reliability and quality of the products meet these expectations.

To ensure appropriate behaviour of the fuel, customers need to perform independent surveillance of the fuel supplier's activities. This surveillance cannot interfere with the supplier's controls and internal decision process so as not to jeopardize the supplier's responsibility. This surveillance is based on a cooperative mindset. Customer surveillance covers all activities from design to delivery of the product, including manufacturing.

Surveillance of the supplier includes the following:

- Review and approval of documents (i.e. qualification and fabrication documentation) prior to use by the supplier;
- Independent cross-checking of data or information (e.g. product data, design or manufacturing document, numerical analysis calculation);
- Monitoring of ongoing manufacturing processes in the workshop;
- Audits of the QMS;
- Technical inspections focused on topics that are important for safety or quality (e.g. welding process, gamma scanning control, design code, test facilities).

Customer surveillance is described in documents covered by the QMS of the supplier. The outcomes of the customer surveillance are recorded so that they can be inspected by the regulator or any person designated by the final user, if required by the customer's QMS.

Customer surveillance for manufacturing requires verifications performed in the supplier's and sub-suppliers' factories on the basis of a sampling approach. The list of subcontractors under customer surveillance is decided in accordance with legal requirements and expected final product quality, considering the risks associated with the subcontractors' activities.

On the basis of the evaluation of the manufacturing process risks that the customer intends to mitigate, the following aspects are to be treated:

— Manufacturing documents of the supplier whose conformity to legal, quality, or contractual requirements are reviewed (drawings, specifications, manufacturing quality plans, inspection programmes and reports, technical notes, non-conformance treatment reports, design or safety calculation notes);
— Activities involved in a manufacturing process related to product characteristics important for the fuel assembly safety and reliability on which surveillance is defined (including the appropriate level of surveillance concerning qualifications of the processes).

Customer surveillance describes documentation of non-conformances and deviations to ensure that the suppliers provide an adequate level of information to enable the fuel assembly user to handle a potential issue in a safe manner and that the legal requirements concerning information supplied to the regulators are fulfilled. In any case, the supplier can support the utility's surveillance through the following actions:

(a) Compiling and providing the customer with an up to date manufacturing schedule for all manufacturing stages. This is important for the customer to schedule surveillance visits during manufacturing stages of interest.
(b) Providing easy access to all manufacturing facilities and manufacturing documents.
(c) Making sure that the customer's requirements regarding manufacturing of the fuel assemblies and their components are correctly understood and considered in the documentation at the manufacturing plants.
(d) Making practical arrangements for the customer's visits to the sub-supplier's plant.
(e) Accompanying and supporting the customer during manufacturing surveillance visits.

5.4.7. Continuous improvement process

Continuous improvement aims to enhance the effective application of the QMS, which may include:

- — A process to continue improvement of the system;
- — A process to ensure conformity to customer and applicable statutory and regulatory requirements.

One example of continual improvement achieved in the nuclear industry is the reduction in the number of cladding failures which was called the Zero by 2010 initiative previously [78–81] and is the Driving to Zero initiative today, and which is discussed in Section 6.2.2.

A key to building up customer confidence that quality targets have been met and will be met in the future is to maintain open communication between the customer and the supplier.

5.5. SPECIFIC REQUIREMENTS APPLICABLE TO FUEL DESIGN

5.5.1. Design control

The following design related activities are subject to quality management requirements:

- — Design input (e.g. design requirements);
- — Design process;
- — Control of interfaces between different disciplines and entities involved in fuel design;
- — Configuration management (e.g. design or computer code version change management);
- — Analysis;
- — Verification;
- — Validation.

The development activities cover a large scope, starting from researching ideas (creativity) to solve a problem, up to being able to put a new product on the market (industrial use). It is important to segment the development activities into phases, moving from one phase to the next one via gate reviews — that is, reviews allowing validation of the development performed and the relevant results (tests and studies) to make sure that the probability of success is sufficiently high before entering the following phase.

In the field of nuclear fuel, the closer the industrialization phase, the more stringent are the quality assurance requirements. Part of the results obtained during the development process will be used later in the licensing process. For that reason, it is important to identify in each phase of the project all of the tests and studies needed to prove at a late stage that the proposed solutions meet the design criteria and the safety rules.

The robustness of the solutions that are proposed to the customers depends on the quality and the exhaustiveness of the test and study programme implemented. This requires having (or having access to) many test facilities, including irradiation facilities and hot cells, as far as fuel product development is concerned.

Fuel assembly components and fuel assembly products need to undergo irradiation tests to ensure adequate in-reactor operation behaviour without affecting the available safety margins.

5.5.2. Design documentation

Design documentation, which is submitted for licensing review, includes the following documents (as a minimum):

(a) Design basis:
   - (i) Design criteria (design and safety limits);
   - (ii) Reasons for criteria (degradation and failure mechanisms);
   - (iii) Justifications for the criteria (analytical and experimental evidence that criteria are adequate);
   - (iv) Design conditions.

(b) Computer code:
   - (i) Description of the code and models;
   - (ii) Summary of the verification and validation results and the quantified model uncertainties;
   - (iii) User's guide.

(c) Design verification:
   - (i) Description of the verification methodology (analytical and experimental methods);
   - (ii) Summary of the results of the verification work demonstrating that the design basis requirements have been met and reference to relevant topical reports for detailed information;
   - (iii) Topical reports referenced in the design verification summary.

5.5.3. Design qualification process

The fact that the new fuel product needs to obtain an operating licence from safety authorities needs to be considered during all of the development steps. Therefore, the development programme needs to include not only tests to demonstrate relevance, but also all of the tests necessary for the demonstration of the safety of the proposed innovations.

As discussed in Section 4.1, minor design changes are usually implemented directly, while major changes require further qualification investigation, including experimental irradiations in test reactors and lead test fuels irradiations in commercial power reactors.

The advantage of power reactors is the ability to irradiate lead test fuels under representative operating conditions, but the drawback is the lack of flexibility inherent in their energy production function; unlike in test reactors, it is impossible to test fuel to the limits in commercial reactors (e.g. fast variations of power up to fuel failure). Consequently, only concepts that have been tested beforehand, and that can be manufactured in an industrial prototypical way are irradiated in commercial power reactors.

However, there are few test reactors in operation today, and sometimes they are only partially representative in terms of operating conditions (e.g. neutron flux, core geometry and irradiation duration, cycle length). As a result, the use of commercial power reactors to validate or qualify new fuel concepts is generally required.

The typical qualification process in commercial power reactors should include the following:

- Establishing design documentation, including outcomes of the validation or qualification tests realized previously (see above);
- Involving the nuclear power plant operator and safety authority as soon as possible;
- Defining a strategy of progressive introduction of the lead test fuels;
- Performing on-site inspections (if needed, examinations in hot cells) after each cycle;
- Reporting inspection results to the plant operator and safety authority to obtain authorization for reloading in the next cycle.

Irradiation campaigns in commercial power reactors, on-site inspection and, if needed, examinations in hot cells are mandatory steps in the development process for new fuel products. They are used to validate the new designs and quantify their performance in representative irradiation conditions. When lead test fuels are loaded in a commercial power reactor, the fuel designer should provide guidelines for the loading pattern design to minimize the risk of detrimental interaction with resident fuel assemblies while providing enough bounding irradiation conditions to enable proper demonstration (e.g. positioning of lead test fuels, restriction on the power peaking factors, compatibility with existing fuels, etc.).

If necessary and approved by the operator and safety authority, campaigns can be extended beyond normal irradiation durations to provide fuel rod candidates for test reactors that allow the study of fuel behaviour beyond the operating limits (e.g. ramp tests for fuel failure, accident test conditions such as LOCA and RIA tests). The success of the irradiation campaigns ensures that future operation with full batch reloads can be performed on an industrial scale without contingency.

Furthermore, as the current fuel assembly discharge burnup target is beyond 60 GWd/t (for LWRs), a long duration (approximately ten years) is required, including irradiation time, cooling time and transportation, as well as all post-irradiation examinations and tests, to achieve convincing results. The costs of measurements and tests on irradiated material or under irradiation are very high and nuclear fuel research and development represent a significant investment over long periods. These time and cost constraints need to be considered in the qualification and licensing processes.

5.5.4. Typical steps for qualification and licensing processes for new types of fuel

The qualification and licensing phases are the responsibility of the fuel supplier on one side and the operator on the other. It is important that the responsibilities of each party involved in the design and safety evaluations are clearly defined. For CANDU systems, design, qualification and licensing of new fuel types are typically the responsibility of the operating utility, with the fuel supplier bearing the responsibility to manufacture fuel according to specifications provided by the utility.

One example of a typical qualification process is as follows. After the fuel designer develops a new type of nuclear fuel that has not previously been manufactured and installed at a nuclear power plant, it communicates with the operator to start the qualification process.

The first step consists of a general design description presented to the operator for a preliminary evaluation of fuel compatibility, the fuel design and the main characteristics of the new fuel. In this step, full scale lead test fuel assemblies are usually presented (together with drawings).

The second step is to present the core physics methodologies used, verification and validation (V&V) of documentation for computer codes and nuclear design reports.

The third step is to present the fuel assembly mechanical design methodology, the V&V documentation for the computer code and the fuel assembly mechanical design report. In this step, a presentation is made regarding the full scale prototype fabrication process, mechanical test results and stress analyses under normal and under LOCA plus safe shutdown earthquake conditions. Here, the thermomechanical design of fuel rods also needs to be presented, taking into account the fuel rod design methodology, V&V documentation for fuel rod analysis computer codes, the properties of the zirconium alloys and the fuel rod mechanical design report.

The fourth step is the fuel assembly thermohydraulic design with the fuel assembly thermohydraulic design methodology, V&V documentation for the computer codes and the fuel assembly critical heat flux and thermohydraulic design reports. In this step, the results of the hydraulic and endurance tests for the full scale fuel assembly mock-up are presented and discussed.

The fifth step is the safety evaluation. In this step, the following are reviewed and discussed:

— V&V documentation for computer codes;
— LOCA analysis methodology;
— Behaviour of zirconium alloys under LOCA and RIA conditions;
— LOCA analysis report;
— Compatibility with resident fuels and safety evaluation report.

If required, a transport container design review is provided.

After the completion of all of the steps related to the design, compatibility constraints and safety of the new fuel, a fuel vendor preliminary qualification needs to be performed. The requirements for the fuel vendor qualification depend on the operator, safety authority and State nuclear and safety laws. However, the usual sequence of such a process is as follows:

(a) Preparation of requirements for qualification of fuel manufacturing process;
(b) Development and coordination of qualification plans;
(c) Qualification of zirconium component manufacturing process;
(d) Qualification of fuel assembly and component manufacturing process;
(e) Audit of design process.

5.6. SPECIFIC REQUIREMENTS APPLICABLE TO FUEL MANUFACTURE

5.6.1. Management of procurement

There are two issues to be considered in nuclear fuel procurement: the quality of the supplied fuel and long term, economical and stable procurement. The latter issue is met by long term supply contracts with major suppliers, placing value on diversification of suppliers and supply areas. Furthermore, by participating in mining projects and maintaining a sufficient level of inventory, procurement management should guarantee the required fuel for the next several years.

In addition to the fuel assemblies, the contract between fuel supplier and utility covers such items as design, submission of design information for licensing, manufacturing, transport and quality management throughout the implementation of the contract responsibilities. The scope might also include the supply of natural uranium, conversion and enrichment services. Moreover, due to the importance of the fuel for radiation safety and the high economic value of the fuel, it is vital that the adequacy of the design and manufacturing is demonstrated by the fuel supplier[15] to the utility and the safety regulators before fuel manufacturing starts.

Before the fuel assemblies may be used in the customer's nuclear reactor, they need to be licensed by the regulatory authority of the customer's State, taking into consideration that there are different licensing requirements in different States. Licensing covers the design of the fuel and its manufacture.

5.6.2. Management of production activities

5.6.2.1. Quality plan to meet the requirements and expectations of designer and customers

A summary of the successive manufacturing steps is presented in Fig. 12. (Major steps in nuclear fuel material, component, fuel rod and fuel assembly manufacturing are described in Appendix III.) The product delivered at the end of this process should have the features required by the designer in the drawings or specifications. A synthetic view of the nature of the product quality requirements at the different key points of the manufacturing sequence is presented in Table IV–7 in Annex IV.

[15] Note that for CANDU reactors, the adequacy of the fuel design is the responsibility of the utility, which is the holder of the operating licence.

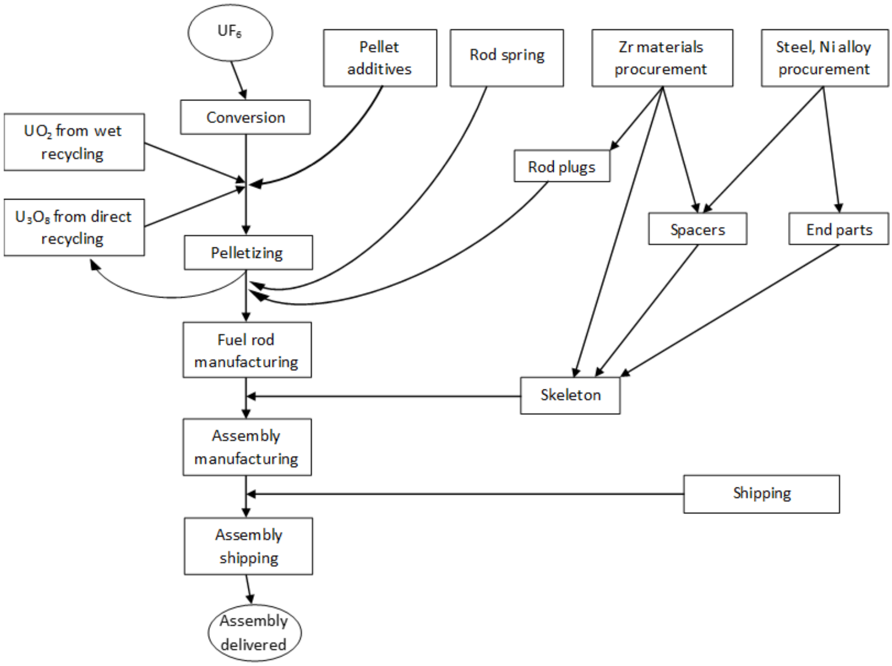


FIG. 12. Summary of the successive manufacturing steps. (Courtesy of Framatome.)

The following three items contribute to achieving and confirming final product conformity:

(a) Preparatory engineering to control the manufacturing process:
   (i) The product is progressed through successive steps, where it is transformed (i.e. conversion, pressing, sintering) or assembled (i.e. pellet loading, skeleton welding) or stored or handled.
   (ii) At each of these steps, the equipment and the operating conditions need to be attentively prepared and defined by the equipment engineer and the process engineer to be sure that this step contributes efficiently to the expected final quality of the product.
   (iii) The manufacturing phase needs to be prepared and some ranges in which the important operating conditions are to be contained need to be identified to provide stable and repetitive manufacturing situations.
(b) Collection of manufacturing samples and testing of samples:
   (i) During the manufacturing process, at appropriate steps, representative samples need to be produced, and destructive tests conducted on these

samples need to prove that the process is under control and generates correct products consistently.

(c) Direct examinations on-line of the actual manufactured product:

(i) During the manufacturing process, direct non-destructive inspections are performed on the product itself to check conformity to the requirements concerning dimensions, visual aspects and cleanliness.

5.6.2.2. Quality assurance system

A quality assurance system for production consists of the following related activities:

- Process qualification;
- Product qualification;
- Maintenance;
- Quality control;
- Quality assessment;
- Quality documentation.

In nuclear fuel manufacturing, the two parties, the supplier and the utility, cooperate in the field of quality assurance and have a certain assignment of tasks. For example, the utilities check the relevant processes and verify that design requirements are adequately translated into the manufacturing documents.

The scope of the vendor in the context of the nuclear fuel fabrication depends on the specific requirement. When the fuel or services are purchased, emphasis needs to be placed on developing the capability for qualification of suppliers, auditing and inspection, and checking upon receipt. In the case where the technology for fuel manufacture or services is purchased, greater emphasis is placed on project management, manpower training, management development, and procedure and equipment qualification.

In the sections below, process qualification, product qualification and maintenance in terms of calibration and control of measuring and testing equipment are described. The quality control system is described in Section 5.6.2.3.

(a) Process qualification

Process qualification means proving that the process meets the requirements of the required product design. A process meets the requirements if the probability of producing a non-conforming product is sufficiently small. A process qualification requires that the parameter limitations in the design are correctly accounted for in manufacturing documents, since the product needs to be produced within these limits.

Manufacturing tolerances and uncertainties also need to be considered. The restrictions from a particular design need to be imposed on the qualified manufacturing process or the process needs to be requalified. QMS, and hence also QA, guarantees long term safety and profitability.

An important feature of process qualification is analysing the inherent process variability. There are two possibilities for assessing process variability: a conventional one and an innovative one.

(i) The traditional variety was introduced by Shewhart in the 1920s [82, 83]. He distinguished between two types of causes of variability, or as it is often called, variation — common, or random, causes of variation, and assignable causes of variation. Common causes of variation are based on randomness and cannot be removed economically, and this type of variation is therefore considered to be unavoidable. The second type of variation that can be observed involves variations where the causes generally affect the mean variation negatively and can be identified precisely and eliminated, such as poor quality in raw materials, an employee who needs more training or a machine in need of repair.

(ii) A new possibility was introduced in the Nuclear Fuel Quality Management Handbook [84], according to which the variability is not classified but characterized by a stochastic model that not only reflects any assignable cause but represents a complete picture of process variability. The model may be used as a reliable basis for making decisions aiming at reducing the overall variability.

The activity of establishing a new process can be termed process selection qualification (see Ref. [85], for example). This is the first step before the process is selected and used for regular production. Owing to a break in the process (such as maintenance or modifications), the already established process also needs to be qualified. For process selection qualification, the manufacturing process is to be designed so that it can meet the required characteristics of the product with satisfactorily narrow variations.

For checking process capability, simply establishing control charts to monitor whether a process is in control is not sufficient. The process needs to be capable and under control before production begins. Traditionally, process capability is measured by the process capability index. For process qualification of an already established process, detailed examination and qualification of the following elements need to be conducted individually:

— Input material to be processed;
— Equipment to be used for processing;
— Procedure to be used for processing;
— Personnel involved directly in the processing.

An example of process qualification is provided in Annex II.

(b) Product qualification

Any product development process is a structured decision making process, often based on go/no-go funding decisions. Generally, it starts with investigating the technological feasibility. The second step consists of a technological evaluation and an early product evaluation, which lead to the actual product qualification, including consideration of the manufacturability, functionality, reliability and application of the product.

As regards nuclear fuel, product qualification needs to analyse nuclear fuel compositions in an accurate and reliable manner and to identify and quantify trace impurities. Microstructural characterization is essential for solving fuel manufacturers' problems. Analytical methodologies are such as to permit the validation of nuclear fuels and the study of fuel handling related parameters.

Product qualification ends with the setting of requirements and specifications that need to be considered when the manufacturing process is designed, developed and qualified. The final product needs to undergo a final qualification before the process and the product are qualified. Once this has been performed, the regular production of the fuel product can be initiated.

(c) Calibration and control of measuring and test equipment

The equipment used for measuring and test procedures for the various manufacturing steps always needs to have the necessary accuracy and precision of measurement. To ensure the reliability of the measurements, a system for verification needs to be established. Such a calibration system needs to take care of the selection, calibration, adjustment, maintenance and control of inspection, measuring and test equipment. All of the calibration standards need to be traceable via secondary standards to the national and international standards. If required, these secondary standards also need to be calibrated sufficiently often with the primary standards available at the national level. Measuring instruments, pressure gauges, temperature controllers, mechanical weighing scales and electronic balances, among other items, are calibrated periodically, and the details of the same are maintained in the respective quality assurance plans.

Much analytical instrumentation is comparative and therefore requires a sample of known composition (reference material) for accurate calibration and measurement. Reference material is material or substance, of which one or more of the property values are sufficiently homogeneous and well established to be used for calibrating an apparatus, assessing a measurement method or assigning values to materials. Certified reference material, accompanied by a certificate, is reference material with one or more property values certified by a procedure that establishes traceability to an accurate realization of the unit in which the property values are expressed, and for which each certified value is accompanied by an uncertainty at a stated level of confidence. Certified material information is available in Annex III.

The main measurement methods used with respect to reference material, as presented in Annex III, are the following spectrometry methods:

— Thermal ionization mass spectrometry;
— Isotope dilution mass spectrometry;
— Inductively coupled plasma mass spectrometry;
— Gas source mass spectrometry.

Techniques and methods for the destructive examination of reactor materials are described in Ref. [86].

Measurement uncertainty is often underestimated, which often leads to interpretation difficulties for measurement or test results. These difficulties can be overcome by describing the inherent variability of the used measurement processes via stochastic models.

5.6.2.3. Quality control and assessment

A quality control system lists the processes, services and products to be monitored, how monitoring is to be performed for each of the listed entities and the specifications to be met, including rejection criteria. The quality control system includes detailed instructions concerning the corrective actions that are to be performed depending on the monitoring result. Quality control is traditionally classified into process control and product control.

(a) Specific quality control standards

The following standards are typical examples used as specific QC standards:

(i) For inspection processes:
   — Standard Guide for Establishing Calibration for a Measurement Method Used to Analyze Nuclear Fuel Cycle Materials (ASTM C1156) (Rev. 3) [87];
   — Calibration Laboratories and Measuring and Test Equipment General Requirements (ANSI Z540-1-1994) [88];
   — Measurement Management Systems: Requirements for Measurement Processes and Measuring Equipment (ISO 10012:2003) [89];
   — Standard Guide for Qualification of Laboratory Analysts of Nuclear Fuel Cycle Materials (ASTM C1297) (Rev. 3) [90];
   — Recommended Practice, Personnel, Qualification and Certification in Non-destructive Testing (ASNT-TC-1A) [91].

(ii) For materials and components:
   — Standard Specification for Wrought Zirconium Alloy Seamless Tubes for Nuclear Reactor Fuel Cladding (ASTM B-811-02) [8];
   — Standard Specification for Wrought Zirconium and Zirconium Alloy Seamless and Welded Tubes for Nuclear Service (Except Nuclear Fuel Cladding) (ASTM B-353) [7];
   — Design and Quality Assurance Practices for Nuclear Fuel Rods (ASTM C934) (Rev. 2) [92].

Apart from these nuclear specific standards, the whole range of quality control standards developed by the ISO and the International Electrochemical Commission (IEC) is applied in the nuclear industry for process control and product control, for internal as well as for external control activities.

(b) Monitoring, inspection and repair

Quality control methods generally provide for three different actions:

(i) Periodic monitoring actions;
(ii) Inspection;
(iii) Repairs.

If monitoring indicates a deviation from requirements and triggers an alarm, an inspection may be performed to verify and localize the deviation. If a deviation is confirmed by the inspection, it is removed, for instance, by repair. There are in-house quality control and controls performed by external inspectors.

Each of the actions performed in the framework of quality control relies on a correctly calibrated measurement device (refer to Section 5.6.2.2). The interpretation of any monitoring and inspection results that are obtained needs to be determined uniquely, otherwise it is impossible to make an appropriate decision. It needs to be certain that unacceptable products or faulty processes are detected for timely corrections.

The activities, observations and reactions need to be documented according to specified regulations and need to be approved by the corresponding authorities. Moreover, the awareness of the quality control staff needs to be maintained by adequate measures.

(c) Process control and monitoring

Process control involves comparison of the output of a process with a standard and the taking of remedial actions in the case of a discrepancy between the two. It involves the determination of the ability of a process to produce a product that meets desired specifications or requirements. Usually, statistical tools are used to establish the process control, which generally consists of monitoring the output from a process. If the process observation is consistent with the given standard, the process is assumed to be under control, otherwise a deviation from the standard is indicated and leads to further actions. Note that ideally the standard is given by a validated stochastic model, which increases the reliability and the accuracy of the necessary comparison.

In many practical situations it is impossible to carry out 100% monitoring of the process, or if an additional supervision is necessary, surveillance is performed. For a monitoring programme to be followed, it is necessary to define the scope (type and extent) and intervals of the monitoring actions. Monitoring functions as an indirect control of the production methods, equipment and personnel that affect the quality. The results of process monitoring, such as the recent rejection rate, type and number of repair actions, are also to be made available to external inspectors to enable a judgement to be formed concerning the actual state of the process. Some of the surveillance checkpoints include the following:

— Validity of the applied procedure and test instructions;
— Calibration (date) of the test and measurement equipment;
— Use of processing parameters according to instructions;
— Checking the manufacture of released material only;
— Comprehensive documentation of the accompanying records;
— Processing and handling in accordance with instructions;
— Labelling and identification of items;
— Qualifications of procedures, machinery and personnel;
— Validity and availability of standards for comparison.

(d) Product control

Product control aims to identify non-conforming products and eliminate them. The subject of product control may be single items or lots or batches of items. In the former case, one or several measurements can be performed, depending on the number of the quality characteristics and their nature. In the latter case, it is often not possible to test each item in a lot or batch, if testing is too expensive, time consuming or destructive. In such cases, acceptance sampling plans are used to decide on the quality of the lot or batch.

Practices for quality control of LWR fuels are introduced in Annex IV.

5.6.2.4. Licensing aspects for fuel manufacturing

Licensing of the manufacturing of the materials, components and complete fuel assemblies is performed with written documents, such as:

— Specifications;
— Drawings;
— Manufacturing documents;
— Inspection plans.

As to the required quality, the supplier and utility cooperate in manufacturing surveillance, fuel assembly design and engineering verification, and verification that design requirements are adequately translated into the manufacturing documents.

The responsibility of the vendor in the context of the nuclear fuel fabrication depends on the specific requirement. When the fuel or services are purchased, emphasis needs to be placed on developing the capability for qualification of suppliers, auditing and inspection, and confirmation upon receipt. In the case where the technology for fuel manufacture or for services is purchased, greater emphasis is placed on project management, manpower training, management development, and procedure and equipment qualification.

There are other typical activities that are the result of cooperation between the supplier and utility during the procurement process. Before manufacturing reload quantities, the fuel type needs to be licensed in the customer’s State. The licensing typically consists of the design basis and design verification as described above.

During manufacture, the customer conducts surveillance of the manufacturing process in the plant. Final acceptance of the fuel typically takes place at the site after the receipt inspection. In addition to the delivery of the fuel assemblies themselves, at each stage of the fuel supply the supplier should submit certain documents to the customer for their acceptance and further for the approval of the safety authority.

5.7. QUALITY ASSURANCE FOR SOFTWARE TOOLS AND CALCULATION TECHNIQUES

5.7.1. General conditions

Software needs to be uniquely identified, and the software name, version identifier and certification status need to be included in the code output. A certified software executable is to be used from controlled access locations that provide safeguards to ensure the integrity of these files. The associated source code, compilation directives, inputs and outputs for test cases are to be stored as for lifetime records. The software certification can be modified by the software configuration administrator only according to specifications (procedure needed). The documentation for all software modifications should be retained in the software certification file supplied to the software configuration administrator. All aspects of the development project are independently reviewed if the software needs to be certified.

5.7.2. Software requirements and development authorization

For any new software development or modification, the following information is to be considered:

- Functional requirements, including specific input, processing and output details;
- Performance requirements, design constraints or quality concerns;
- Any customer or project specific requirements, such as external interfaces, quality concerns and project design reviews, among other things;
- Code modification requests that are open.

5.7.3. Software development or modification

During development, the responsible programmer documents the software design and implementation details. This software design documentation should include a description of the product via the detailed software documentation.

5.7.4. Software verification and validation plan

Each computer code version and revision is to be tested according to the software verification and validation plan (SVVP) that is included in the certification file. The SVVP includes a description of the methods to be employed by the responsible programmer or engineer to ensure accurate implementation of the software design.

Each test case specified in the SVVP should include acceptance criteria for the qualification of the software. The acceptance criteria should consider conservatisms required for consistency with applicable regulations and regulatory guidelines.

5.7.5. Software qualification

Software qualification needs to be demonstrated by satisfactory execution of the SVVP. Execution of the SVVP is to be documented in the certification file and includes:

- A record of execution of the SVVP;
- Documentation of important input, output and script files;
- Comparison of test case results with experimental data or reference output;
- Demonstration that test results comply with the code requirements, for example through the requirements traceability matrix;
- A review of differences relative to acceptance criteria;
- Justification of all unexpected differences between test case results and reference output;
- A statement of conclusions.

5.7.6. Software review

The software development and maintenance activities performed should be confirmed by independent review. The independent reviewer reviews the entire code certification file for the software to be released. Satisfactory completion of the independent review is evidenced by the reviewer's signature on the software release authorization. This independent review should be performed by one or more qualified individuals who have not prepared any of the materials being reviewed. The reviewer may be from the same organization, including the direct supervisor or manager, provided that the reviewer did not prescribe or limit the methods or inputs to be reviewed and the supervisor or manager is the only competent reviewer available within the schedule requirements of the software project. Justification for review by a manager should be documented to include the extent of the reviewing manager's input into the software items and approved by the next highest level of management on each occasion.

5.7.7. Software release

A software release authorization is used to authorize the installation of the software on the computer system for production use. The software release authorization includes the following:

(a) Applicable computer platforms or specific machine definitions for all installed software;
(b) Description of the effect of the code modifications in topical reports approved by the regulatory body;
(c) For internally developed or qualified software, a designation of the product version to be installed as certified software in a controlled access system and its level of certification.

5.7.8. Software documentation

Certified software is to be documented and available at the time the software release authorization is signed. The following guidelines apply:

(a) The documentation should address functional (theory), application (user) and programming information for the software.
(b) To the maximum extent practical, the user and theory documents are to be maintained in a complete, non-fragmented form to minimize the confusion of multiple references in the official documentation.
(c) Regardless of the documentation structure (multiple manuals or a single manual), the following elements need to be addressed:
    (i) Application information (user manual);
    (ii) Functional definition (theory manual);
    (iii) Programming information (programmer manual).

5.7.9. Problem reporting

If an error is discovered in any software version or related documentation that could impact on past results, error reporting for the software is to be completed in compliance with a clear procedure.

5.7.10. Software retirement

When it is determined that a computer code is no longer required, software retirement is to be effected by the issue of a software release authorization that authorizes and announces the retirement and disablement of all remaining versions of the computer code.

6. GOOD PRACTICES TO IMPROVE FUEL RELIABILITY IN REACTOR OPERATION

6.1. GOOD PRACTICES TO ACHIEVE FAILURE FREE FUEL PERFORMANCE

Fuel assemblies should be designed and manufactured as appropriate to ensure that:

(a) Fuel assemblies do not fail and release fission products during normal operation and AOOs;
(b) Suitable geometries of the fuel assemblies are maintained so that the insertion of control rods is not impeded;
(c) Reactor core coolability is maintained during postulated accidents.

For each fuel assembly design, the fuel vendors or fuel designers determine the design limit on the linear power generation rate (also known as bounding power history or thermal design limit) to ensure that all design criteria are met and that their operating or economic restrictions are also considered in order to ensure that there are no PCI induced failures during power ramps, as shown in Fig. 13.

For all fuel assembly designs in the reactor core, the most limiting design limit during normal operation (design limit — normal) on the linear power generation rate is identified to establish the core operating limit, which needs to be respected during operating in normal operation (see Fig. 14). This should also ensure that the most limiting design limit during AOOs (design limit — upset) on the linear power generation rate is respected.

In addition, the cycle specific fuel design verification is performed by the core designer (either the fuel vendor or the operator) to ensure that the core loading pattern is appropriate and that the core operating limits (and fuel fit for service limits for CANDU reactors) are not exceeded. Finally, the operator should

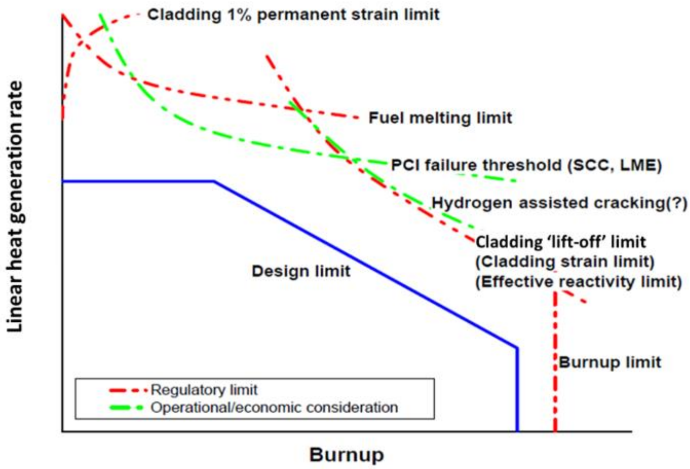


FIG. 13. Definition of design limit on the linear power generation rate. (Courtesy of C. Patterson from ANT International.)

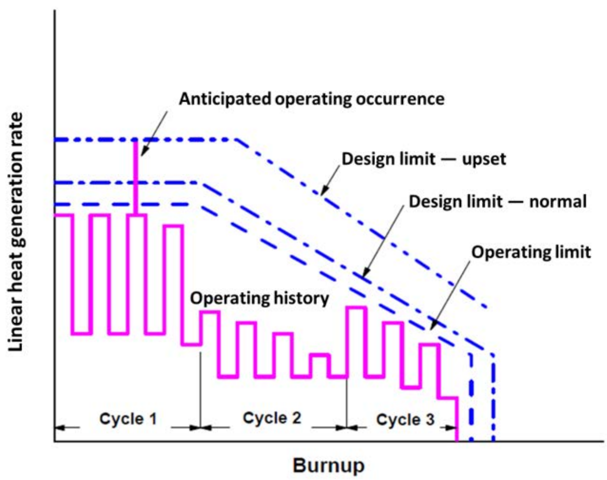


FIG. 14. Definition of core operation limit on the linear power generation rate. (Courtesy of C. Patterson from ANT International.)

monitor the core operation to ensure that all fuel assemblies operate within the core operating limits throughout their operating life.

To ensure trouble free performance of the fuel, it is crucial that the operator carries out the following:

(a) Design audits to ensure that the safety analysis is performed according to the standard and contractual requirements. Quality assurance of the contractor is also audited.
(b) Fuel manufacturing audits to ensure that the qualified manufacturing processes are enforced effectively, that the verifications are performed as required, and that the quality plan is applied effectively.

Such good practices aiming at trouble free fuel operation have been consolidated into guidelines by industrial organizations such as the World Association of Nuclear Operators (WANO), INPO and the Electric Power Research Institute (EPRI). The available industrial guidelines are summarized in Section 6.2.

6.2. EXAMPLES OF INDUSTRIAL GUIDELINES

6.2.1. WANO guidelines for fuel reliability

The WANO intends "to maximize the safety and reliability of nuclear power plants worldwide by working together to assess, benchmark and improve performance through mutual support, exchange of information and emulation of best practices" as stated on the WANO member web site[16].

After collecting members' reports on events via the secure web site, the WANO Operating Experience Central Team reviews all reports and prepares Significant Operating Experience Reports or Significant Event Reports and Just-in-Time briefings to address important topics. The most relevant WANO report on fuel reliability and performance is Managing Core Design Changes (SOER 2004-1) [93].

WANO also provides good practices for improving safety and reliability over all the key nuclear disciplines, including operations, maintenance, etc.

The performance indicator is to provide a quantitative indication of plant safety and reliability along with personnel safety. Fuel reliability indicator is one of the important indicators that each member needs to report quarterly. It can be used by fuel designers, manufacturers and operators to improve fuel reliability.

[16] Refer to: https://www.wano.info

6.2.2. INPO guidelines for achieving excellence in nuclear fuel performance

INPO intends "to promote excellence in the operation of commercial nuclear power plants" in the United States of America (as stated on the INPO web site[17]).

Guidelines for achieving excellence in nuclear fuel performance (see e.g. [19]) can be summarized in terms of seven key attributes necessary for achieving and sustaining failure free fuel performance. The characteristics of each attribute are described to help define station specific standards and expectations. The INPO web site currently provides additional detailed information that can be used to enhance performance in each of these areas. The seven attributes are as follows:

(a) "Attribute 1 — the management team focuses the organization on achieving and maintaining failure free fuel performance;
(b) Attribute 2 — effective controls are used to prevent foreign material (debris) from causing fuel cladding failures;
(c) Attribute 3 — design and operating practices minimize the potential for cladding failures caused by PCI;
(d) Attribute 4 — changes to plant design and operation, core design and fuel management practices, and chemistry programmes are appropriately monitored, and chemistry parameters are controlled, to prevent corrosion related and crud induced fuel failures;
(e) Attribute 5 — GTRF failure mechanisms are systematically addressed and eliminated;
(f) Attribute 6 — fuel monitoring and inspection activities assess current margins, evaluate the impacts of changes and determine the causes of fuel cladding failures;
(g) Attribute 7 — fuel fabrication oversight activities provide additional assurance that new fuel delivered to the station does not result in a fuel cladding failure."

6.2.3. EPRI guidelines and handbooks

EPRI's overall Fuel Reliability Program is outlined in Fig. 15. It aims to maintain excellent fuel performance and reliability with a continuous learning process where guidance drives actions to monitor and assess fuel performance, obtain operating experience along the way and then feed that operating experience back into the guidance. In Fig. 15, the operating experiences are recorded in

[17] Refer to: https://www.inpo.info

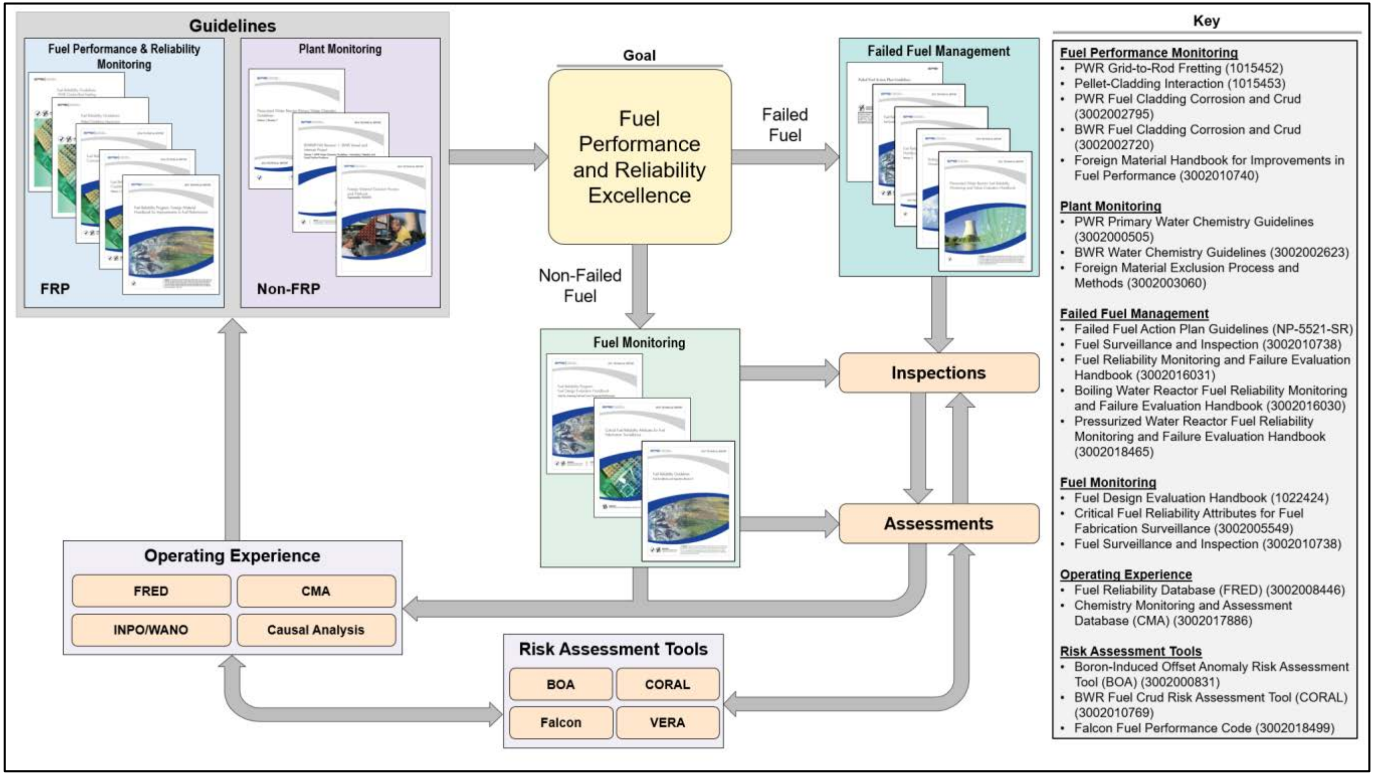


FIG. 15. Fuel reliability programme implementation for achieving excellent fuel performance and reliability. (Courtesy of EPRI.)

EPRI's Fuel Reliability Database (FRED), EPRI's Chemistry Monitoring and Assessment (CMA) database, related INPO and WANO databases, and utility and fuel vendor causal analyses reports. The risk assessment tools include EPRI's Boron-induced Offset Anomaly (BOA) code for PWR crud related issues, EPRI's Crud Deposition Risk Assessment Model (CORAL) code for BWR crud related issues, EPRI's Falcon fuel performance code and the US Department of Energy's Virtual Environment for Reactor Applications (VERA) code.

Establishing a fuel reliability programme aimed at achieving this goal involves applying several EPRI guidelines that capture the industry's up to date knowledge and provide specific guidance and good practices to prevent fuel failures. From there, plant assessment and inspection programmes are established for monitoring non-failed fuel to assess overall fuel reliability and performance and to check for potential precursors to fuel failure or other fuel related issues.

In this process, the qualitative assessment of the margin for various fuel failure mechanisms, along with change management, drive the need to perform non-failed fuel inspections to characterize fuel reliability and performance quantitatively. Meanwhile, failed fuel inspections are utilized to accurately determine the cause of fuel failure and identify gaps in the established guidance or its implementation. Quantification of margins, considering either failed or non-failed fuel inspection data, often includes utilizing various risk assessment tools or performing operating experience reviews to establish confidence in achieving the goal of excellence in fuel performance and reliability.

As more operating experience is accumulated and analytical tools improve, the data and lessons learned are utilized to revise and enhance the EPRI guidelines documents in order to provide the industry with the most up to date and comprehensive collection of resources to ensure excellent fuel reliability and performance.

An important foundation for excellence in fuel reliability is ownership by all levels of the nuclear power plant organization. As shown in Fig. 16, the number of US fuel cycles (which are typically 18 to 24 months long) with failed fuel has decreased significantly since the early 2000s. In Fig. 16, the pie charts detail the mechanisms of fuel failures for each time frame, where the colours denote different mechanisms per the legend at the top right of the figure. Note that the unknown leaker mechanism includes uninspected and inspected but indeterminant causes of failure. Some failures have yet to be inspected in the most recent time frame.

The EPRI FRED database recorded 167 fuel cycles with at least 1 fuel failure in the cycle in the 2001–2007 time frame. Most of those failures were due to GTRF and the second leading cause of failure was debris fretting (i.e. from foreign material). For the 2015–2022 time frame, only 47 cycles had fuel failures, and those were primarily from debris. This significant improvement

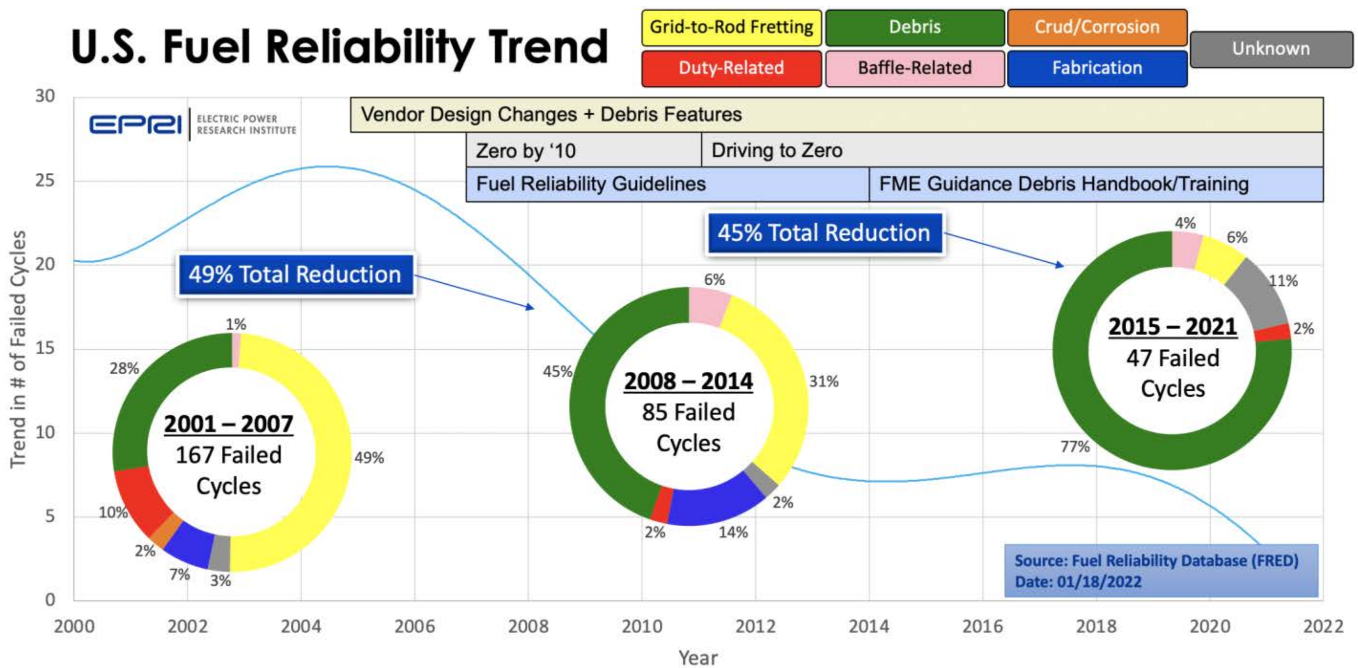


FIG. 16. Evolution of US industry fuel failures alongside industry initiatives and efforts addressing fuel reliability challenges. (Courtesy of EPRI.)

(~70% reduction) in reliability was due to the Zero by 2010 goal (subsequently renamed Driving to Zero) set by all of the chief nuclear officers to eliminate fuel failures. Many stakeholders were essential to the noted improvements in reliability: senior nuclear power plant management set the expectations; fuel reliability, engineering and maintenance staff implemented policies and best practices; INPO developed guidelines and conducted assessments; EPRI developed technical guidelines, handbooks and training; and fuel vendors improved fuel designs and manufacturing practices. Consistent focus on the goal of zero fuel failures by all stakeholders is essential to excellence in fuel reliability.

Appendix I

FUEL DESIGN AND SAFETY LIMITS

I.1. FUEL ROD DESIGN AND SAFETY LIMITS

I.1.1. Design limits for normal operation and anticipated operational transients

To ensure reliable operation, the following criteria need to be satisfied for normal operation and anticipated operational occurrences (AOOs).

The design limits presented in this Section are applicable to fuels for light water reactors (LWRs) [94], including water water energy reactors (WWERs) [95], and are in general applicable to fuels for pressurized heavy water reactors (PHWRs). Note that specific design limits for PHWR fuel are described in detail in Ref. [4]. The applicability of current design limits to advanced technology fuels (ATFs) is discussed in Ref. [96].

I.1.1.1. Fuel temperature

(a) Criterion

To avoid fuel rod failure due to overheating for normal operations and AOOs, the maximum temperature of the pellets needs to remain below the fuel melting temperature, considering the burnup dependent thermal conductivity degradation and specific burnable absorbers effects (e.g. gadolinia), as well as the uncertainties.[18]

(b) Technical basis

Overheating of the fuel pellet can lead to centreline fuel melting. If the volume of melted fuel is significant enough, the pellet expansion due to phase

[18] As an example, for PWR UO_2 and $(U,Gd)O_2$ fuel, minimal melting temperature, considering its burnup dependent degradation, is usually given by [97]:

$$T_{melting}\ (°C) = 2810 - 7.6 \times 10^{-4}\ Bu$$

where Bu is the fuel pellet burnup in MWd/tU.

In practice, a conservative value of 2590°C may be considered for fuel that considers various manufacturing tolerances, uncertainties relative to centreline temperature computation and thermal barriers that can be related to aspect defaults.

transformation, together with melted fuel/cladding interaction phenomena, can lead to cladding damage and failure.

(c) Applicability to ATF

The same criterion and considerations can be used for ATF if the thermal expansion coefficient of the advanced fuel pellets is comparable to the UO_2 thermal expansion coefficient.

Nevertheless, for fuel concepts including cladding with poor elastic–plastic behaviour, such as SiC cladding, the maximum fuel temperature needs to be specified. To avoid excessive stresses within the cladding (and subsequent fibre microcracks), it might be necessary to maintain the maximum centreline temperature below the fuel melting temperature limit.

I.1.1.2. Overheating of the cladding

(a) Criterion

Fuel cladding is not to be overheated under operational states. As an uncoupled approach, this criterion is covered by the thermal margin criterion (e.g. departure from nucleation boiling ratio (DNBR) for pressurized water reactors (PWRs), critical power ratio (CPR) for boiling water reactors (BWRs)) for normal operation and anticipated operational occurrences.

(b) Technical basis

This is intended to prevent excessive cladding temperature due to the degradation of heat transfer from fuel to coolant (e.g. excessive fission gas release, corrosion or crud, high duties).

(c) Applicability to ATF

A similar criterion applies to ATF pellet concepts. However, specific critical heat flux tests may be required in order to reassess the applicability of the limit value, and new DNBR correlation may be needed.

Revolutionary cladding concepts such as Mo cladding or SiC cladding are supposed to be resistant to higher temperatures and, as such, could withstand conditions beyond the DNBR criterion. Nevertheless, these cladding concepts are far from ready for use in a commercial nuclear power plant, and their anticipated high performance at high temperatures is better suited for accident conditions than for normal operation and AOOs.

I.1.1.3. Cladding oxidation and hydriding

(a) Criterion

Cladding oxidation and hydriding buildup are to be limited during normal operation to maintain adequate fuel to coolant heat transfer, cladding ductility and strength.

For PWRs, in the absence of detailed justifications, the 'best estimate' maximum circumferential average oxidation thickness is typically to be limited to 100 μm at the end of life. Alternatively, or equivalently, the best estimate maximum volume average hydrogen content pick-up by the cladding is to be limited to an acceptable low value (i.e. 600 ppm at the end of life).[19]

For CANDU reactors, a limit on the fuel cladding oxidation is established with a different value (i.e. <10 μm or <3% of the cladding thickness).

(b) Technical basis

Limitations on the cladding surface temperature and on the cladding oxide layer are needed to avoid conditions for accelerated oxidation or hydriding and conditions where the cladding mechanical properties are degraded. This criterion might be crucial in case of crud deposit.

Excessive cladding oxidation due to accelerated oxidation or hydriding could lead to cladding failure. Oxidation essentially accelerates because of the thermal feedback effect of the growing corrosion layer thickness on the temperature at the interface, where the corrosion rate has an Arrhenius type of dependence on temperature. As a result, the end of life criterion of 100 μm has been set to limit the cladding temperature (and then the corrosion acceleration), to limit cladding thinning (based on the Pilling–Bedworth coefficient, 100 μm of corrosion corresponds to a standard 10% clad thinning) and thus cladding embrittlement (related to the hydrogen pick-up fraction) and the risk of extensive corrosion layer spallation.

The hydrogen pick-up criterion limits the loss of ductility due to hydrogen embrittlement.

Note that limitations on crud buildup are also needed to avoid crud induced power shift or axial offset anomaly. This requires a strict water chemistry control and surveillance programme. The United States Nuclear Regulatory Commission (NRC) requests that the impact of the cruds on the fuel rod design and safety analysis be considered in the future.

---

[19] Alternatively, a limit on the maximum oxide–metal surface temperature can be specified.

(c) Applicability to ATF

A similar criterion is enforced for ATF cladding. Since ATF cladding may present significant differences in terms of corrosion and hydriding behaviours, specific limits need to be determined for each cladding concept.

I.1.1.4. Fuel rod internal pressure

(a) Criterion

During normal operation, including operational transients[20], fuel and burnable poison rod internal gas pressures need to remain below the system pressure or are allowed to exceed the system pressure under the following conditions:

- No cladding lift-off;
- No hydride reorientation in the cladding radial direction;
- No DNB propagation.

(b) Technical basis

During normal operating conditions, the internal pressure due to the fission gas release and initial pressurization needs to remain lower than the value that, when power is held constant, would lead to an increase or a reopening of the pellet to cladding diametric gap.

This criterion precludes the rate of the outward clad creep from exceeding the rate of the fuel swelling and therefore ensures that the gap does not reopen during steady state operation. If DNB occurs on a fuel rod in which rod internal pressure is above coolant pressure, it needs to be demonstrated that its potential ballooning does not result in local flow blockage and DNB occurrence in neighbouring rods.

This prevents the accelerated release of fission gases at high burnup and avoids high burnup fuel becoming a limiting case from a loss of coolant accidents (LOCA) viewpoint.

Limited end of life rod internal pressure also makes it possible to avoid cladding creep-out issues during long term storage of the fuel rods, when the coolant counterpressure is gone. Mixed oxide (MOX) fuel is quite sensitive

[20] The impact of massive load following may be considered in terms of additional fission gas release.

to rod internal pressure issues because of its higher fission gas releases during operation and its specific He releases during storage.

In CANDU reactors, fuel cladding is designed to collapse under coolant pressure to establish the required fuel pellet to coolant heat transfer. The internal fuel rod fission gas pressure remains lower than coolant pressure so that the fuel sheath remains collapsed, and the required heat transfer is maintained.

(c) Applicability to ATF

The same requirements apply to the ATF rod system (fuel pellet and cladding) — rod internal pressure and cladding creep properties need to be such that there is no risk of cladding lift-off during normal operation and AOOs.

I.1.1.5. Cladding stresses

(a) Criterion

The maximum[21] fuel clad stresses under normal operation and under AOOs need to remain below the design limits (maximum allowable values).

(b) Technical basis

This design criterion is defined to prevent cladding damage due to excessive fuel clad stresses under static and cyclic loads.

Typically, for operational states, the maximum average stress on the clad thickness needs to remain below the cladding material temperature dependent yield strength determined using biaxial out-of-pile tests on irradiated tubes ($\sigma_{0.2}$).

(c) Applicability to ATF

The mechanical properties of the ATF cladding should be assessed in the same way as for the standard claddings, and the stress design limits should be quantified likewise.

For coated claddings, it might be necessary to define specific stress limits if there is a risk of spallation of the coating at stress levels lower than those of the cladding material temperature dependant yield strength.

For cladding based on SiC fibres, maximum stresses during normal operation and AOOs should prevent microrupture of fibres and subsequent

[21] The maximum cladding stress is located at the outside diameter for pressure driven loading (rod internal pressure) or at the inside diameter for strain driven loading (PCMI).

significant cladding damage. As such, the maximum allowable cladding stress during normal operation and AOOs should be defined in correlation with a specific maximum allowable cladding damage index.

I.1.1.6. Cladding radial strains

(a) Criterion

The maximum fuel clad strains under normal operation and under anticipated operating occurrences are to be less than the design limits (maximum allowable values[22]).

(b) Technical basis

This design criterion is defined to prevent cladding damage due to excessive fuel clad strains under static and cyclic loads.

(c) Applicability to ATF

The mechanical properties of the ATF cladding need to be assessed in the same way as for the standard claddings, and the strain design limits need to be quantified likewise.

For coated claddings, it might be necessary to define specific strain limits if there is a risk of spallation of the coating at strain levels lower than those used for standard Zr based claddings.

For cladding based on SiC fibres, the maximum strain during normal operation and AOOs needs to prevent microrupture of fibres and subsequent significant cladding damages. As such, the maximum allowable cladding strain during normal operation and AOOs needs to be defined in correlation with a specific maximum allowable cladding damage index.

I.1.1.7. Fuel rod axial growth

(a) Criterion

Dimensional changes need to be limited to prevent fuel failures and functional hurdles (e.g. fuel assembly handling).

---

[22] During normal operation, the total plastic tensile creep strain (increase in rod diameter due to uniform cladding creep and fuel pellet expansion) is typically limited to +1.0% from the unirradiated initial condition.

(b) Technical basis

Irradiation induced axial growth of fuel rods can lead to significant interference between the rod upper end cap and the upper nozzle, resulting in rod bowing or overstressing thimble tube to nozzle joints.

Axial clearance between the fuel rods and the fuel assembly structure needs to be sufficient to accommodate expected dimensional changes of the fuel assembly and fuel assembly components during the irradiation lifetime of the fuel assembly.

(c) Applicability to ATF

The same criterion applies to ATF rods. Irradiation growth of ATF cladding material needs to be quantified under prototypical irradiation conditions to validate the fuel assembly performance calculation codes used for the reactor core design.

I.1.1.8. Cladding fatigue

(a) Criterion

The cumulative number of strain fatigue cycles for the structural components of the fuel assemblies needs to remain significantly below the design fatigue lifetime.

(b) Technical basis

The design fatigue lifetime is determined on the basis of appropriate data, taking account of "a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles", as stated in Ref. [98]. Other proposed limits need to be justified.

(c) Applicability to ATF

The same requirement applies to the ATF claddings. As such, the fatigue lifetime of the ATF cladding needs to be determined under prototypical test conditions (e.g. maximum stress and/or strain amplitudes expected during load following).

Special attention needs to be given to coated cladding (risk of coating spallation) and SiC cladding (poor elasticity with a risk of microcracks among the constitutive fibres).

I.1.1.9. Fretting wear

(a) Criterion

Fretting wear at the contact points between fuel assemblies' components needs to be limited. Typically, fretting wear needs to remain below 10% of the nominal thickness of the cladding tube.

Further, to avoid the lateral vibration of fuel rods because of grid to rod fretting (GTRF), the normal force exerted by the grid cells on the rod needs to remain positive throughout fuel assembly lifetime.

(b) Technical basis

Under the effect of the coolant flow induced vibrations, the relative motion of the fuel rods and the grid support dimples could lead to cladding wear at the contact points, which could cause a cladding failure.

(c) Applicability to ATF

The same requirement applies to the ATF cladding. As such, the ATF cladding properties under irradiation (i.e. irradiation creep and growth) need to be determined under prototypical conditions to permit the proper design of the fuel rod support within the fuel assembly grid cells in order to ensure that there is enough lateral support of the fuel rod throughout its lifetime. Coated claddings might exhibit better resistance to GTRF phenomena.

I.1.1.10. Cladding elastic stability (beginning of life freestanding)

(a) Criterion

At beginning of life, during the hot hydrostatic tests, the cladding properties are such that coolant pressure does not lead to collapse or plastic deformation of the cladding.

(b) Technical basis

The criterion precludes the formation of the instantaneous collapse of the clad onto the fuel pellet stack or onto the plenum zone owing to the difference of pressure across the cladding.

(c) Applicability to ATF

The same requirements should apply to ATF claddings.

The elastic resistance of coated and SiC claddings is expected to be higher than that of standard Zr based claddings, making this requirement easier to fulfil.

I.1.1.11. Cladding collapse (long term buckling or flattening)

(a) Criterion

The cladding of the fuel rod is not allowed to collapse during the handling and operation of the fuel rod bundle in the reactor.

(b) Technical basis

Cladding collapse refers to the dimpling of cladding into short, unsupported axial gaps that can form within the fuel column.

Axial gaps need to be avoided or minimized, first by verifying that there are no gaps formed during manufacturing, second by ensuring that the compression force generated by the plenum spring on the pellet stack is sufficient, third by setting handling and shipping arrangements to avoid excessive axial accelerations and, finally, by avoiding excessive in-reactor fuel pellet densification.

The cumulative action of inner and outer differential rod pressure and cladding creep may cause increasing ovality of the cladding up to an instability threshold. This may lead to circumferential clad buckling at any axial gaps in the fuel stack and therefore to cracking.

Hence, the fuel rod needs to be designed so that the cladding is not susceptible to collapse from the long term effects of cladding creep. The designer needs to specify the criterion used for collapse.

CANDU reactor fuel is designed with collapsible thin walled fuel cladding that functions to establish the required fuel pellet to coolant heat transfer characteristics. Due to the collapsible feature of the thin cladding, there exists a potential for cladding collapse along the unsupported length of the cladding between the endcap and end pellet (axial gap region) or between two neighbouring pellets.

(c) Applicability to ATF

Since axial fuel pellet gaps may occur in ATF rods, the same requirement should apply. As such, the mechanical properties under Zr irradiation of the ATF claddings need to be determined under prototypical conditions.

I.1.1.12. Stability of the fuel stack — fuel rod hold down spring

(a) Criterion

Prior to irradiation, the fuel rod hold down spring is designed so as to ensure a minimum hold down force on the fuel pellet stack in the axial direction during handling and shipping, The current limit is based on established specification limits for shipping and handling (e.g. 4 g for PWR fuel assemblies).

(b) Technical basis

Hold down spring force applied to the top of the fuel stack prevents any significant fuel pellet motion and chipping during shipping and handling operation.

Expected maximum shipping and handling loads are based on the shipping and handling procedures and the corresponding specifications and qualifications. For example, if the specification indicates a maximum axial acceleration of 4 g, this implies that the shipping and handling procedures will ensure that the fuel stack will never experience accelerations higher than 4 g, and, therefore, a minimum hold down force on the fuel pellet stack in the axial direction of four times the nominal weight of the fuel stack will prevent pellet motion.

(c) Applicability to ATF

If the ATF pellets' mechanical properties are comparable to those of the standard fuel pellets, the same design requirement should apply to the ATF rods. It might be necessary to confirm this statement with proper post-irradiation examinations.

I.1.1.13. End plug weld integrity

(a) Criterion

The fuel rod end plug weld is such as to maintain fuel rod integrity (fuel rod leak tightness) during operational states and does not contribute to any additional fuel failures above those already considered for accident conditions.

(b) Technical basis

The aim is to maintain fuel rod integrity (i.e. fuel rod leak tightness).

(c) Applicability to ATF

The same requirement applies to ATF cladding.

If the cladding is made of two or three material layers (e.g. coated cladding), the end plug welding needs to be such that there is no weak area at the connection location.

For the specific case of SiC cladding, the technique used for the end plugs connections to the cladding requires a comprehensive demonstration under prototypical conditions.

I.1.1.14. Fuel rod bow

(a) Criterion

The maximum acceptable fuel rod bow needs to be determined, taking into account other design limits, such as DNBR correlations.

(b) Technical basis

It has been observed that some fuel rods (and burnable poison rods) bow during operation. As a result, the lateral spacing between fuel rods can vary. The maximum rod bow should be determined (usually based on comprehensive campaigns of poolside rod bow measurements), and the effects should be considered in the fuel design:

— The local variation of the fuel–moderator volume ratio of the peak local fuel rod power level;
— Variations in coolant subchannel diversions of the DNB margin or critical heat flux.

(c) Applicability to ATF

The same assessment needs to be carried out for ATF rods. This might require determining the specific mechanical properties of the ATF rod system (irradiation creep and growth), complemented with a series of poolside rod bow measurements (in-reactor irradiation experience).

I.1.2. Design or safety limits for accident conditions

I.1.2.1. Fuel melting

(a) Criterion

For operational states, the centreline fuel temperature remains below the fuel melting temperature in all cases. The volumetric fraction of fuel melting at the hot spot is to be limited in accident conditions.

(b) Technical basis

The number of failed fuel rods is used to assess the radiological consequences of the accidental transients. The traditional practice is to assume that fuel rod cladding failure occurs whenever the centreline temperature is higher than the fuel melting temperature.

In practice, a more realistic approach, based on demonstrative experiments, consists of limiting the amount of fuel melting in volumetric fraction to an acceptably low value (e.g. 5% or 10%) to limit fuel thermal expansion and subsequent cladding failure as well as melted fuel ejection into the coolant.

(c) Applicability to ATF

Cladding failure is mainly due to the excessive thermal expansion of the overheated fuel pellet. As such, the allowable fuel pellet overheating (or fraction of centreline fuel melting in accident conditions) should be limited according to the expected tensile resistance of the cladding.

Special care should be given to SiC cladding, which cannot accommodate large strains and may exhibit local microcracks liable to challenge the leak tightness of the fuel rod.

I.1.2.2. Cladding overheating

(a) Criterion

During operational states, DNB is precluded by design. For accident conditions, the number of fuel rods reaching DNB needs to be limited.

(b) Technical basis

As the number of failed fuel rods is used to calculate the accident radiological consequences of the accidents, it is commonly assumed that any fuel rod experiencing DNB should be considered failed.

However, DNB is not a failure mechanism per se, it is a thermohydraulic phenomenon leading to local clad overheating and related physical consequences (e.g. excessive cladding corrosion). If validated multiphysics calculation codes are available, the number of failed fuel rods can be quantified more realistically on the basis of actual fuel failure mechanisms (e.g. cladding runaway oxidation or embrittlement).

In addition, the applicability of the DNB correlations to rapid accidents such as reactivity initiated accidents (RIAs) should be demonstrated.

(c) Applicability to ATF

To avoid excessive cladding overheating, the same requirements apply to ATF cladding as for DNB phenomenon occurrence.

Nevertheless, some of the proposed ATF claddings exhibit much higher resistance to high temperature corrosion. As such, new DNB limitations could beneficially be defined on the basis of appropriate validation experiments.

I.1.2.3. Cladding runaway oxidation during non-LOCA accidents

(a) Criterion

Maximum peak clad temperature (PCT) needs to remain below the post-DNB cladding temperature limit[23] for non-LOCA accidents such as RIAs and locked rotors.

(b) Technical basis

This criterion aims at preventing post-DNB cladding failure due to high temperature runaway oxidation during non-LOCA accidents.

---

[23] The typical limit value of 1482°C was taken from early data estimates of the fuel failure boundary for LOCA conditions (2700°F or 1482°C and 17% of clad thickness oxidized by metal–water reaction). The rationale for a higher temperature limit was that non-LOCA transients were of short duration and fuel rods could withstand short periods of DNB without serious damage.

The occurrence of oxidation and hydrogen absorption during normal operation can cause an additional reduction of ductility at high burnup. This indicates that the PCT during the transient may need to be adjusted to accommodate normal corrosion formed before the transient. Cladding materials (e.g. the use of Nb as an alloying agent) are also expected to affect this PCT criterion due to their effects on ductility. A PCT limit as a function of fuel burnup is being proposed by fuel designers.

Note that the PCT can significantly exceed the limit without clad failure due to embrittlement caused by excessive oxidation. Time after DNB is also important for cladding overheating failure. Other mechanistic limits (ECR), time at temperature, strain) are also being proposed by fuel designers.

(c) Applicability to ATF

ATF claddings have been specifically designed to withstand high temperature conditions. As such, the post-DNB temperature limitations that have been defined for the standard claddings could apply to the ATF claddings with significant margins. If the applicant wishes to relax the post-DNB temperature limits to take advantage of high performance ATF claddings, specific tests should be carried out under the appropriate test conditions to determine the new limits.

I.1.2.4. Cladding embrittlement during LOCA

(a) Criterion

The limits on the PCT and ECR are to be met to ensure that fuel rod fragmentation due to cladding embrittlement is avoided.

(b) Technical basis

The criteria were based on the consideration that retention of ductility could be the best guarantee against potential fragmentation under various types of loading (thermal shock, bundle constraints, hydraulic, handling, seismic forces).

Recent LOCA tests with high burnup fuels have shown that these embrittlement criteria may not be sufficient to protect all failure mechanisms during LOCAs. Consequently, the embrittlement criteria are subject to change towards performance based criteria, such as the ECR limit as a function of the hydrogen content or burnup in the cladding.

(c) Applicability to ATF

ATF claddings have been specifically designed to withstand high temperature conditions such as those encountered during a LOCA transient. As such, the LOCA criteria that have been defined for the standard claddings could apply to the ATF claddings with significant margins. If the applicant wishes to relax those criteria to take advantage of high performance ATF claddings, specific tests should be carried out under the appropriate test conditions to determine the new limits.

I.1.2.5. Fuel dispersal or expulsion during RIAs

(a) Criterion

The limits on the peak fuel enthalpy need to be met to prevent catastrophic fuel rod failure due to fuel melting and dispersal and hence avoid molten fuel–coolant interaction.

(b) Technical basis

The limits on the peak fuel enthalpy were based on the early RIA tests showing that melting of UO_2 causes cladding failure and possible expulsion of fuel particles, which in turn leads to energetic fuel–coolant interactions and subsequent pressure pulses in the coolant within the fuel assembly bundles.

Recent RIA tests with highly irradiated fuel have indicated that the current peak fuel enthalpy criterion is not conservative enough to cover all fuel failure mechanisms. Various new criteria are being proposed for the allowable fuel enthalpy or enthalpy increase as a function of fuel burnup, clad oxidation or hydriding, or differential pressure between the fuel rod and the coolant.

(c) Applicability to ATF

Assuming that the ATF cladding is susceptible to failure under RIA conditions, the risk for fuel fragment dispersion and fuel–coolant interaction depends in the first order on the fuel pellet characteristics.

If the ATF pellet design is close to licensed fuel pellets (e.g. UO_2, MOX, Gd_2O_3 fuel pellets) the risk for fuel dispersal is covered by the current RIA limits.

If the ATF pellet concept and characteristics differ significantly from those of standard fuels, specific investigations need to be undertaken to better characterize the loading applied by the fuel pellets to the cladding during the transient. The propensity of the irradiated fuel pellets to break into small fragments that are liable to disperse and interact with the coolant also needs to be assessed.

Special attention needs to be given to the following physical mechanisms that may take place in ATF pellet concepts:

- — The amount of fission gas trapped in the fuel matrix prior to the RIA transient;
- — The fission gas bubble distribution within the fuel matrix at the beginning of the RIA transient;
- — The fragment size spectrum of the irradiated fuel pellets subjected to a rapid thermal transient.

I.1.2.6. Fuel coolability during LOCAs and RIAs

(a) Criterion

Fuel coolability criteria[24] are given for all major damage mechanisms (e.g. cladding embrittlement, fuel dispersal or expulsion, fuel melting) and apply only to postulated accidents described in Chapter 15 of the safety analysis report (SAR).

(b) Technical basis

The coolability criteria are defined to prevent significant fuel failures due to cladding embrittlement, fuel dispersal and fuel melting. They need to be met at all parts of the core to ensure a coolable geometry during postulated accidents, such as LOCAs and RIAs.

---

[24] For example, to meet the requirements of general design criteria GDC 27 and 35 of the US NRC 10 Code of Federal Regulations Part 50 Appendix A [99], criteria for the coolability or coolable geometry of the core are defined to cover the following major fuel failure mechanisms:

- — Cladding embrittlement. US NRC 10 Code of Federal Regulations Part 50 §50.46 [100] indicates acceptance criteria for the LOCA: 1204°C on peak cladding temperature and 17% on maximum local cladding oxidation.
- — Fuel dispersal or expulsion. In the case of RIAs, the US NRC Regulatory Guide 1.236 [101] indicates a radially averaged enthalpy limit of 230 cal/g.
- — Fuel melting. In the case of RIAs, the US NRC Regulatory Guide 1.236 [102] indicates that fuel melting needs to be restricted to the innermost 10% or less of the fuel pellet at the hot spot.

(c) Applicability to ATF

ATF claddings have been specifically designed to withstand high temperature conditions. As such, the fuel coolability criteria that have been defined for the standard claddings could apply to ATF claddings with significant margins. If the applicant wishes to relax those criteria to take advantage of high performance ATF claddings, specific tests need to be carried out under the appropriate test conditions to determine the new limits.

I.1.2.7. Fuel failure thresholds for RIAs

(a) Criterion

Fuel rod failure thresholds are defined to determine whether or not a fuel rod should be considered failed in radiological release assessments. They should cover the known failure mechanisms, namely:

- Post-DNB high temperature cladding failure (balloon/burst, runaway oxidation induced embrittlement);
- PCMI;
- Fuel melting;
- Molten fuel induced cladding failure.

(b) Technical basis

The cladding failure limits for RIAs were initially set by the NRC [98] as the maximum radially averaged fuel enthalpy of 170 cal/g for BWRs and as the DNB criterion for PWRs.

Based on recent RIA experiments with fuel rods irradiated to a burnup of ~50 GWd/t or higher, it appears that these limits need to evolve to cover additional mechanisms such as PCMI. Various limit values, such as function of fuel burnup, clad oxidation or hydrogen, or differential pressure between the rod and the reactor core, have been proposed (Figs B–1 and B–2 in Ref. [98]).

(c) Applicability to ATF

The primary fuel rod failure mechanism during an RIA transient is PCMI. As such, the applicant needs to demonstrate, based on appropriate mechanical property tests carried out under prototypical test conditions, that the ATF claddings exhibit the same level of resistance as standard claddings. Nevertheless, if the fuel pellets characteristics (e.g. transient gaseous swelling, thermal expansion coefficient, melting temperature, etc.) differ significantly from those of the standard fuel pellet concepts, specific tests should be carried out under prototypical conditions to reassess the actual PCMI loading provided by the fuel pellets on the cladding.

If the fuel rod survives the primary PCMI failure mode, it may have to face a high temperature secondary failure mechanism. Since ATF claddings have been specifically designed to withstand high temperature conditions, it should not be problematic for the applicant to demonstrate that the limitations that have been defined for standard claddings are applicable to ATF claddings.

I.2. FUEL ASSEMBLY MECHANICAL DESIGN CRITERIA

I.2.1. The relevant specific acceptance criteria for fuel assembly mechanical design are recalled here.

I.2.1.1. Loads, stresses and strains during normal operation and AOOs

(a) Criterion

The loads, stresses and strains in the structural components of the fuel assembly need to be lower than the design limits.

(b) Technical basis

Structural integrity has to be maintained on the fuel assemblies during normal operation and AOOs. As such, the loads applied to the structural components of the fuel assembly should not result in structural deformations that can affect the bases or the assumptions of the nuclear, thermal and hydraulic design of the core, or the RCCA or instrument probe functionality.

Stress and strain limits that are obtained by methods such as those given in Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) [103] are acceptable. Other proposed limits have to be justified [98].

I.2.1.2. Assembly growth

(a) Criterion

Maximum possible fuel assembly growth needs to leave a positive axial clearance between the top of the fuel assembly and the upper core plates under cold conditions (more penalizing than under hot conditions).

Maximum possible lateral grid growth needs to maintain clearance, as appropriate, between the peripheral fuel assemblies and the core baffle, and between adjacent fuel assemblies, or between the grid and the channel in channelled fuel assemblies (including penalizing cold conditions).

(b) Technical basis

Excessive loads on fuel assembly and internals need to be prevented. Such loads could lead to fuel assembly bow and guide thimble tube distortion, generate handling issues during loading and unloading operations, impede correct RCCA insertion and damage the connections between nozzles and guide thimbles.

I.2.1.3. Compatible with the reactor vessel internals structures

(a) Criterion

The design needs to be compatible with the reactor vessel internal structures, with consideration of worst case dimensional tolerances.

(b) Technical basis

Ensure that fuel assembly is adequately supported and located in the core, can be inserted and removed, and fit-up with the upper internals, or, in other words, ensures correct alignment of the assemblies’ nozzle holes (or pins) with the core plates pins (or holes).

I.2.1.4. Compatibility with the fuel handling equipment

(a) Criterion

Design needs to be compatible with the fuel handling equipment.

(b) Technical basis

Ensure proper fit-up and interaction with fuel handling equipment and other equipment used for shipping, storage and inspection of the fuel.

I.2.1.5. Compatibility with the RCCA and other core components

(a) Criteria

The following criteria need to be met:

— Core component rods need to not contact the guide thimble tube screw.
— Assembly growth needs to not crush thimble plugs or secondary sources.
— Guide thimble design needs to allow adequate RCCA drop times and slow down, with adequate cooling.

(b) Technical basis

Ensure that RCCAs and other core components can be fully inserted, and that excessive axial forces or impact velocity do not occur during a scram. Adequate cooling prevents the development of excessive corrosion in the thimbles.

I.2.1.6. Fretting wear

(a) Criterion

Fretting wear at the contact points between the fuel rods and the grid springs and dimples needs to be limited. The allowable fretting wear (e.g. 10% of the initial cladding thickness) is defined and accounted for when assessing stress and fatigue limits [98].

For CANDU reactor fuel bundles, bearing pad wear is limited to a 0.3 mm reduction in pad height. Fuel cladding does not contact fuel handling equipment during loading or discharge or the pressure tube during residence of the fuel bundle in-core.

(b) Technical basis

Excessive cladding wear could lead to cladding failure. To assess the risk for fretting wear, appropriate modelling has to be used, accounting for various flow induced vibration mechanisms, locally and globally, and for the irradiation effects. Full scale testing of dummy fuel assemblies under representative bounding flow conditions, including long term tests (endurance tests), is used to validate the models.

I.2.1.7. Assembly hold down force[25]

(a) Criteria

Hydraulic loads for normal operation need to remain below the hold down capability of the fuel assembly (if applicable) either by gravity or through the hold down springs system. Vertical lift-off forces need to not unseat the lower fuel assembly tie plate from the fuel support structure [98] It should be ensured that:

- The hold down springs provide sufficient hold down force to ensure that the fuel assembly does not lift off during any normal operation and AOO event other than a hot pump over-speed transient;
- The hold down force is sufficient at the beginning of life and throughout the fuel cycle. The effects of fuel assembly growth and spring force reduction, due to permanent set and irradiation induced stress relaxation, are to be considered.

The calculated minimum net hold down force needs to exceed the lift-off threshold by 100 lb (45.36 kg) (recommendation).

(b) Technical basis

Unseating a fuel bundle can challenge RCCA functionalities and generate undesired wear of the internal in contact with the fuel assemblies.

PWR fuel assemblies are equipped with hold down springs attached to the upper nozzle. Hold down springs prevent fuel assembly lift-off that may be caused by hydraulic loads during plant operational states. Note that some limited and temporary lifting is tolerated, provided that the hold down spring system maintains enough elasticity to prevent fuel assembly lift-off once the transient is terminated.

[25] Equivalently, for CANDU reactors with fuelling directions counter to the coolant flow direction, possible fuel string relocation during a postulated coolant inlet header break accident has been calculated to result in a possible reactivity insertion into the core. This adverse scenario is managed by limiting the available gap between the coolant inlet fuel bundle and the fuel channel shield plug by measurement of this gap using the fuel handling equipment and by use of 'long' fuel bundles to control the length of the fuel string in the fuel channel.

The fuel assembly hold down force acts with compressive forces on the guide tubes, leading to high fuel assembly bow due to irradiation induced guide tube creep, while high compressive forces can be generated in turn from excessive guide tube growth. An appropriate design for the hold down spring system should provide well balanced downward forces (to prevent lift-off) while preventing excessive compressive loading on the guide tubes.

Guide tube growth is correlated with fast neutron fluence. At high burnup levels, high corrosion and hydrogen pick-up of the guide tube accelerate the rate of guide tube growth above that of the fast neutron irradiation growth. Corrosion and hydrogen pick-up depend on coolant temperature and on the guide tube's material composition and conditions. To ensure that the guide tube's corrosion and hydrogen pick-up are acceptable, therefore, its design and materials need to be selected appropriately.

In summary, the safety requirements regarding the hold down system design remain unchanged for new fuel assembly concepts and component designs, provided that all necessary mechanical properties are determined, especially at high burnup, under prototypical conditions and used with validated analytical tools.

I.2.1.8. Fuel assembly bow and twist

(a) Criterion

Fuel assembly bow due to fuel assembly design and irradiation needs to not impede control rod insertion, which could cause an increase of the control rod drop time.

(b) Technical basis

Excessive fuel assembly bow impedes control rod insertion, which in turn results in an increase of the control rod drop time. Fuel assembly bow increases water gaps between fuel assemblies and can result in higher local power peaking factors and modified coolant flow distribution, which in turn can reduce the local DNBR margins. Fuel assembly bow is a multifactorial phenomenon that is dependent on fuel assembly lateral stiffness, irradiation creep, compressive axial load, non-uniform heat rates and flux rates, assembly–assembly interaction, inter-fuel assembly gap size distribution, rod to grid interaction, etc.

I.2.1.9. Blowdown/seismic loads

(a) Criterion

Core coolability and control rod insertability need to be ensured under the combined seismic and LOCA loads. Specifically, the following requirements are to be met:

(i) Fuel rod fragmentation is not to take place, which can be ensured through demonstration that fuel rod stresses are within limits per Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) [103], Appendix F.
(ii) Control rod insertion is not to be impaired, which can be ensured through demonstration that stresses are within the reference limits in item (1), that guide tubes are not buckled, that spacer grid crush limit is not exceeded or has limited deformation, etc.
(iii) Spacers are not to be distorted sufficiently to affect rod coolability, which can be ensured through demonstration that the hydraulic section of the grid cells is maintained.

The mechanical loads resulting from a combination of a seismic event and a LOCA need to not cause fuel assembly damage to such an extent that control rod insertion is prevented or coolability cannot be maintained.

(b) Technical basis

During a seismic and LOCA event the fuel assemblies can sway back and forth, causing impacts with neighbouring fuel assemblies and with the baffle structures. Jet forces due to blowdown from one side of the vessel through a broken pipe could also accelerate the vessel motion in the lateral direction, causing similar contact to occur.

I.2.1.10. Compatibility with the instrumentation probe interface

(a) Criterion

The location of the instrumentation tube in the assembly is to be corrected, with adequate inside diameter and straightness.

(b) Technical basis

Ensure that the in-core instruments can be properly inserted.

I.2.1.11. Capture of fuel rods within the nozzles

(a) Criterion

The fuel rods may not be axially ejected from the assembly and need to stay within the nozzle space.

(b) Technical basis

Ligaments in the top and bottom nozzle plates are to be distributed adequately to ensure that the fuel rods remain within the fuel bundle tolerances throughout the fuel assembly lifetime.

I.2.1.12. Grid mismatch

(a) Criterion

A minimum contact height is necessary at each support or mixing grid elevation between adjacent fuel assemblies considering tolerances and differences in fuel assembly growth.

(b) Technical basis

It is necessary to ensure that the cross-flow is minimized, fuel rods do not bear on adjacent grids, and adequate lateral support is provided to fuel assemblies in case of earthquake or LOCA events (grid crush resistance determination is based on tests implying lateral shocks on the grid sides).

I.2.1.13. Fuel assembly identification and orientation

(a) Criteria

The following criteria need to be met:

(i) Unique identification numbers are to be used for each fuel assembly to ensure accurate handling and shipping operations.

(ii) Unique reference orientation of the fuel assembly is to be implemented during the manufacturing process to ensure safe handling and in-core operation.
(iii) The standard identification and orientation markings on grids and nozzles are to be readable throughout the lifetime of the fuel assembly.

(b) Technical basis

Ensure proper identification and orientation of the fuel assemblies to load the fuel according to the required loading pattern.

I.3. NEUTRONIC DESIGN LIMITS

The neutronic design limits are the limits used in the safety analyses for the key neutronic parameters, namely the nuclear key safety parameters.

SSG-52 [1] Paragraph 3.18 states that:

> "Nuclear key safety parameters influencing the neutronic design of the core and fuel management strategies should be established from the safety analyses that verify compliance with the specific fuel design limits".

The typical neutronic design limits to be considered in the fuel design include the following:

- Reactivity coefficients;
- Power peaking factors;
- Criticality and shutdown margin;
- Enrichment;
- Burnup.

I.4. THERMOHYDRAULIC DESIGN CRITERIA

The objectives related to the thermohydraulic design are as follows:

(a) Fuel design needs to provide acceptable margins of safety for conditions that would lead to fuel damage in operational states.
(b) There needs to be verification that the fuel design does not lead to thermohydraulic instability in operational states.

(c) The largest hydraulic loads on core components in operational states and accident conditions need to be determined. These loads are used for evaluation of the fuel hold down device.

The specific thermohydraulic design limits are given below.

I.4.1. Thermal design limit

I.4.1.1. Criterion

The hot fuel rod in the core needs to survive with at least 95% probability at a 95% confidence level when experiencing a DNB in operational states.

I.4.1.2. Technical basis

DNBR is used to confirm fulfilment of the criterion for the prevention of departure from nucleate boiling. This ratio is determined as the ratio of critical heat flux to local heat flux. DNBR is validated in consideration of parameter deviations and uncertainties, including reactor plant parameter deviations, manufacturing tolerances, computer code errors and errors of critical heat flux calculations. The DNB criterion, if met, ensures reliable heat removal from fuel rod (U–Gd fuel rod) claddings in the reactor core.

The maximum temperature of the fuel rod (U–Gd fuel rod) outer cladding surface is limited due to high heat transfer coefficients and exceeds the coolant saturation temperature at working pressure by several degrees Celsius.

I.4.2. Reactor coolant flow

I.4.2.1. Criterion

The hydraulic characteristics of the reactor core with the chosen fuel product need to ensure coolant flow rate through the reactor in operational states within the established design limits:

$$G_{\min} < G < G_{\max} \tag{I.1}$$

where G is the best estimate for the reactor flow rate, $G_{\min}$ is the thermal design flow and $G_{\max}$ is the mechanical design flow.

I.4.2.2. Technical basis

A reduction of the primary coolant flow rate to below the design limit may lead to violation of the DNBR criterion and, as a result, to fuel rod damage. An increase in the primary coolant flow rate to above the design limit may lead to violation of the design criterion for the prevention of fuel assembly lifting.

I.4.3. Thermohydraulic instability

I.4.3.1. Criterion

Stable coolant circulation needs to be provided in operational states; that is, there should be no hydrodynamic instability in the primary circuit and reactor core.

I.4.3.2. Technical basis

Loss of circulation hydraulic stability may lead to a change of the core thermohydraulic condition and violation of the DNB criterion and to the advent of unallowable cyclic loads on fuel rods and fuel assembly components, with further disturbance of fuel rod integrity.

I.4.4. Hydraulic loads

I.4.4.1. Criterion

The worst case hydraulic loads that occur in operational states should not exceed the hold down capability of the fuel assembly.

I.4.4.2. Technical basis

This criterion requires that the assembly does not levitate from hydraulic loads in operational states, to permit safe insertion of the control rods.

Therefore, for normal operation and AOOs the weight and spring loads of the submerged fuel assembly need to be greater than the hydraulic loads, for both cold and hot conditions, and at the maximum flow specified for the reactor.

One exception is the turbine over-speed transient associated with a loss of external load. The fuel assembly hold down spring pack is designed to tolerate temporary over-deflection associated with this transient and still provide contact with the lower core plate following this transient.

Appendix II

VERIFICATION OF FUEL DESIGN AND SAFETY LIMITS

II.1. FUEL ROD THERMOMECHANICAL DESIGN VERIFICATION (FOR OPERATIONAL STATES)

II.1.1. Fuel temperature

The melting temperature is defined on the basis of experimental databases, in consideration of the uncertainties. The melting temperature of unirradiated UO_2 is usually taken as 2800°C, decreased by, for example, 33°C per 10 MWd/kgU burnup [97]. For gadolinia fuel, the melting temperature is further reduced by an additional ~4°C per weight per cent of gadolinia [97]. A centreline fuel melting temperature of ~2600°C is appropriate for UO_2 fuel rods with 62 MWd/kgU burnup [97].

AOO events leading to local power increase are more limiting for this criterion than normal operation. In principle, all AOO transients of the design basis need to be investigated, but the most limiting ones can be identified (e.g. inadvertent control rod group withdrawal, starting a non-operating loop with lower temperature). These limiting transients can be calculated by system thermohydraulic and reactor neutronic codes to generate a bounding design power history.

The fuel temperature can be obtained from the fuel behaviour code calculations or from the hot channel thermohydraulic code analysis. Uncertainties in the fuel data and code are considered.

II.1.2. Overheating of the cladding

Refer to Section II.5 on fuel thermohydraulic design verification.

II.1.3. Cladding oxidation, hydriding and crud

The SAR needs to discuss allowable oxidation, hydriding and crud levels, and demonstrate their acceptability.

The following criteria are applied to evaluate the effect that corrosion of fuel assembly components has upon fuel performance:

— Corrosion behaviour characteristics of fuel assembly materials are to be obtained under conditions that are representative of the reactor environment.

— The effects of corrosion and crud film buildup on heat transfer surfaces are to be addressed in the calculation of pellet and cladding temperatures.
— The effects of fabrication processes such as cold work, heat treatment, stress relief and welding on corrosion behaviour are to be considered.
— The effects of water chemistry specifics on corrosion behaviour and crud build-up are to be taken into account.

Note that the hydrogen pick-up model and associated uncertainty are important, as the verification of the new LOCA/RIA criteria are based on the initial hydrogen content.

II.1.4. Fuel rod internal pressure

The maximum allowable pressure in the fuel rod can be greater than the system pressure[26] and depends on the pellet swelling and the clad creep resistance properties.

The variations in pressure that occur over the life of the fuel rods are considered. The fuel rod performance is affected significantly by internal pressure from creep, ballooning or rod collapse.

Calculations of fuel rod internal pressure need to account for the following phenomena:

— Different thermal expansions between pellets and cladding (in axial and radial directions);
— Irradiation induced swelling of the fuel pellets;
— The accumulation of non-volatile fission products;
— Irradiation induced densification in the rod;
— Release of absorbed gases from the fuel;
— Solubility of the fill gases in the fuel pellets;
— Helium gas release;
— The release of gaseous fission products from the fuel pellets;
— The temperature of gases contained in pellet end dishes, pellet cracks and pellet open porosity, in comparison with the temperature of gases in the annulus between pellets and cladding or in the plenum;
— Irradiation induced growth and creep of the cladding, as well as thermal creep of the cladding;
— Expected variations in initial fill gas pressure and component dimensions.

[26] For CANDU reactors, the fuel rod internal pressure is kept below the heat transport system pressure. This maintains good fuel to sheath contact and stable fuel to coolant heat transfer, per design.

It is beyond the scope of this safety reference to describe how the fuel designer calculates the rod internal pressure. However, the methods used need to be described and justified in the fuel rod design methodology report, including consideration of the uncertainties. A statistical uncertainty analysis is acceptable.

In particular, the calculated non-lift-off pressure could be validated by an available experimental data basis.

Note that since the cladding non-lift-off criterion is to prevent DNB propagation, it seems necessary to consider AOO events. One fuel vendor calculates the maximum rod internal pressure for an AOO transient, while other fuel vendors calculate it only for normal operation. This aspect needs to be clarified with the fuel vendors during the new fuel contracts.

II.1.5. Cladding stresses

The maximum allowable stress is usually defined as a function of both the offset yield stress and/or tensile strength at operating temperature. Other proposed limits need to be justified. The maximum transient stress needs to be calculated using design bounding power histories for AOO events.

Fuel performance code is used to calculate the maximum volume averaged (or clad thickness average) effective stress (e.g. the von Mises equation) taking into account pellet–clad contact, pellet deformation due to thermal expansion and swelling, uniform clad creep and pressure differences between fuel rods and coolant.

In calculating stresses, consideration needs to be given to the effects of temperature and irradiation. Uncertainties in the fuel data and code need to be considered. A statistical uncertainty analysis is acceptable.

II.1.6. Cladding radial strains

These design limits are verified analytically by the fuel vendor using qualified fuel performance codes. The design bounding power histories for normal operation and AOO events need to be used, and the uncertainties in the fuel data and code also need to be considered.

The margins between the design limits and actual strain levels are affected by the material properties of the fuel and cladding and on the burnup range.

II.1.7. Fuel rod axial growth

The axial growth of fuel rods is usually verified as part of fuel assembly mechanical design. Demonstration of compliance with this requirement needs to consider the following phenomena:

- Differential thermal expansion between the fuel rods and the fuel assembly structure.
- The effect of tolerances.
- Differential irradiation induced growth between fuel rods and between fuel rods and the fuel assembly structure. This allowance should include evaluation of axial extension of cladding induced by interaction between fuel and cladding.
- The effect of fuel assembly structure axial compression and creep.

II.1.8. Cladding fatigue

The fatigue damage of the fuel rod claddings is evaluated analytically to cover the foreseen operation modes and the whole duration of irradiation. A simplified bounding methodology could be used (e.g. using a bounding design curve independently of the loading patterns).

II.1.9. Fretting wear

The verification of this criterion is usually presented in detail in the fuel assembly mechanical design report. Fretting wear tests and analyses need to account for grid spacer spring relaxation. The allowable fretting wear needs to be specified in the SAR. The stress and fatigue limits need to presume the existence of this wear. The adequacy of the spacer grid design and its position within the fuel assembly need to be established by testing or analysis under conditions that are consistent with the intended reactor's operating coolant temperature, pressure, flow rate, and chemistry.

II.1.10. Cladding elastic stability (beginning of life freestanding)

The freestanding criterion is to be verified by ensuring the elastic stability and non-yielding of the cladding tube. Allowance needs to be made for any ovality defects of the tubes or cladding thickness differences that are consistent with manufacturing. Combinations of fuel rod dimensional tolerance, and fill gas pressure tolerances need to be considered.

II.1.11. Cladding collapse (long term buckling)

Demonstration of compliance with this requirement needs to consider the following:

- The effect of fuel rod burnup and power level on internal pressure, including the effect of a conservatively low assessment of fission gas release.
- Irradiation induced densification of fuel pellets and the solubility of the fill gas in fuel.
- The range of power histories to which the rod is likely to be subjected.
- Combinations of fuel rod dimensional tolerance and fill gas pressure tolerances. Allowance needs to be made for any ovality defects of the tubes or cladding thickness differences consistent with manufacturing.

The design of the rods through their pressurization and the use of stable fuel under irradiation avoid any risk of circumferential buckling of the cladding.

If axial gaps between fuel pellets in the column occur as a result of fuel densification because the ultimate pressure of the coolant has been exceeded, the cladding will have the potential to collapse into the fuel pellet axial gap, resulting in clad flattening and then possible fuel rod failure. Pre-pressurization of a fuel rod and higher as fabricated pellet densities have minimized this concern to a large extent.

An evaluation is performed using fuel performance code to demonstrate that the coolant pressure during normal operation and during a pressure test of the reactor vessel remains below the critical pressure leading to cladding collapse, accounting for the safety margin factor (e.g. 1.5). The critical pressure is determined by two factors, namely the yield stress and the ovalization creep process, both depending on the actual thermomechanical state of the cladding.

Only the steady state of the fuel is of concern for this criterion, therefore steady state neutronic codes are satisfactory for providing fuel behaviour codes with power history.

II.1.12. Stability of the fuel stack — hold down spring

Specific calculations need to be performed to verify the criterion, based on standard equations describing the mechanical characteristics of the spring, and considering rod design parameters and the shipping and handling specifications for allowable maximum accelerations.

II.2. FUEL ROD SAFETY EVALUATION (ACCIDENT CONDITIONS)

II.2.1. Non-LOCA fuel safety evaluation

The non-LOCA fuel safety evaluation needs to be performed for each new fuel assembly design or major modifications to the existing designs on the basis of the reference non-LOCA safety analysis results as documented in the final safety analysis report (FSAR) and the information provided in the nuclear steam supply system–fuel interface file (NFIF).

This evaluation is intended to demonstrate that the reference FSAR non-LOCA safety analysis remains valid with the new fuel products. This demonstration can be made either by quantitative analysis or by qualitative justification for non-reanalysis.

The non-LOCA heat-up analysis needs to demonstrate the fulfilment of the licensing criteria for the limiting cases identified in the FSAR (e.g. locked rotor and rod ejection) for the penalizing fuel rod (hot rod) and under the worst core thermohydraulic, initial and boundary conditions in the NFIF regarding the acceptance criterion. The choice of this fuel rod needs to be justified.

Based on the availability of information contained in the NFIF and the fuel modifications to be evaluated, the non-LOCA heat-up analyses are performed. Fuel rod non-LOCA heat-up analyses based on NFIFs are applicable only for situations where the minor fuel modifications do not affect the global plant thermohydraulics.

It is necessary to first verify that the geometrical dimensions and hydraulic parameters of the new fuel assembly remain within the bounding limits of the reference fuel assembly that are used in the non-LOCA safety analysis. If the bounding limit verification is performed, the global core thermohydraulics remains applicable for boundary conditions in the NFIF. If the reference data are not bounding, the use of global core thermohydraulic boundary conditions in the NFIF should be justified.

The bounding limits of the reference fuel rod geometrical and thermomechanical data also need to be verified. If the fuel rod data are bounding, a qualitative evaluation can be made. If the fuel rod data are not bounding or the FSAR safety margins are insufficient, specific non-LOCA fuel rod heat-up analyses using the NFIF then need to be performed.

In the case of insufficient information from the NFIF or incompatibility of the information, a simplified non-LOCA core thermohydraulic analysis based on the available information may be performed to generate the necessary data for ‘hot assembly’ or ‘hot rod’ heat-up analysis.

Currently, no complete non-LOCA plant thermohydraulic analysis as performed for the FSAR is acceptable for safety authorities for fuel safety evaluation.

The non-LOCA heat-up analysis should use codes, models and methodologies that have been or will be approved by the safety authorities.

II.2.2. LOCA fuel safety evaluation

The LOCA fuel safety evaluation should be performed for each new fuel assembly design or major modifications to the existing designs on the basis of the reference LOCA safety analysis results as documented in the FSAR and the information provided in the NFIF.

It is required to first verify whether the geometrical and hydraulic parameters of the new fuel assembly remain within the bounding parameters of the reference fuel assembly that are used in the LOCA safety analysis. The bounding character of the fuel rod geometrical and thermomechanical data should also be verified.

If the fuel rod data are bounding, a qualitative evaluation can be made. If the fuel rod data are not bounding or if sufficient data are not available to evaluate the bounding character, specific LOCA fuel heat-up analyses need to then be performed using the NFIF.

The LOCA heat-up analysis needs to demonstrate the fulfilment of the licensing criteria during a large break LOCA for the penalizing fuel rod (hot rod) and under the worst core thermohydraulic boundary conditions. The choice of this fuel rod needs to be justified.

The LOCA heat-up analysis uses codes, models and methodologies that have been or will be approved by the safety authority.

The analysis needs to cover the whole burnup range (time in life) of the fuel rod up to the design burnup limit, even if this burnup effect has not been considered in the reference LOCA analysis. The mixed core effect is also considered.

II.3. FUEL ASSEMBLY MECHANICAL DESIGN VERIFICATION

II.3.1. Overall design specifications

Reliable mechanical performance relies strongly on the planned conditions of operation during its lifetime. This includes the following:

— ICFM (in-core fuel management: cycle length, residence time, fuel management scheme or equilibrium cycle, startup/shutdown operating procedures, plant technical specifications, etc.).
— ICFM leads to the definition of power histories of the core, of the fuel assemblies and for individual fuel rods.
— ICFM also defines the coolant thermohydraulic environment and applicable chemistry.

The overall design specifications are defined as functional requirements in contractual documents, and applicable limits are included in the vendor's design documentation.

The general design objectives are defined by contract and the vendor's documentation, including at a minimum:

— The design mechanical performance requirements;
— The applicable design procedures, with applicable design margins;
— The requirement to meet current regulatory mechanical criteria.

The functional requirements and design objectives should be consistent with the contractual and licensing requirements.

To complete the analysis, the utility needs to provide the fuel vendor with interface information, such as:

— Load definition for accident conditions, basically core plate movements and loads transmitted to the fuel during LOCA and design basis earthquake;
— Compatibility information for internals, core components, instrumentation probe, existing fuel, storage racks and handling tools;
— Operational parameters — power, flow and coolant temperatures.

For CANDU reactors, the design specifications for fuel bundles are defined by the design organizations, which are either in-house at the operating utilities (the owners of the operating licence) or are provided to the licence holder by service companies under contract. The manufacturers of CANDU fuel bundles do so in response to detailed design specifications provided by the holders of the operating licences. Operating parameters for reactors are defined by detailed design and safety analyses, which are conducted either in-house or under contract by service companies.

II.3.2. Mechanical design bases

The basis of mechanical design is to maintain a coolable geometry of the fuel assembly and maintain the control rod insertability in normal and accident situations. More specifically, the following four objectives are targeted [98]:

(a) The fuel assembly is not to be damaged because of normal operation and AOOs.
(b) Fuel assembly damage is never to be so severe as to impede control rod insertion when it is required.
(c) The number of fuel rod failures is not to be underestimated for postulated accidents.
(d) Coolability is always to be maintained. Coolability means that the fuel assembly retains its rod and bundle geometries with adequate coolant to allow for removal of residual heat even after a severe accident.

In a few cases the specified acceptable fuel design limits provide the design limits, but in most cases it is up to the fuel vendor to recommend a design limit value, taking a specific failure mechanism into account. The fuel vendor also needs to provide the background data to ensure that the design limit is both necessary and sufficient.

The reliability and mechanical performance of the fuel assembly and its components are affected by the applied loads and deformations, stresses and strains, dimensional changes, hydriding and the effects of corrosion during its operation in the reactor, in conjunction with the direct effects of irradiation on the materials properties and on the coolant condition.

The stresses, strains and cyclic loading are due to the following:

— Interaction between the fuel assembly and the core cavity, neighbouring fuel assemblies and RCCAs;
— Pressurized fuel rods, including fission gas release, PCI or PCMI;
— Differential growth and thermal expansion between fuel rods and the fuel assembly structure and between the fuel assembly structure and the core cavity and control rods;
— Hydraulic forces;
— Vibration forces;
— Power history, which mainly affects fuel rod cladding stresses due to changes in temperature, power transients, power cycling or low power operation;
— On-power fuelling operations in CANDU reactors.

Dimensional changes are due to the following:

- Creep of materials under irradiation;
- Thermal expansion (radial, axial);
- Irradiation growth;
- Fuel rod elongation due to PCMI;
- Hydrogen pick-up induced growth;
- Oxidation induced growth.

Other factors affecting mechanical performance include the following:

- Coolant boiling conditions;
- Coolant hydrolysis due to irradiation;
- Coolant pH;
- Water chemistry additives (boron, lithium, zinc, hydrogen);
- Presence of debris in coolant;
- Irradiation of the structural materials that affect mechanical properties (e.g. ductility, strength, resistance to corrosion).

Hydrogen released during corrosion of Zr materials is partially absorbed and reduces their ductility. These parameters are to be considered in the mechanical design of the fuel assembly and of each component taken separately. Additional factors may be needed for specific components.

The structural components of the assembly are subject to a somewhat different combination of parameters than the fuel rods, as they are not fuelled and do not generate fission products or significant heat (apart from gamma heating), but they are subject to significantly larger structural loads. Their temperature is close to the coolant temperature, but structural components are in contact with the coolant on two surfaces, as opposed to one surface for the fuel rod cladding. This can result in higher hydrogen content.

II.3.3. Verification of design limits

II.3.3.1. Stresses

Stress design limits that are defined by Section III of the Boiler and Pressure Vessel Code of the ASME [103] or equivalent are acceptable. Other proposed limits need to be justified.

For the fuel assembly components that require stress analyses to demonstrate their mechanical integrity, the following requirements apply:

— For components subject to multiaxial stress conditions, the analysis is to use one of the recognized methods for combining stresses, such as the maximum strain energy or maximum resolved shear stress, and the criteria used for determining acceptable results need to be identified.
— For components that are subjected to cyclic loading, the cumulative effect is to be determined. The method used needs to be identified.
— For structural components that are subject to significant creep strains because of operational loadings, the magnitude of the resultant creep strain needs to remain below the limit that can produce rupture of the component.
— For components that are subjected to both cyclic loadings and creep strains, an acceptance criterion that takes both constant and cyclic loads into account needs to be established.

II.3.3.2. Hydraulic loads

Hydraulic loads need to be evaluated for the worst case for applicable plant states (normal operation, AOOs, accident conditions). For operational states, these worst case hydraulic loads need to continue to not exceed the hold down capability of the fuel assembly (with both gravity and hold down springs).

For designs that utilize hold down mechanisms (e.g. springs) to accommodate hydraulic loads, the designer needs to show that an adequate hold down force exists based on the following:

— Stress relaxation for hold down springs;
— Maximum flow rate for normal operation;
— Fuel assembly pressure drop, including possible increases due to crud deposition within the assembly;
— Combination of dimensional tolerances of the fuel assembly and supporting structures;
— Differential thermal expansion between the fuel assembly and reactor internals;
— Fuel assembly irradiation induced growth;
— Mixed core.

The required hold down force is calculated by:

$$FHD = FHY + B - W \quad \text{(II.1)}$$

where

FHD is the required hold down force;
FHY is hydraulic force;
B is buoyancy force;
W is fuel assembly weight.

The required hold down force is evaluated for cold startup and hot full power conditions, at the beginning of life and the end of life, and for hot pump over-speed. In the evaluation, tolerances and uncertainties for the following parameters are generally accounted for:

- Axial spaces between the lower and upper reactor core plate;
- Length of the assembly;
- Weight of the assembly;
- Coolant flow rate;
- Coefficients of pressure loss;
- Deflection curve and spring constant of the hold down spring;
- Axial growth of the guide tube;
- Spring relaxation.

II.3.3.3. Blowdown/seismic loads

Analyses usually consider mechanical and hydraulic loads in both the horizontal and vertical directions. Critical crushing loads determine whether these lateral impacts cause grid deformation, which leads to a reduction of coolant flow and degraded performance of the emergency core cooling system and impedes the insertion of RCCAs. Fuel rod fragmentation is also evaluated on the basis of other mechanical properties to ensure that there is no loss of coolable geometry and to ensure that guide tubes do not fracture and prevent control rods from being inserted.

Design verification is performed analytically based on the NFIF. Independent analyses are usually completed in the horizontal and in the vertical directions, and the results are combined. Horizontal analysis is a core analysis because lateral interactions among fuel assemblies in the core need to be assessed and, as fuel designs may have different dynamic properties, it needs to bound different mixed core scenarios.

Safety criteria in these areas are not directly affected by the new design elements. However, considering the analytical verifications just mentioned, the methods used to analyse the seismic or LOCA event should be well verified and validated. In some cases (e.g. when the dynamic characteristics of the fuel or the pressure losses of a new fuel differ significantly from those of the existing fuel), the introduction of a new fuel design may require the recalculation of the reference (NFIF) blowdown and seismic loads.

Design requirements to establish allowable structural loads under earthquakes during and after a LOCA may need to be altered at high burnup due to a significant change in the strength and ductility of the cladding and guide tubes at high burnup. These changes in material properties need to be verified.

II.3.3.4. Fuel assembly bow and twist

The amount of acceptable fuel assembly bow and twist needs to be accounted for in the design in accordance with the following:

- It needs to be shown that the maximum expected bow and twist can be accommodated by the handling equipment and material in fuel storage facilities.
- The effect of the fuel assembly bow and twist on control rod motion (e.g. through friction drag) needs to be assessed.
- The effect of fuel assembly bow and twist on local power and coolant flow distribution needs to be assessed.

Ideally, compliance with this criterion needs to be demonstrated by experimental measurement of the control rod drop time in a manner consistent with the plant's limits and conditions (for PWRs). If the control rod drop time exceeds the value assumed in the safety analysis, corrective actions need to be taken. These corrective actions may include demonstrating that sufficient DNB margin and control rod worth exist to accommodate increased control rod drop times, limit core power levels or even shut down the plant.

In practical terms, fuel assembly deformation and the control rod drop time are measured during plant startup as an indication of the fuel assembly bow and twist.

II.4. NEUTRONIC DESIGN VERIFICATION

The nuclear design analyses include a nuclear fuel assembly neutronic design analysis and a core neutronic design analysis.

The fuel assembly neutronic design analysis is assembly specific, and accounts for the nuclear characteristics of the fuel design parameters, such as fuel enrichments, burnable absorber content, etc. The fuel assembly neutronic design analysis is performed to assure that the new fuel design features meet the nuclear design criteria established for the fuel and core. It is specific to the assembly design, accounting for the nuclear characteristics of the fuel design parameters, such as fuel enrichments, burnable absorber content, etc.

The core neutronic design analysis needs to use an acceptable method to confirm that the fuel design limits and the neutronic design limits (i.e. nuclear key safety parameters) are not exceeded for operational states. It also needs to be confirmed that neither significant damage to the pressure boundary nor impairment of the capability to cool the core take place in the postulated reactivity accidents.

To satisfy these fuel design limits, the new fuel assembly and a reference ICFM core design need to be analysed. The design limits for power densities (and thus for peaking factors) during normal operation need to be such that associated fuel design limits are met for anticipated transients and for LOCAs, respectively.

The limiting power distributions are then determined under the condition of maintaining the limits on power densities (and thus on peaking factors) in operation. These limiting power distributions can be maintained administratively (i.e. not by automatic scrams), with information and alarms available from the reactor instrumentation to keep the operator informed.

Compliance with the acceptance criteria for power distribution can be demonstrated by a reasonable probability of meeting associated fuel design limits within the expected operational range of the reactor, where the following need to be accounted for:

- The analytical methods and data used for the design calculations;
- The uncertainty analyses and experimental comparisons presented for the design calculations;
- The sufficiency of the design cases calculated covering times in cycle, rod positions, load following transients, etc.;
- Special problems, such as power spikes due to densification, possible asymmetries and misaligned rods.

As regards criticality during refuelling, the following need to be reviewed:

- Discussions and tables giving values of k_{eff} for single assemblies and for groups of adjacent fuel assemblies up to the number required for criticality;
- Assumptions for dry or wet storage of the assemblies.

For CANDU reactors, the design process defines an equilibrium ‘reference flux shape’, which defines the target channel power distribution for the reactor during operations. A range of acceptable deviations from these reference channel powers is permitted during operations, to accommodate the daily on-power fuelling operations and other operational occurrences, such as fuelling machine unavailability. Reactor physics simulations are performed to support each fuelling operation. The maximum deviation of centre core channel power from the reference is called the channel power peaking factor. A wide margin to trip is always maintained between the channel power peaking factor and the neutron overpower trip set point.

II.5. THERMOHYDRAULIC DESIGN VERIFICATION

II.5.1. Thermal design limit

A check on fulfilment of the DNB criterion is run in the thermohydraulic design and in safety analyses using qualified computer codes.

The statistical design limit DNBR (SDLDNBR) and safety analysis limit DNBR (SALDNBR) are established to estimate the reactor core DNBR margin. The SDLDNBR is determined using a statistical method in consideration of the uncertainties of reactor core parameters, of critical heat flux calculation by empirical correlation, of local heat flux and of coolant heating, ensuring a probability of not less than 95% at a confidence level of 95%. The SALDNBR is determined by adding the penalty coefficients (margins) to the SDLDNBR, as required by the regulatory authority.

Phenomena that may affect the DNBR limits in a negative direction, such as fuel densification or rod bowing, need to be considered as an appropriate design penalty. To meet the DNB criterion, the minimum DNBR value in normal operation and AOOs needs to meet the SALDNBR criteria. Protection against DNBR in reload designs is typically achieved by using a nuclear peaking limit. The overall level of DNB performance is established by the fuel design and the core operating conditions (core power level, temperature, pressure, flow) are fixed by the event or the transient being analysed. This allows the fuel radial and axial peaking to be the variables that determine the required level of DNBR performance.

This is typically further simplified by fixing either the axial power shape or the radial peaking value and calculating the other value. For example, if axial power distribution is fixed, a maximum radial peaking factor is calculated to yield the minimum DNBR value for the analysed case. At the end, the result of such analyses are fuel rod peaking values that provide acceptable DNB results and that can be applied to core design predictions on a reload basis to determine satisfactory thermohydraulic performance.

CANDU reactors are operated to the analysis based equilibrium reference flux shape and reactor heat generation load throughout their operating lives.

The reactor regulating system controls spatial zone thermal output within the parameters of the nominal flux shape, which is established by analysis and sometimes fine tuned during reactor operations. For example, during the pre-equilibrium period immediately after first commissioning or after core reload following reactor refurbishment, flux shape and thermal zone powers are managed through the loading of depleted fuel bundles and the implementation of temporary nominal flux shapes into the reactor regulating system. As core equilibrium is established, all depleted fuel bundles are removed from the core via normal fuelling procedures, and reference flux shapes in the reactor regulating system are returned to the design nominal. Adherence to the licensed fuel bundle and fuel channel power limits is always maintained. At all times the channel selections of the fuelling engineer (for daily fuelling) are based on frequent reactor physics simulations targeting strict adherence to the design basis nominal flux shape and in compliance with the regulated fuel bundle and fuel channel power limits.

II.5.2. Reactor coolant flow rate

The analysis of reactor core cooling utilizes minimum primary coolant flow rate (G_{min} — thermal design flow rate) reduced by reactor core bypass flow. Bypass flow values are determined considering the following core bypass flows: reactor outlet nozzle flows and core baffle cooling flows, as well as the guide tube and instrumental tube flows. Both minimum and maximum bypass flow need to be determined. The maximum design coolant flow rate (G_{max} — mechanical design flow rate) is used to determine hydraulic loads applied on fuel assemblies and reactor core components.

Thermohydraulic analyses of the reactor core and hydraulic loads calculations consider reductions in average flow rate through a hot assembly and non-uniform distribution of flow rate in the reactor inlet chamber over the core cross-section.

As CANDU reactors age, coolant flow rates through fuel channels can vary slightly from the as designed condition due to fuel channel diametral changes caused by radiation induced pressure tube creep. The operating utilities monitor these conditions carefully and take remedial measures as required. These measures can include reactor regulating system nominal changes, bulk reactor power reductions and replacement of individual pressure tubes. If these degraded conditions are extensive, they can be an important factor in the decision to execute full scale reactor refurbishment.

II.5.3. Thermohydraulic instability

Fulfilment of the criterion on hydrodynamic stability of primary coolant flow is verified by circuit stability analysis, which is based on analysis of the main coolant pump head characteristics and the primary circuit hydraulic characteristics for reactor plant operation.

Coolant parameters, primary circuit hydraulic characteristics, main coolant pump characteristics and core characteristics with the chosen fuel assembly ensure stable coolant flow in the reactor and core that is confirmed by long term experience of reactor plant operation with various fuel assembly types.

II.5.4. Hydraulic loads

Hydraulic loads for this evaluation are reviewed as described in Ref. [102]. The hydraulic force on the fuel assembly depends on the rate of the coolant flow and the coefficients of pressure loss along the fuel assembly. The analysis for the required hold down force is conducted with a conservatively determined high flow rate (i.e. mechanical design flow rate).

Appendix III

MAJOR STEPS OF NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS

The major fabrication steps and the respective quality control steps are presented in Table III.1.

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Powder			
Install the UF_6 cylinder in the autoclave	None		
UF_6 vaporization from autoclave	Vaporization process needs to be stable to feed the reactor with a constant UF_6 flow		
UF_6 hydrolysis	Hydrolysis process needs to be stable to feed the next calciner with homogeneous UO_2F_2		
UO_2F_2 calcination	Driving parameters (heater temperature, vapour flow, H_2 flow) controlled to achieve a full transformation of UO_2F_2 into UO_2 throughout the calciner		

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
UO_2 powder cooling	Powder is cooled down and dried into an inert N_2 flow		
Powder sampling and powder quality measurement	Powder conformity features are measured by destructive tests on samples (U_5 content, isotopy, O/U ratio, low level of moisture, low level of impurities (F) and powder sintering ability features)	Yes	UO_2 sintering features are important because they have a direct impact on fuel pellet–cladding gap size evolution during irradiation and may result in undesired PCMI[a] cladding stresses (see Section I.1.1)
Powder storage container	None		
Powder preparation	This operation contributes to powder homogeneity		
Powder pneumatic transfer	Movement through pneumatic transfer cannot alter the powder homogeneity (prevent reagglomeration)		

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Pellet			
Procure and prepare additives	None		
Blending	Constitute the blending batch ($UO_2 + U_3O_8$ + burnable absorber + pore former) Achieve a homogenized batch at the end of blending cycle Blending batch traceability is to be kept	Yes	Blending batch determines pellet pore size distribution and density, which are important for pellet dimensional changes during operation (densification and swelling) Additionally, a homogenous distribution of fuel and burnable absorbers (i.e. gadolinia) is assumed in the fuel performance codes (see Section I.1.1)
Prepressing and granulation	The powder is pressed to form low density flat pellets. These pellets are granulated in a mill (granulator) and then forced to pass through a calibrated sieve, generating granulates of homogeneous size. The objective of this activity is to obtain a material with good fluidity for pressing		
Conditioning	Spheroidization rounds the shape of granulate Lubricant addition needs to be performed carefully to be well distributed over the volume of the batch		

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Pellet press	Press force is set to achieve a given density for the ‘green’ pellet or uncooked pellet		
Pellet boot loading	Automatic equipment arranges pellet layers into the sintering boxes		
Movement of the pressed pellets boot to the sintering furnace	None		
Sintering furnace	The main parameters to operate the sintering process are temperature profile, flow of gas sent inside the furnace (N_2, H_2), level of moisture, furnace atmosphere, speed of boxes In the preheating zone the organic additives are evacuated At the middle of the furnace, the pellets obtain the final sintered features required by the designer Sintered pellet samples are collected at the exit of the furnace to check pellet density	Yes	Fuel pellet density is a data input for the fuel performance codes used to quantify the available design margins (PCMI, DNBR[b], etc.). Any deviation has an impact on the outcomes (see Section I.1.1)

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Pellet diameter grinder	Grinder process is monitored through a control chart Diameter of ground pellet is measured periodically Grinder is reset when deviation is observed on diameter control chart	Yes	Pellet diameter is an important input data of the fuel performance code used to calculate the pellet–clad gap evolution during irradiation. It impacts on the fuel failure margins (PCMI) and the heat transfer margins (DNBR) (see Section I.1.1)
Pellet diameter sorting	A machine automatically inspects every pellet in line and immediately ejects any pellet whose diameter is out of the specified range Some lines are also equipped with equipment to check pellets for surface imperfections	Yes	Any fuel fragments generated during pellet insertion may result in premature PCMI rod failures if they are squeezed into the fuel pellet–cladding gap. Further, surface imperfections (missing pellet surface anomalies) can result in high local stresses in the cladding and premature PCMI/PCI rod failures
Pellet installation on trays	A machine automatically arranges pellets on the trays		
Pellet visual aspect sorting	If automated surface inspection is not in place, all pellets are inspected to eliminate any showing abnormal visual signs (chips or cracks of unacceptable size)	Yes	Same as for pellet diameter sorting above

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Pellet sampling and quality measurement	Pellet conformity features can be measured by destructive tests on samples (e.g. U_5 content, O/U ratio = 2, low level of impurities)	Yes	
Pellet tray storage	Trays of pellets are stored until they are used to constitute fuel rods		
Rod			
Rod component preparation (tube, plug, spring, spacer, fill gas)	Components as received are clean and ready to be used When unpacking and installing the components on the line, cleanliness is confirmed through an inspection	Yes	
Bottom plug welding	Weld quality is obtained first by controlling the welding process; some of the main parameters of the process need to be operated within predetermined range Periodic weld samples are created and appropriate destructive tests (tensile test, metallographic examinations, corrosion measurements) on these samples prove that the process is efficient and stable	Yes	An imperfect end plug weld may not be leak tight and lead to water and steam ingress during operational conditions. As a result, secondary hydriding could occur away from the primary leak and massive embrittlement of the cladding can result in rod failure and fuel fragments dispersal in the coolant

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Weld dimension and visual examination	In addition to the above actions to control and check the quality of the welding process, different examinations (dimensional, visual) are performed on-line on every rod	Yes	Same as above. Additionally, oversized weldments can damage grid springs during fuel rod loading into the skeletons, which, in turn, may be a cause of undue vibrations and fretting wear
Pellet loading	The parameters of the loading process need to be selected to prevent any damage on the pellets		Fuel fragments (chips) in the gap generate highly concentrated stresses in the cladding and can result in premature PCMI rod failures
Pellet stack final length adjustment	The conformity of the length of the pellet stack is checked and pellets are added or removed as necessary	Yes	Fuel stack dimensions should be consistent with the neutronic design of the fuel core
Tube end face preparation	After the pellet stack has been loaded, it is cleaned to make sure that the end face of the tube, where the top weld is, is free of any random UO_2 debris		The weld should not be polluted by foreign material to ensure leak tightness during irradiation
Spring installation and top end plug installation (not welded)	None		The spring fixes the fuel pellet stack, preventing axial movements of the pellets, avoiding pellet damage (chipping) and the creation of axial gaps that may cause local LHR[c] spikes and/or local cladding flattening

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Pressurization and top plug installation and welding	He is used in the welding chamber at the pressure required by the designer for rod pressure When the pressure has been installed in the rod, the top plug is pushed against the tube end and the weld closing the rod can be performed; alternatively, the rod can be pressurized after welding of the upper end plug using a filling hole that is sealed afterwards The welding process is controlled as already described for the bottom plug (welding parameters control and tests on weld samples)	Yes	The initial rod internal pressure is part of the fuel design and used as input data for the fuel performance codes used to assess the available in-reactor margins. Any deviation of the initial rod internal pressure will have an impact on the pellet–clad gap closure evolution (PCMI, PCI–SCC[d], heat transfer/DNBR, cladding lift-off margins, etc.) and the post-DNB[e] behaviour of the fuel rods (ballooning and burst during a LOCA[f] transient). It has an impact on fission gas releases and on the end of life rod internal pressure (waste management) (see Section I.1.1)

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Top weld dimension and visual examination	Examinations (dimensional, visual) are performed online on every rod, as described for the bottom plug	Yes	An imperfect end plug weld may not be leak tight and lead to water and steam ingress during operational conditions. As a result, secondary hydriding could occur away from the primary leak and massive embrittlement of the cladding can result in rod failure and fuel fragment dispersal in the coolant. Additionally, oversized weldments can damage grid springs during fuel rod loading into the skeletons, that, in turn, may be a cause of undue vibrations and fretting wear

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Gamma scanner	Rod conformity features are verified from the gamma scanner, such as conformity of the pellet stack constitution (pellet to pellet axial gaps, pellet column zone lengths with different enrichment or burnable poison concentration, plenum length, etc.) Gamma scans can detect any abnormal events that could have changed the ^{235}U linear density along the pellet stack, such as a pellet of different enrichment or a gap in the pellet stack	Yes	Fuel performance codes used to assess the available safety margins assume that the fuel stack is homogeneous and consistent with the technical specifications. Any deviation may have an effect on the local cladding stresses (fragments in the gap, abnormal pellet density, axial pellet gap) or on the local cladding hydriding embrittlement (axial pellet–pellet gap) An error in the enrichment may have a significant impact on the power and then the safety margins
Densitometry	Rod conformity features are verified using densitometry picture analysis (spring plenum dimension, spring presence, a gap in the pellet stack)	Yes	Same as above
Rod weight	Measurement could give complementary information on the UO_2 quantity per rod	Yes	UO_2 quantity has a direct impact on the core neutronic design and energy production

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
He leak test	Rod conformity features are verified using the He leak test If the helium sensor does not detect any helium, the rod welds are tight	Yes	Initial fuel rod leak tightness is assumed in the fuel performance codes. Leakage generates secondary hydriding and fuel rod failures during normal operation (including potential fuel fragments dispersal in the coolant)
Rod visual examination	Attentive examination of the rod confirms that the rod tube has not experienced any concerning event (mechanical damage, incidental pollution) during travel on the line	Yes	Any surface defects can generate an incipient crack. Initial straightness of the fuel rod is an important parameter regarding DNBR margins
End part			
Procure the incoming metallic raw materials (stainless steel, Ni based alloys)	Quality of the raw materials used is obtained through the following actions: Technical requirements for materials delivered are specified to the selected supplier; Supplier manufacturing sequence is reviewed; Source inspections are performed to release the materials lots	Yes	

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Operate the different processes used to make subcomponents (machining, cutting, bending)	When implementing the different processes to obtain the parts, it needs to be verified that every process selected can generate parts that contribute to the expected final quality (part dimensional features and materials surface features)		
Operate the different fastening processes (welding, brazing)	Weld or brazing quality is obtained first by controlling the process; some of the main parameters of the process need to be operated within predetermined range Periodic samples are created and appropriate destructive tests (tensile test, metallographic examinations, corrosion) on theses samples prove that the process is stable (repetitive)	Yes	
Operate different finishing processes (electrical discharge machining, electropolishing, sand blasting, pickling)	The quality of the resulting parts is obtained first by controlling the process; some of the main parameters of the process need to be operated within an appropriately selected range	Yes	

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Operate different materials heat treatment processes	Tests on part samples prove that when operating within this range the resulting part has the expected materials quality		
Operate different cleaning processes	Cleanliness of the parts is obtained by controlling the main parameters of the washing process (washing cycle, washing, rinsing liquid properties) Tests on part samples prove that when operating within this range the resulting parts will have the expected cleanliness quality	Yes	
Perform dimensional inspection of the parts performed at different manufacturing steps	Measurements of part dimensions are performed at some steps of the manufacturing sequence using tactile and optical coordinate measuring machines and gauges	Yes	Fuel rod and fuel assembly designs are compiled with given dimensions. Any deviation may have an impact on the safety margins that have been assessed by the designers (e.g. support given to the fuel rod by the grid springs within the grid cells)

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Perform visual examinations of the parts at different manufacturing steps	Visual examinations of parts are performed at some steps of the manufacturing sequence and address the following aspects of quality: Cleanliness of the part examined; No presence of foreign objects in the part examined	Yes	Presence of foreign material can result in cladding fretting wear and fuel rod leakage at the beginning of the in-reactor irradiation
Procurement of incoming metallic raw materials (Zr alloy strips, Ni based alloy strips)	The quality of the raw materials used is ensured through the following actions: Technical requirements on materials delivered are specified to the selected supplier; Supplier manufacturing sequence is reviewed; Source inspections are performed to release the materials lots	Yes	The mechanical properties of the materials used to manufacture the fuel products are input data for the fuel performance codes used to assess the available margins

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Operate stamping processes to obtain spacer and mixing grid subcomponents: straps and springs	When implementing the stamping processes to obtain the parts, it needs to be verified that every process selected can generate parts that will contribute to the expected final quality with respect to the dimensional features and the materials features The quality of the parts is over checked via regularly performed surveillances at the sub-suppliers' manufacturing shop	Yes	
Operate different cleaning processes (strap washing, spring washing)	Cleanliness of the parts is obtained by controlling the main parameters of the cleaning process (washing cycle, washing, rinsing liquid properties) Tests on samples prove that when operating within this range the resulting parts have the expected cleanliness quality	Yes	

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Operate different materials heat treating processes (Ni based alloys ageing treatment for spacer grid components and complete spacer grids, Zr strap stress release treatment)	Quality of the treated parts is obtained first by controlling the process; some of the main parameters of the process need to be operated within selected ranges Tests on part samples prove that when operating within these ranges the resulting part has the expected materials quality	Yes	The mechanical properties of the small fuel assembly parts are input data for the simulation codes used to assess the fuel assembly behaviour throughout its lifetime. In particular, stress corrosion resistance of the grid spring providing the support to the fuel rods to prevent GTRF[g] should be such that no failures occur throughout the fuel assembly lifetime
Operate different mounting and fastening processes (spacer welding, spring welding)	Weld quality is obtained first by controlling the welding process; some of the main parameters of the process need to be operated within predetermined range Periodic weld samples are created and appropriate destructive tests (tensile test, metallographic examinations, corrosion) on these samples prove that the process is stable	Yes	The overall structural resistance and stiffness of the fuel assembly are input data for the calculation codes used to assess the fuel assembly behaviour during its in-reactor lifetime. Fuel assembly bow, debris trapping, fuel assembly growth, etc. are examples of key mechanisms that may impact on operating margins (control rod drop times, DNBR margins, GTRF, handling, neutronic instabilities, etc.)

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Perform dimensional inspection of the parts manufactured at the different manufacturing steps (strap, spring, strap with spring installed, completed spacer)	Measurements of part dimensions are performed at the successive steps of the manufacturing sequence (strap, spring, strap with spring installed, completed spacer)	Yes	Same as above
Final visual examination of the parts made at the different manufacturing steps (strap, spring, strap with spring installed, completed spacer)	Visual examinations of parts are performed at the different steps of the manufacturing sequence and address the following aspects of quality: Conformance of the weld points or weld fillets; Cleanliness of the parts examined; No presence of foreign object in the parts examined; Strap configuration according to the technical file	Yes	Same as above
Positioning of the parts to be fastened on the skeleton welding equipment	Accurate position of the different parts is required to contribute to the final dimensional quality of the completed skeleton		Same as above

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Different fastening processes (welding, swaging)	Weld or swaging quality is obtained first by controlling the fastening process; some of the main parameters of the process need to be operated within a predetermined range Periodic samples are created and appropriate destructive tests (tensile test, metallographic examinations, corrosion) on these samples prove that the process is stable	Yes	Same as above
Measurement of dimensional features of skeleton	Measurements cover overall dimensions length, straightness and verticality For some products, these are performed on the level of the completed fuel assembly	Yes	Same as above
Final visual examination of skeleton	Visual examinations of skeleton address the following aspects of quality: Conformance of the weld points and swaging junctions; Cleanliness of the skeleton examined; No presence of foreign objects	Yes	Same as above

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Assembly			
Install and position the skeleton to be loaded	The skeleton needs to be accurately installed and strongly clamped on the assembly bench		
Rod loading operation	The pulling/pushing forces the rod to slide through the spacers		Rod loading into the skeleton may result in cladding shavings, which in turn could become a source of debris, generating fuel rod fretting wear and leakage. Further, if the shavings accumulate under the grid spring/dimples ('gall balls'), it might cause some fuel rod loosening, or even small gaps, that, in turn, lead to GTRF The manufacturing process should be such that shavings are not produced or are minimized
End part installation	Accurate positioning of the parts needs to be attained to contribute to the final quality features of the completed assembly (verticality, length)		Design margins are based on prescribed fuel assembly dimensions. Any deviation will have a direct impact on the fuel assembly hold down spring system, which is designed to avoid fuel assembly lift-off and excessive fuel assembly bow

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Movement of the assembly from the horizontal position to the vertical position and further handling by a hoist system	When completed, the assembly is stored and handled in the vertical position Handling is performed at very low speed to minimize the risks of overload and damage		Any damage to the fuel assembly has to be analysed to confirm that it does not affect the design margins
Cleaning operation	Different kinds of cleaning can be applied. During this operation most of the cladding shavings should be removed		
Measurement of the dimensions of the completed fuel assembly	Features measured include assembly length, verticality, straightness and spaces between rods Some features could already have been checked on a skeleton stage	Yes	Design margins are based on prescribed fuel assembly dimensions. Any deviation has a direct impact on the fuel assembly hold down spring system, which is designed to avoid fuel assembly lift-off and excessive fuel assembly bow
Check for interlocking of fuel elements (for CANDU[h] reactors)	Use a bent tube gauge to verify that fuel elements have not become interlocked during manufacturing and handling, and that the fuel bundle maximum diameter is as required	Yes	Interlocking of fuel elements increases fuel bundle diameter at mid-plane and hinders fuel bundle passage through fuel channels. The issue is more pronounced in aged (sagged) fuel channels

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Check control rod insertion (PWRs[i] only)	The control rods need to be moved without any significant friction	Yes	Control rods and control rod drop times are key safety features to ensure safe reactor core control
Final visual examination of the assembly	Attentive examination checks that the assembly has not experienced any concerning event (mechanical damage, pollution) during travel on the line Protect the assembly inside a plastic bag	Yes	Any damage to the fuel assembly has to be analysed to confirm that it does not affect the design margins
Fuel channelling (BWRs[j] only)	Mount the fuel channel to the fuel assembly		
Storage in rack of released assemblies	The assembly is handled and stored in a vertical position The handling and storage conditions look like the further conditions applied in the nuclear plant		

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Storage of fuel bundles in shipping boxes (CANDU reactors)	Completed fuel bundles are strapped at their mid-plane to prevent element interlocking during transportation, wrapped in plastic and loaded horizontally onto foam packing in shipping/storage boxes. Accelerometers are mounted on the shipment boxes to record any occurrence of rough handling between loading of boxes at the fuel manufacturer and unloading of fuel bundles at the utility site		Shipment to the reactor site has to be completed without any damage to the manufactured fuel bundles, including inadvertent interlocking of fuel elements due to rough handling. All fuel bundles are checked thoroughly again at the reactor site, before they are loaded into core
Prepare the container used to ship the assemblies	There are precise quality requirements for the fixture where the assembly is placed and clamped to prevent any damage at the assembly installation		
Install the assembly into the container	The assembly is installed and clamped on the container fixture Then the assembly is moved at very low speed to prevent any damage Detailed instructions define the way the assembly needs to be installed and clamped on the supporting fixture		

TABLE III.1 MAJOR STEPS IN NUCLEAR FUEL MANUFACTURING AND RESPECTIVE QUALITY CONTROL STEPS (cont.)

Successive activities to manufacture the product or check the product quality	Comments on the main product quality expectation being considered when driving the manufacturing or inspection activities	Quality control or sampling step	Relevance to fuel reliability and performance in operational states
Handle the loaded container when preparing the shipping of the assemblies to the nuclear plant	The loaded container is moved at very low speed to prevent any damage The loaded container should be instrumented with accelerometers to confirm that excessive acceleration has not occurred during shipping		

[a] PCMI: pellet–cladding mechanical interaction.
[b] DNBR: departure from nucleate boiling ratio.
[c] LHR: linear heat rating.
[d] PCI–SCC: pellet–cladding mechanical interaction/stress corrosion cracking.
[e] DNB: departure from nucleate boiling.
[f] LOCA: loss of coolant accident.
[g] GTRF: grid to rod fretting.
[h] CANDU: Canadian deuterium–uranium.
[I] PWRs: pressurized water reactors.
[j] BWRs: boiling water reactors.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Reactor Core for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-52, IAEA, Vienna (2019).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Fuel Cycle Objectives, IAEA Nuclear Energy Series No. NF-O, IAEA, Vienna (2013).

[3] NUCLEAR ENGINEERING INTERNATIONAL, Fuel design data, Nucl. Eng. Int. **54** (2009) 36–48. Also refer to https://www.neimagazine.com/.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Technical Review of Acceptance Criteria for Pressurized Heavy Water Reactor Fuel, IAEA-TECDOC-1926, IAEA, Vienna (2020).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Advances in Small Modular Reactor Technology Developments, a Supplement to the IAEA Advanced Reactor Information System (ARIS), 2020 edn, IAEA, Vienna (2020).

[6] STRASSER, A., RUDLING, P., Fuel Fabrication Process Handbook, Rep. ANTI 05-0008R, ANT International, Surahammar (2004).

[7] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Specification for Wrought Zirconium and Zirconium Alloy Seamless and Welded Tubes for Nuclear Service (Except Nuclear Fuel Cladding), ASTM B-353, ASTM, West Conshohocken, PA (2007).

[8] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Specification for Wrought Zirconium Alloy Seamless Tubes for Nuclear Reactor Fuel Cladding, ASTM B-811-02, ASTM, West Conshohocken, PA (2007).

[9] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels, NEA No. 7317, OECD/NEA, Paris (2018).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Failures in Water Reactors: Causes and Mitigation, IAEA-TECDOC-1345, IAEA, Vienna (2003).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Optimization of Water Chemistry to Ensure Reliable Water Reactor Fuel Performance at High Burnup and in Ageing Plants (FUWAC), IAEA-TECDOC-1666, IAEA, Vienna (2011).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Operation and Licensing of Mixed Cores in Water Cooled Reactors, IAEA-TECDOC-1720, IAEA, Vienna (2013).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, High Burnup Fuel: Implications and Operational Experience, IAEA-TECDOC-1798, IAEA, Vienna (2016).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Review of Fuel Failures in Water Cooled Reactors (2006-2015), IAEA Nuclear Energy Series No. NF-T-2.5, IAEA, Vienna (2019).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Coolant Chemistry Control and Effects on Fuel Reliability in Pressurized Heavy Water Reactors, IAEA-TECDOC-1942, IAEA, Vienna (2021).

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Progress on Pellet–Cladding Interaction and Stress Corrosion Cracking: Experimentation, Modelling and Methodologies Applied to Support the Flexible Operation of Nuclear Power Plants, IAEA-TECDOC-1960, IAEA, Vienna (2021).

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Fuel Handling and Storage Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-63, IAEA, Vienna (2020).

[18] ORGANISATION FOR ECONOMIC COOPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, Safety Margins Action Plan — Final Report, NEA/CSNI/R(2007)9, OECD/NEA, Paris (2007).

[19] INSTITUTE OF NUCLEAR POWER OPERATIONS, Excellence in the Management of Design and Operating Margins, Report 09-003, INPO, Atlanta, GA (2009).

[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Margins of Operating Reactors: Analysis of Uncertainties and Implications for Decision Making, IAEA TECDOC-1332, IAEA, Vienna (2003).

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Implications of Power Uprates on Safety Margins of Nuclear Power Plants, IAEA-TECDOC-1418, IAEA, Vienna (2004).

[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Power Uprate in Nuclear Power Plants: Guidelines and Experience, IAEA Nuclear Energy Series No. NP-T-3.9, IAEA, Vienna (2011).

[23] ZHANG, J., et al., "An improved fuel design specification and evaluation process for Belgian nuclear power plants", Proc. 2017 Water Reactor Fuel Performance Meeting (WRFPM/TOPFUEL2017), Jeju lsland, Republic of Korea, 10–14 September 2017 (2017).

[24] ZHANG, J., WAECKEL, N., YUEH, K., "Utility perspective on needs related to nuclear fuel operation", Proc. OECD/NEA Workshop on Nuclear Fuel Modelling to Support Safety and Performance Enhancement for Water-Cooled Reactors, Paris, 7–9 March 2017, OECD/NEA, Paris (2021).

[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

[26] ZHANG, J., KOVTONYUK, A., SCHNEIDESCH, C., Towards a graded application of best estimate plus uncertainty methodology for non-LOCA transient analysis, Nucl. Eng. Des. **354** (2019) 110189.

[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

[28] OBERKAMPF, W. L., ROY, C. J., Verification and Validation in Scientific Computing, Cambridge University Press, Cambridge (2010).

[29] SWILER, L., "VVUQ best practices in computational science/engineering problems with some thoughts about extensions/limits to complex systems models", SAND2016-5737C, Complex Systems Working Group, 22–23 June 2016.

[30] DE ROCQUIGNY, E., DEVICTOR, N., TARANTOLA, S., Uncertainty in Industrial Practice: A Guide to Quantitative Uncertainty Management, Wiley, Chichester (2008).

[31] ZHANG, J., The role of verification & validation process in best estimate plus uncertainty methodology development, Nucl. Eng. Des. **355** (2019) 110312.

[32] TARANTOLA, A., Inverse Problem Theory and Methods for Model Parameter Estimation, Society for Industrial and Applied Mathematics (SIAM) (2005), E-book from https://www.siam.org/publications/books/e-books-for-institutions

[33] BACCOU, J., et al., SAPIUM: a generic framework for a practical and transparent quantification of thermal-hydraulic code model input uncertainty, Nucl. Sci. Eng. **194** (2020) 721–736.

[34] PETRUZZI, A., et al., Thermal-hydraulic system codes in nuclear reactor safety and qualification procedures, Sci. Technol. Nucl. Install. **2008** (2008).

[35] GEELHOOD, K.J., BALES, M.E., PORTER, I.E., “Code qualification for traditional LWR fuel”, Proc. TopFuel, Prague, Czech Republic, 30 September–4 October 2018 (2018).

[36] ZHANG, J., SCHNEIDESCH, C.R., “Development, qualification and application of the coupled RELAP5/PANTHER/COBRA code package for integrated PWR accident analysis”, Use and Development of Coupled Computer Codes for the Analysis of Accidents at Nuclear Power Plants, IAEA-TECDOC-1539 Companion CD, IAEA, Vienna (2007) CD-ROM.

[37] AVRAMOVA, M., ABARCA, A., HOU, J., IVANOV, K., Innovations in multi-physics methods development, validation, and uncertainty quantification, J. Nucl. Eng. **2** (2021) 44–56.

[38] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/NUCLEAR ENERGY AGENCY, CSNI Integral Test Facility Validation Matrix for The Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients, Committee on the Safety of Nuclear Installations, OCDE/GD(97)12, OECD/NEA, Paris (1996).

[39] AKSAN, N., et al., Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, Vol. I: Phenomena Characterization and Selection of Facilities and Tests, Vol. II: Facility and Experiment Characteristics, Committee on the Safety of Nuclear Installations, CSNI Report, OECD/NEA/GD (94)/82, OECD/NEA, Paris (1994).

[40] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/NUCLEAR ENERGY AGENCY, CSNI International Standard Problems (ISP), Brief Descriptions (1975–1999), Committee on the Safety of Nuclear Installations, NEA/CSNI/R(2000)5, OECD/NEA, Paris (2000).

[41] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/NUCLEAR ENERGY AGENCY, Contribution from Twenty-Two Years of CSNI International Standard Problems, Committee on the Safety of Nuclear Installations, NEA/CSNI/R(97)29, OECD/NEA, Paris (1998).

[42] MISFELDT, I., “The D-COM blind problem on fission gas release: experimental description and results, summary report”, Proc. OECD-NEA-CSNI/IAEA Specialists’ Meeting on Water Reactor Fuel Safety and Fission Product Release in Off-Normal and Accident Conditions, Risø, Roskilde, Denmark, 16–20 May 1983, IAEA-IWGFPT/16 IAEA, Vienna (1983) 411–421.

[43] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling at Extended Burnup (FUMEX), IAEA-TECDOC-998, IAEA, Vienna (1998).
[44] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling at Extended Burnup (FUMEX-II), IAEA-TECDOC-1687, IAEA, Vienna (2012).
[45] INTERNATIONAL ATOMIC ENERGY AGENCY, Improvement of Computer Codes Used for Fuel Behaviour Simulation (FUMEX-III), IAEA-TECDOC-1697, IAEA, Vienna (2013).
[46] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Modelling in Accident Conditions (FUMAC), IAEA-TECDOC-1889, IAEA, Vienna (2019).
[47] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, International Fuel Performance Experiments (IFPE) Database, OECD-/NEA, Paris, http://www.oecd-nea.org/science/fuel/ifpelst.html
[48] ROSSITER, G., et al., "OECD/NEA benchmark on pellet–clad mechanical interaction modelling with fuel performance codes", NEA/CSNI/R(2018)9, OECD/NEA, Paris (2018).
[49] DOSTÁL, M., et al. "OECD/NEA benchmark on pellet–clad mechanical interaction modelling with fuel performance codes: impact of number of radial pellet cracks and pellet–clad friction coefficient", Proc. TopFuel, Prague, Czech Republic, 30 September – 4 October 2018 (2018).
[50] SOBA, A., et al., "OECD-/NEA benchmark on pellet–clad mechanical interaction modelling with fuel performance codes: influence of pellet geometry and gap size", Proc. TopFuel Santander, Spain, 24–28 October 2021 (2021).
[51] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT-/ NUCLEAR ENERGY AGENCY, CSNI Report on Benchmark Calculations on Halden IFA-650 LOCA Test Results, NEA/CSNI/R(2010)6, OECD/NEA, Paris (2010).
[52] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT-/ NUCLEAR ENERGY AGENCY, Fuel Fragmentation, Relocation and Dispersal (FFRD), NEA/CSNI/R(2016) OECD/NEA-CSNI WGFS Report, OECD/NEA, Paris (2016).
[53] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/NUCLEAR ENERGY AGENCY, RIA Fuel Codes Benchmark, Volume 1, NEA/CSNI/R(2013)7, OECD/NEA, Paris (2013).
[54] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, Reactivity Initiated Accident (RIA) Fuel Codes Benchmark Phase-II: Vol. 1: Simplified Cases Results — Summary and Analysis, NEA/CSNI/R(2016)6/VOL1, OECD/NEA Paris (2016).
[55] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, Reactivity Initiated Accident (RIA) Fuel Codes Benchmark Phase-III: Uncertainty and Sensitivity Analyses, NEA/CSNI/R(2020), OECD/NEA Paris (2020).
[56] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, Reactivity Initiated Accident (RIA) Fuel Codes Benchmark Phase-II: Uncertainty and Sensitivity Analyses, NEA/CSNI/R(2017)1, OECD/NEA, Paris (2017).

[57] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, WGFS RIA Fuel Rod Codes Benchmark: Phases 1-3 Synthesis Report, NEA/CSNI/R(2020)10, OECD/NEA, Paris (2020).
[58] ROY, C.J. and OBERKAMPF, W.L., A Comprehensive Framework for Verification, Validation, and Uncertainty Quantification in Scientific Computing, Computer Methods in Applied Mechanics and Engineering **200** (2011) 2131–2144.
[59] RAMM, A.G., Inverse Problems, Mathematical and Analytical Techniques with Applications to Engineering, Springer (2005).
[60] GLAESER, H., GRS method for uncertainty and sensitivity evaluation of code results and applications, Sci. Technol. Nucl. Install. **2008** (2008) 798901.
[61] MARTIN, R.P., PETRUZZI, A., Progress in international best estimate plus uncertainty analysis methodologies, Nucl. Eng. Des. **374** (2021) 111033.
[62] ZHANG, J., SEGURADO, J., SCHNEIDESCH, C., “Towards an industrial application of statistical uncertainty analysis methods to multi-physic modelling and safety analyses”, Proc. OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations, Barcelona, Spain, 16–18 November 2011 (2011).
[63] MARCHAND, O., ZHANG, J., CHERUBINI, M., “Uncertainty and sensitivity analysis in reactivity-initiated accident fuel modeling: synthesis of organisation for economic co-operation and development (OECD)/nuclear energy agency (NEA)" benchmark on RIA codes phase-II”, Proc. 2017 Water Reactor Fuel Performance Meeting (WRFPM2017), Jeju Island, Republic of Korea, 10–14 September 2017 (2017).
[64] ZHANG, J., BOULORÉ, A., “IAEA FUMAC benchmark on the uncertainty and sensitivity analysis for fuel rod code simulation of the Halden LOCA test IFA-650.10”, Trans. TopFuel 2018, Prague, Czech Republic, 30 September–4 October 2018 (2018).
[65] WU, X., XIE, Z., ALSAFADI, F., KOZLOWSKI, T., A comprehensive survey of inverse uncertainty quantification of physical model parameters in nuclear system thermal-hydraulics codes, Nucl. Eng. Des. **384** (2021) 111460.
[66] WU, X., KOZLOWSKI, T., MEIDANI, H., Kriging-based inverse uncertainty quantification of nuclear fuel performance code BISON fission gas release model using time series measurement data, Reliab. Eng. Syst. Saf. **169** (2018) 422–436.
[67] ZHANG, J., et al, “An improved fuel design specification and evaluation process for Belgian nuclear power plants”, Proc. TopFuel 2017, Jeju Island, Republic of Korea, 2017 (2017).
[68] DRUENNE, H., et al., “The belgian approach to in-core fuel management with mixed fuel assemblies”, Proc. TopFuel, Manchester, 2–6 September 2012 (2012).
[69] OELRICH, R.L., Jr., et al., “Fuel performance risk assessment during loading pattern development,” Proc. Top Fuel, Manchester, 2–6 September 2012 (2012).
[70] VIOUJARD, N., et al., “Crud/corrosion risk assessment tools of AREVA are minimizing fuel crud/corrosion risks for US plants,” Proc. Top Fuel, Manchester, UK, 2–6 September 2012 (2012).

[71] ZHANG, J., DETHIOUX, A., UMIDOVA, Z., DRIEU, T., SCHNEIDESCH, C., "Development and applications of a loading pattern PCI risk assessment tool for flexible core design", Pellet–Clad Interaction (PCI) in Water-Cooled Reactors: Paper Presented at Workshop of NEA Working Group on Fuel Safety (WGFS), Lucca, Italy, 22–24 June 2016, NEA/CSNI/R(2018)9, OECD/NEA, Paris, 2018.
[72] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Quality Management Systems — Fundamentals and Vocabulary, ISO 9000:2015, ISO, Geneva (2015).
[73] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Nuclear Safety and Security Glossary, Non-serial Publications, IAEA, Vienna (2022).
[74] INTERNATIONAL ATOMIC ENERGY AGENCY, Quality Assurance and Quality Control in Nuclear Facilities and Activities: Good Practices and Lessons Learned, IAEA-TECDOC-1910, IAEA, Vienna (2020).
[75] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).
[76] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Quality Management Systems — Requirements, ISO 9001:2015, ISO, Geneva (2015).
[77] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, Safety Guide Series No. GS-G-3.1, IAEA, Vienna (2006).
[78] HELMERSSON, B., YOUNG, M., WIKMARK, G., HAHN, P., "Maintaining and improving fuel reliability", WRFPM TopFuel 2009 (Proc. Water Reactor Fuel Perf. Mtg Paris, 2009), SFEN, Paris (2009) 85–93.
[79] EDSINGER, K., et al., "Recent US fuel reliability experience", Proc. Water Reactor Fuel Perf. Mtg Paris, 2009, SFEN, Paris (2009) 100–106.
[80] TOMPKINS, B., Advancing the cause of fuel reliability, Nucl. News **51** (2008) 34–36.
[81] STRASSER, A., EPPERSON, K., HOLM, J., RUDLING, P., "Design reviews for reliable fuel performance", Proc. LWR Fuel Perf. Mtg Orlando, 2010, ANS, LaGrange Park, IL (2010) 1–10.
[82] SHEWHART, W.A., Economic Control of Quality of Manufactured Product, D. Van Nostrand Co., New York (1931).
[83] SHEWHART, W.A., Application of Statistical Methods to Manufacturing Problems, Bell Telephone Labs, New York (1938).
[84] BAUR, K., VON COLLANI, E., Nuclear Fuel Quality Management Handbook, Vol. I: Fabrication, Operation, Disposal and Transport of Nuclear Fuel, ANT International, Skultuna (2010).
[85] GRANT, E.L., LEAVENWORTH, R.S., Statistical Quality Control, 6th edn, McGraw-Hill, New York (1988).
[86] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on Destructive Examination of Water Reactor Fuel, Technical Report Series No. 385, IAEA, Vienna (1997).
[87] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Guide for Establishing Calibration for a Measurement Method Used to Analyze Nuclear Fuel Cycle Materials, ASTM C1156 (Rev. 3), ASTM, West Conshohocken, PA (2003).

[88] AMERICAN NATIONAL STANDARDS INSTITUTE, Calibration Laboratories and Measuring and Test Equipment General Requirements, ANSI Z540-1-1994, ANSI, Washington, D.C. (1994).
[89] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Measurement Management Systems: Requirements for Measurement Processes and Measuring Equipment, ISO 10012:2003, ISO, Geneva (2003).
[90] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Guide for Qualification of Laboratory Analysts of Nuclear Fuel Cycle Materials, ASTM C1297 (Rev. 3), ASTM, West Conshohocken, PA (2011).
[91] AMERICAN SOCIETY FOR NONDESTRUCTIVE TESTING, Recommended Practice, Personnel, Qualification and Certification in Non-destructive Testing, 8th edn, ASNT-TC-1A, ASNT, Columbus, OH (2011).
[92] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Design and Quality Assurance Practices for Nuclear Fuel Rods, ASTM C 934 (Rev. 2), ASTM, West Conshohocken, PA (1990).
[93] WORLD ASSOCIATION OF NUCLEAR OPERATORS, Managing Core Design Changes, SOER 2004-1, WANO, London (2004).
[94] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, Nuclear Fuel Safety Criteria Technical Review, 2nd edn, OECD/NEA, Paris (2012).
[95] INTERNATIONAL ATOMIC ENERGY AGENCY, Analysis of Differences in Fuel Safety Criteria for WWER and Western PWR Nuclear Power Plants, IAEA-TECDOC-1381, IAEA, Vienna (2003).
[96] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT/ NUCLEAR ENERGY AGENCY, Applicability of Nuclear Fuel Safety Criteria to Accident Tolerant Fuel Designs, CSNI Technical Opinion Papers No. 19, Nuclear Safety, OECD 2022, NEA No. 7576 (2022).
[97] NUCLEAR REGULATORY COMMISSION, SCDAP/RELAP5/MOD 3.3 Code Manual, MATPRO — A Library of Materials Properties for Light-Water-Reactor Accident Analysis, NUREG/CR-6150, Vol. 4 (Rev. 2), USNRC, Washington D.C. (2001).
[98] NUCLEAR REGULATORY COMMISSION, Standard Review Plan, Section 4.2, Fuel System Design, NUREG-0800 (Rev. 3), USNRC, Washington D.C. (2007).
[99] NUCLEAR REGULATORY COMMISSION, Code of Federal Regulations Title 10, Part 50 — Domestic Licensing of Production and Utilization Facilities, Appendix A, General Design Criteria for Nuclear Power Plants, 10 CFR Part 50, USNRC, Washington D.C. (2020).
[100] NUCLEAR REGULATORY COMMISSION, Code of Federal Regulations Title 10, Part 50 — Domestic Licensing of Production and Utilization Facilities, §50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, 10 CFR Part 50, USNRC, Washington D.C. (2020).
[101] NUCLEAR REGULATORY COMMISSION, Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents, Regulatory Guide RG-1.236, USNRC, Washington D.C. (2020).

[102] NUCLEAR REGULATORY COMMISSION, Standard Review Plan, Section 4.4, Thermal and Hydraulic Design, NUREG-0800 (Rev. 3), USNRC, Washington D.C. (2007).
[103] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, Rules for Construction of Nuclear Facility Components, Section III, Division 1, 2015 edn, ASME, New York (2015).

Annex I

EXAMPLES OF FUEL INSPECTION AND SERVICE TECHNIQUES

I–1. SIPPING TEST

As a rule, sipping tests are not performed regularly during outages in all plants, but only when the contamination level of the primary circuit, which is monitored at all times, indicates that fuel rod failures may have occurred. In this case, the objective is always to preclude detrimental evolution of the radiation exposure by identifying and removing the leaking fuel rod(s) as quickly as possible.

The sipping test consists in heating up the fuel assembly to force the contaminated water or gaseous fission products to escape from the defective fuel rod(s) to be measured. The same phenomenon is observed if the coolant pressure is reduced.

When testing fuel assemblies inside the reactor core (in-core sipping is used in boiling water reactors (BWRs)), such an effect is achieved by using a sipping hood placed on several fuel assemblies. The sipping hood temporarily interrupts the coolant flow within the fuel assemblies. The fuel assemblies then begin to heat up, and the fission products released by the defective fuel(s) are trapped in the sipping hood system. Water samples, taken via the sipping hood, are continuously degasified and the resulting activity is measured. By comparing the value with the background activity, one can conclude that the fuel assemblies are sound or defective.

In pressurized water reactors (PWRs), the most common technology is the mast sipping technique. During an outage, the fuel assembly is lifted out of the core, and the water pressure around the fuel assembly is reduced. As a result, the fission products of the defective fuel rod are released into the coolant. As for in-core sipping, water samples are continuously collected and degasified to measure the activity and determine the fuel assembly status either sound or defective.

When analysing fuel assemblies long after core removal (i.e. several years), the standard sipping test based on natural residual heating or coolant pressure drop procedures no longer works. A sipping can that includes heaters and closing lids is used to compensate the low residual heat. The test procedure is then similar to the one used with the in-core sipping hood.

Vacuum can sipping was first devised to analyse gaseous activity at poolside and eliminate the slow and labour intensive laboratory analysis of radioactive water samples in the plant’s chemistry laboratory. Vacuum can sipping is accomplished by introducing an air bubble into a can or inverted box containing a fuel assembly. The added air bubble will result in most of the gaseous activity partitioning into the air bubble. In addition, the gas release can be enhanced by withdrawing some of the water from the bottom of the closed can, which reduces the external pressure of the fuel rods, hence the term ‘vacuum’ can sipping. In some systems, air is first injected into the can to drive out water before the can is sealed and vacuum pull. Can or vacuum can sipping is slow because it requires special fuel movement into and out of one or sometimes two sipping cans operated in parallel in comparison to in-mast or telescope sipping, which is accomplished during a normal fuel move — generally in the reactor core or from the core to the spent fuel pool. However, vacuum can sipping has the potential to have the greatest sensitivity due to the total enclosure of the leaker and the concentration of fission products. For this reason, vacuum can sipping is also a good choice for the identification of old leakers in the spent fuel pool and to confirm any ambiguous calls that may result from in-mast or telescope sipping.

I-2. INSPECTION DEVICE: MULTI-INSPECTION DEVICE EQUIPPED WITH A REMOTE CONTROLLED CAMERA

In many PWRs, the multi-inspection device can be combined with the new fuel elevators that have been implemented in the spent fuel storage area. The usual closed basket has been replaced with a new basket that is opened on one side, permitting full access to a moving fuel assembly inspection carriage. This x–y–z movable carriage is equipped with a video camera that is monitored remotely by staff located on the poolside away from the fuel assembly and the refuelling machine (there is a beneficial effect on employee radiation dose). The fuel assembly can rotate within the basket to inspect all of the fuel assembly faces. Similar services can be provided using a modified storage rack instead of the above mentioned basket.

As the refuelling machine is no longer used when the fuel assembly is inserted into the multi-inspection device, other handling activities can be carried out within the spent fuel pool.

Multi-inspection was developed to perform visual inspections in PWR plants but can provide other types of services, such as oxide layer thickness measurement for all the fuel rods or dimensional measurement of the fuel assembly structure and components. Rod cluster control assemblies (RCCAs) can be inspected in the same way.

In BWRs, the video camera is mounted on a hand operated swivel arm in front of the fuel assembly parking position and the dechannelling machine. The manipulator has a large degree of freedom and can be equipped with various inspection or measurement devices.

I-3. ULTRASONIC TESTING OF FUEL ASSEMBLIES

Ultrasonic testing devices have been developed to locate defective fuel rods within a light water reactor (LWR) fuel assembly bundle that has already been identified as leaking in a sipping test device. The system detects the presence of water in the defective fuel rod.

The test device is used in a spent fuel pool storage rack and is controlled remotely. The fuel bundle inspection, the defective fuel rod detection and on-screen display and printed results are performed quite quickly and can be carried out on both PWR and BWR fuel assemblies.

To save time, single rod sipping can be carried out together with a fuel rod exchange device that replaces the defective fuel rod with a dummy rod.

I-4. FUEL ASSEMBLY REPAIR AND SINGLE ROD INSPECTION

Irradiated fuel assemblies that contain defective fuel rods and have not yet reached their scheduled burnup can be reconstituted for further use in the reactor by replacing the defective fuel rods with either a fuel rod with the appropriate enrichment or a dummy rod. Fuel assembly reconstitution results in two main advantages for the plant owner:

(a) Since in most cases, defects affect only one fuel rod within the fuel assembly, which is not damaged, repair allows for operation until the anticipated discharge burnup of the fuel assembly (proper utilization of resources);
(b) Defective fuel rods can be further encapsulated to confine radioactive fuel particles. The capsule can be shipped safely to hot cells for further investigation (root cause analysis) or to waste disposal structures.

Some LWR nuclear power plants are equipped with standard fuel assembly repair equipment, which can be used to detect and replace defective fuel rods and to withdraw individual fuel rods for further extensive hot cell examination. To mutualize the equipment investment, PWR and BWR portable repair equipment is available and can be used on demand.

The fuel assembly repair equipment is suited for fuel assemblies with removable upper and/or lower end fittings, depending on the design of the assembly. A tilting device, which can be part of the PWR fuel assembly repair equipment, is used to turn the fuel assemblies with lower removable end fittings upside down.

A fuel assembly repair facility is capable of:

— Dechannelling the BWR fuel assemblies;
— Exchanging the upper and/or lower end fittings;
— Removing defective fuel rods, one at a time;
— Collecting fuel pellet debris and preparing subsequent safe storage;
— Rebuilding a fuel assembly with a new skeleton in case of damage to the skeletons (i.e. severe grid damage during handling) by reusing all the fuel rods of the damaged assembly.

When the defective fuel rod is removed from the fuel assembly bundle with the fuel rod exchange device, specific measurements can be performed during the operation (i.e. eddy current coil). If the defect detection process is uncertain, a second test, a single rod sipping test, is performed. In this case a hose needs to connect the poolside control cabinet, the mast sipping system (or the in-core sipping system) and the fuel rod exchange equipment. Water samples are analysed throughout the fuel rod lifting process within the fuel rod exchange device.

If the suspected fuel rod is eventually confirmed as sound, it is replaced in the fuel assembly and reloaded. On the other hand, if the fuel rod is clearly identified as defective, it is removed and replaced. The defective fuel rod is then inserted into the single rod inspection equipment for further investigation (i.e. confirmation of the defect location and nature using a more accurate eddy current probe moving along the fuel rod, oxide thickness measurement, diameter measurement, etc.). To prevent loss of fuel particles during the inspection (some of the leakers are heavily embrittled because of secondary hydriding), a fuel fragment catcher is positioned at the bottom of the inspected fuel rod.

Eventually, the repaired fuel assembly is checked visually and again positioned in the sipping test device to document the success of the whole operation. The repaired fuel assembly can then be reinserted into the reactor core.

I-5. RCCA TESTS AND REPAIR

RCCAs in LWRs are safety related core components. In addition to routine in-reactor functional tests to check drop times and the reactivity worth of the RCCAs, regular inspections and non-destructive examinations are undertaken during outage to confirm their mechanical and structural integrity.

Based on the inspection results, the operating performance (or maintenance strategy) of RCCAs can be usefully anticipated. RCCA inspections include the following activities:

— Visual inspection (PWR and BWR) to detect fretting wear.

— Eddy current probes to confirm the structural integrity of all the control rods and to measure the control rod swelling (PWR). The eddy current probes are mounted on the refuelling machine to allow fast and non-destructive RCCA inspection.
— Control rod profilometries to detect local clad thinning (PWR multi-inspection device).
— Control rod diameter evolution to anticipate the maintenance strategy (PWR multi-inspection device).

Measurements are compared with the anticipated damage index of the RCCA rods, which is used to build up the maintenance strategy. If one of the RCCA rods is damaged beyond a predetermined limit, the whole RCCA is replaced. In certain cases, a repair of the RCCA rod can be envisioned, to limit the amount of waste.

I-6. FAILED FUEL DETECTION, LOCATION AND SUPPORTING FUEL SERVICES ON-SITE IN PRESSURIZED HEAVY WATER REACTORS

A suspect defective bundle is detected by the delayed neutron system, by the feeder scan system or by heat transport chemical sampling during the operation of the reactor. The suspect defective bundle can be discharged from the reactor core during the refuelling of the suspect channel identified, for example, by the delayed neutron system. Confirmation that the defective bundle has been discharged should take place before the suspect fuel is placed in the defective fuel carousel in the spent fuel storage bay.

If it is not possible to confirm that a defective bundle has been discharged, the station should take appropriate action to ensure that all bundles that were residing in the suspect channel at the time of the delayed neutron scan have been removed.

The initial assessment of the defect cause is normally performed at on-site dedicated fuel inspection facilities. The inspections are performed underwater in the irradiated fuel storage bays, normally using visual inspection techniques (periscopes, telescopes, underwater cameras). In some cases, ultrasonic tooling has been developed to supplement these visual inspections. If multiple fuel failures are observed, the defect cause should be identified as soon as possible so that a corrective action plan can be implemented. If required for defect root cause assessments, more detailed examinations are undertaken in hot cells.

Annex II

EXAMPLE OF PROCESS QUALIFICATION

The following special (i.e. safety relevant) processes — those that cannot be verified later in the fabrication sequence — were qualified for use during the advanced CANDU reactor mixed oxide (MOX) fuel fabrication campaign:

— Low enriched uranium powder blending;
— MOX powder blending;
— Sintering of MOX fuel pellets;
— Welding of fuel elements;
— Welding of MOX fuel bundles.

The procedure for these qualifications is illustrated by the examples of the sinter process for MOX fuel pellets and the MOX powder blending process. Note that the main aim of these qualification processes is to demonstrate that the corresponding specifications are met.

The densification of MOX fuel pellets depends on the operating parameters of the sintering process as well as the initial density (green density) of the pellets, which in turn depends on the operating parameters of the final press. Appropriate parameters for the final pressing and sintering processes of MOX fuel pellets are determined prior to the qualification process. Using these parameter values and based on the results from the scoping tests, a qualification test is performed. For this qualification test, 10 sintered pellets per tray (there are 80 pellets in total on a tray) are selected diagonally across the tray for inspection by immersion density. Figure II–1 depicts a typical density distribution for the sintered pellets along the diagonal sample for each sintering furnace tray.

Merging the results obtained from the ten trays yields an average sintered density of 10.45 g/cm^3 with a standard deviation of 0.06 g/cm^3 meeting the density specification (average density ≥ 10.45 g/cm^3).

Proceeding in this way does not meet the requirement to consider the uncertainty generated by randomness appropriately because the two values 10.45 g/cm^3 and 0.06 g/cm^3 are the result of a random process, and any repetition of the qualification process could result in quite different values. Considering uncertainty means taking the observed values (10.45 g/cm^3, 0.06 g/cm^3) and determining all those values of the expectation μ and the standard deviation σ that are consistent with them. In the above example with a sample size n = 80 (10 samples of size 8) and a confidence level of β = 0.95, the following sets of consistent values would appear:

(a) $10.43 \leq \mu \leq 10.47$,
(b) $0.05 \leq \sigma \leq 0.07$.

These values can be used to develop a set of probability distributions that cover the true but unknown probability distribution and hence the entire existing uncertainty. Such a model reveals possible weak points of a process and may therefore be used not only for the purpose of qualification but also to devise a quality improvement plan. Figure II–1 reveals another issue that should be carefully considered. The differences of the probability distributions for the different trays on the one hand and the localizations of the samples on the other indicate possible differences in the underlying probability distributions that might again be a hint for possible improvements.

The grain size distribution of the sintered pellets is checked by ceramography, while autoradiography is used to determine plutonium rich areas in the MOX powder blending process. Standard ceramography and alpha autoradiography analyses are performed for every six batches of sintered pellets to determine the plutonium rich area and grain size distribution. Figure II–2 depicts a typical ceramographic image of an etched MOX pellet.

The average grain size was determined to be 8 μm and the standard deviation 0.5 μm in this sample. As with the case of the density, the observed

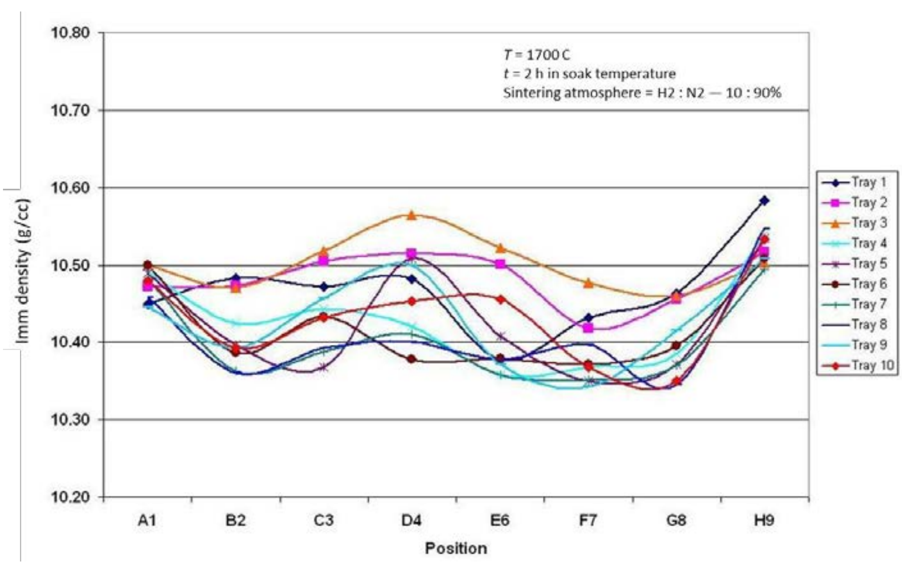


FIG. II–1. Sintered densities of the qualification batch.

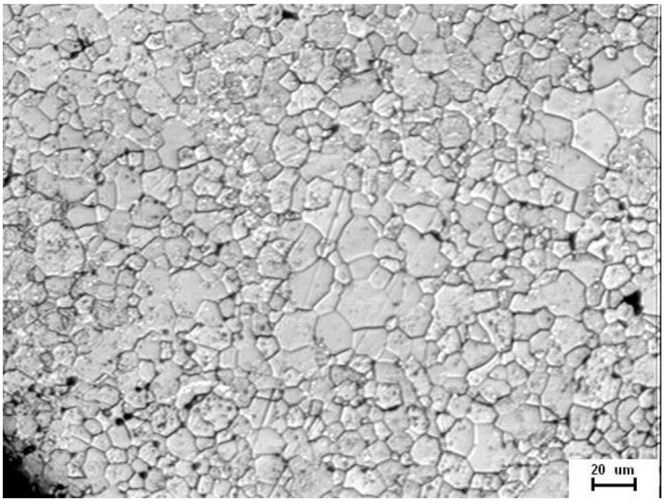


FIG. II–2. Grain size distribution of an etched MOX pellet.

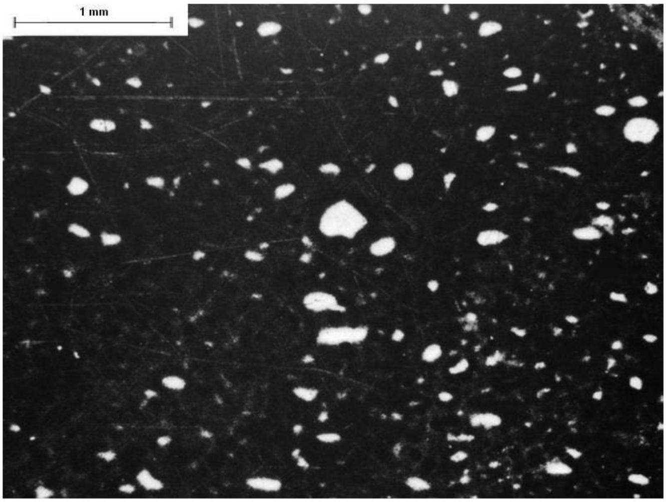


FIG. II–3. Alpha autoradiography for a MOX pellet.

empirical values should be used to develop a stochastic model that covers the entire existing uncertainty.

Figure II–3 shows a typical autoradiograph of a MOX pellet. From alpha autoradiography, it was concluded that a uniform distribution of plutonium rich areas was normally produced in the MOX powder blending process.

Annex III

CERTIFIED REFERENCE MATERIALS INFORMATION

Tables III–1 and III–2 provide information about certified reference nuclear materials and about their producers and suppliers, respectively.

TABLE III–1. CERTIFIED REFERENCE MATERIALS AVAILABLE FOR NUCLEAR MATERIALS

Reference material	Properties and relative uncertainties	Measurement methods
U metal	U content (0.005%) Isotopic composition (0.1–0.001%)	Titration IDMS[a] TIMS[b]
UF_6	U content (0.05%) Isotopic composition (0.1–0.001%)	GSMS[c] Titration IDMS
U_3O_8	U content (0.01%) Isotopic composition (0.1–0.001%)	Titration TIMS IDMS
UO_2 fuel pellet	U content (0.05%) U-235	Titration TIMS
U enrichment	Range from < 0.02% U-235 to > 99% U-235	TIMS MC-ICP[d]
Pu metal	Pu content (0.03%) Isotopic abundance (0.001%)	IDMS TIMS Coulometry
Pu oxides, nitrates	Pu content and/or isotopic composition	Various
MOX[e] pellets	U, Pu content (0.01%)	Various
ThO_2	Th content (uncertainty 0.01%)	Gravimetry ICP-AES[f]
$ThO_2 + UO_2$	UO_2 content 2.5% (uncertainty 0.02%)	Gravimetry Volumetry ICP-AES

[a] IDMS: isotope dilution mass spectrometry.
[b] TIMS: thermal ionization mass spectrometry.
[c] GSMS: gas source mass spectrometry.
[d] MC-ICP: multi-collector inductively coupled plasma.
[e] MOX: mixed oxide.
[f] ICP-AES: inductively coupled plasma atomic emission spectrometry.

TABLE III–2. EXAMPLES OF SOME CERTIFIED REFERENCE MATERIAL PRODUCERS AND SUPPLIERS

Certified reference material producer	Web site
Analytical Methods Committee (CETAMA, Commission d’etablissement des méthodes d’analyse)	https://cetama.partenaires.cea.fr/
Joint Research Centre of the European Commission (formerly Institute for Reference Materials and Measurements)	https://joint-research-centre.ec.europa.eu/
NBL Program Office	www.energy.gov/nnsa/nbl-program-office
Nuclear Fuel Complex (NFC)	http://www.nfc.gov.in

Annex IV

PRODUCT CONTROL FOR PRESSURIZED WATER REACTOR, BOILING WATER REACTOR AND WATER WATER ENERGY REACTOR FUELS

This annex deals with quality control for the components of pressurized water reactor (PWR), boiling water reactor (BWR) and water water energy reactor (WWER) fuel. The focus is on product control and deals with quality characteristics, with the type of inspections, and with taking and analysing samples.

IV-1. ZIRCONIUM ALLOY PRODUCTS

Product and process control methods need to be applied for zirconium alloys to cover sensitive material characteristics and sensitive process parameters. Specific expertise is required because of the properties of zirconium alloys, for example their anisotropy or high chemical affinity to oxygen and other gases.

IV-1.1. Sponge

The sensitive chemical elements in zirconium sponge analysis are given in Table IV–1, in accordance with the ASTM B-349 standard [IV–1].

In addition to chemical analysis, visual inspection is performed to eliminate individual grains of sponge presenting a discolouration due to oxide or nitride zirconium.

The process for separating zirconium and hafnium by extractive distillation developed and used by AREVA represents a best practice for a more environmentally friendly chemical separation process compared to existing processes, and a guarantee of very low hafnium content in the final zirconium based product [IV–2].

IV-1.2. Ingot

Three points are of major interest for the characterization of the final ingot:

(a) The chemical composition and homogeneity (see Table IV–2);
(b) Workability, which means deformability during the subsequent reduction steps;
(c) Sampling is monitored to have the cross-section and the top middle bottom positions of the ingot.

TABLE IV–1. SENSITIVE CHEMICAL ELEMENTS IN ZIRCONIUM SPONGE ANALYSIS

Chemical element	ASTM limit (ppm)	Relevance of preceding process steps	Relevant points for further use
Cl	1300	Chemical separation and reduction	Melting
Hf	100	Chemical separation	Neutron economy
Mg	n.a.[a]	Reduction	Ingot specimen
N	50	Reduction	Scrap recycle
O	1400	Reduction	Scrap recycle

[a] n.a: not applicable.

TABLE IV–2. SENSITIVE CHEMICAL ELEMENTS IN ZIRCONIUM ALLOY INGOT ANALYSIS

<table>
<tr><th>Chemical element</th><th>ASTM limit (ppm)</th><th>Relevance of proceeding process steps</th><th>Relevant points for further use</th></tr>
<tr><td colspan="4">Alloying elements</td></tr>
<tr><td>Sn</td><td rowspan="5">According to Zircaloy specification</td><td rowspan="4">Melting and scrap recycling</td><td rowspan="4">Mechanical and corrosion properties</td></tr>
<tr><td>Fe</td></tr>
<tr><td>Cr</td></tr>
<tr><td>Ni</td></tr>
<tr><td>O</td><td>Scrap recycling</td><td>Mechanical properties</td></tr>
<tr><td colspan="4">Impurities</td></tr>
<tr><td>N</td><td>≤ 80</td><td rowspan="2">Melting and scrap recycling</td><td rowspan="2">Gas uptake during subsequent process steps Corrosion</td></tr>
<tr><td>H</td><td>≤ 25</td></tr>
<tr><td>C</td><td>≤ 270</td><td>Scrap recycling</td><td>Mechanical properties</td></tr>
</table>

Data from Table B3/IV in Ref. [IV–3], but with peak nitrogen concentration corrected according to ASTM B350 in Ref. [IV–4], published in 2011.

The element analysis is performed mainly by plasma emission spectrometer, and great care is to be taken to use the right reference materials because of structural impact.

IV-1.3. Semi-final product

The main non-destructive test techniques used to control the thick walled seamless zirconium alloy tubes are given in Table IV–3.

TABLE IV–3. NON-DESTRUCTIVE TESTING TECHNIQUES FOR THICK WALLED SEAMLESS ZIRCONIUM ALLOY TUBES (HOLLOWS AND TREXes)

Test target		Test technique	Calibration	Accuracy or resolution
Overall	Specific			
Dimensions	OD[a]/ID[b]/wall thickness	Ultrasonic transit time method	Standard test tube	±50 µm
	Straightness	Comparison with straightness standard	n.a.[c]	0.1 mm
Defects	Inclusions Cracks Shrink holes Imperfection	Ultrasonic pulse echo method in four directions	Standard defects, V and U shaped notches at ID/OD, transverse and longitudinal	Cracks and notches ≥100 µm
Surface	Imperfections Contaminations Roughness	Visual inspection Roughness measurement device Microscopic	Standards Standards samples	5–10× magnification 0.1 µm

[a] OD: outer diameter.
[b] ID: inner diameter.
[c] n.a.: not applicable.

IV-1.4. Cladding and thimble tubes

Non-destructive and destructive testing are used to control the final tube, flat or bar products, as presented in Table IV–4.

Destructive testing techniques are mainly used for mechanical properties, chemical composition, corrosion properties, microstructure and texture

TABLE IV–4. NON-DESTRUCTIVE TESTING TECHNIQUES FOR THIN WALLED SEAMLESS ZIRCONIUM ALLOY TUBES FOR CLADDING

Test target		Test technique	Calibration	Accuracy or resolution
Overall	Specific			
Dimensions	OD[a]/ID[b]/wall thickness	Ultrasonic transit time method	Standard test tube	±3 µm
	Straightness	Comparison with straightness standards	n.a.[c]	0.01 mm
Defects	Inclusions Cracks Shrink holes	Ultrasonic pulse echo method in four directions	Standard defects, V and U shaped notches at ID/OD, transverse and longitudinal	Rejection level according to specification
	Imperfection	Eddy current inspection	Standard defects according to specification	Rejection level according to specification
Surface	Imperfections Contaminations	Automatic optical inspection	Standards	n.a.
	Roughness	Roughness measurement device	Standards samples	0.1 µm

[a] OD: outer diameter.
[b] ID: inner diameter.
[c] n.a.: not applicable.

measurement. Standards ASTM E8 [IV–5] and ASTM E21 [IV–6] specify the requirements to apply to a tensile test.

Room or elevated temperature tensile tests are performed to measure yield strength, ultimate yield strength and uniform or fracture elongation. As far as qualification tests are concerned, creep tests or burst tests can be requested for characterization of the multiaxial properties of zirconium alloy tubing. For flat products, bending tests and or dimple tests can be performed.

If metallographic techniques are used to check grain size on a manufacturing lot, texture measurement by X ray diffraction methods can be performed for the purpose of product qualification.

Corrosion measures are performed in steam in accordance with the ASTM G2 standard [IV–7] for uniform corrosion resistance acceptance for the lot reception. The corrosion rates should be less than 22 mg/dm^2 after 3 days or 38 mg/dm^2 after 14 days.

Tests with samples exposed to pressurized water can be applied to test welding or mechanical conditioning. The in-reactor chemical conditions can be simulated in such tests in order to characterize the material performance.

IV-1.5. Bars

Bars are used mainly to machine cladding end plugs and need to be controlled dimensionally but also by ultrasonic test and visual inspections, such as those presented in Table IV–5.

TABLE IV–5. NON-DESTRUCTIVE TESTING TECHNIQUES FOR ZIRCONIUM ALLOY BARS

Test target		Test technique	Calibration	Accuracy or resolution
Overall	Specific			
Dimensions	Diameter	Gauges at middle and ends	Standard test tube	±3 µm
	Straightness	Gauges check space to straightness standard	n.a.	0.01 mm

TABLE IV–5. NON-DESTRUCTIVE TESTING TECHNIQUES FOR ZIRCONIUM ALLOY BARS (cont.)

Test target		Test technique	Calibration	Accuracy or resolution
Overall	Specific			
Defects	Imperfections	Ultrasonic immersion pulse echo technique	Standard defects, radial and axial	≤0.8 mm according to specification
	Inclusions Shrink holes	Metallography cross-sections at ends by sampling	Holes Microscopic examination	10 µm
Surface	Imperfections Contaminations	Visual inspection	Standards	5–10× magnification
	Roughness	Visual inspection	Standards	5–10× magnification

IV-1.6. Flat products

Flat products are used for strips for grids or sheets for channel boxes. Tests on the mechanical properties in the longitudinal and transverse directions and for microstructure and corrosion are performed for each manufacturing lot by sampling. The relevant non-destructive test methods are presented in Table IV–6.

TABLE IV–6. NON-DESTRUCTIVE TESTING TECHNIQUES FOR ZIRCONIUM ALLOY BARS

Test target		Test technique	Calibration	Accuracy or resolution
Overall	Specific			
Dimensions	Thickness	Electromagnetic probe micrometer	Calibration gauges Standard samples	±0.004 mm
	Length	Ruler measurement	Calibration gauges Standard samples	±0.05 mm
	Width	Calliper with gauges	Calibration gauges Standard samples	±0.01 mm for channels ±0.1 mm for spacer strips
	Flatness	Gauges	Calibration gauges Standard samples	±0.01 mm to 0.25 mm, depending on RA[a]
Defects in slabs	Imperfections Inclusions Shrink holes	Ultrasonic immersion pulse echo technique	Standard defects: flat bottom holes in standard block	n.a.[b]
Surface	Imperfections Contaminations	Optical/visual		5–10× magnification
	Roughness	Roughness measurement device	Standard samples	±0.08 mm

[a] RA: rod assembly.
[b] n.a.: not applicable.

IV-2. QUALITY PLAN TO MEET THE REQUIREMENTS AND EXPECTATIONS OF DESIGNERS AND CUSTOMERS

The product delivered at the exit of this process should have the features required by the designer in the drawings or specifications. Table IV–7 provides a synthetic view of the nature of the product quality requirements at different key points of the manufacturing sequence.

TABLE IV–7. PRODUCT QUALITY REQUIREMENTS AT KEY POINTS OF THE MANUFACTURING SEQUENCE

Product	Main quality points to be managed
Powder	Powder materials features
Pellet	Pellet materials final features Pellet dimension: diameter Visual aspect Cleanliness
Fuel rod	Welding process quality Fuel rod content (pellet column, spacer, rod spring) Visual aspect Incoming components and rod cleanliness
Spacers, end parts and skeleton	Part materials features Process quality: welding, brazing, conventional machining, electropolishing, electrical discharge machining, sand blasting and deburring Cleaning process quality Materials special heat treatment Part final dimensions Part cleanliness
Assembly	Assembly manufacturing operations: rod loading and end part installation Cleaning operation (blowing or washing) Assembly handling conditions Assembly dimensional features Assembly cleanliness Assembly shipping conditions: installation in container and container handling

Three types of actions contribute to the achievement and confirmation of final product conformity:

(a) Action 1 — preparatory engineering actions to control the manufacturing process:
   (i) The product is elaborated through successive steps, where it is transformed (i.e. conversion, pressing, sintering), assembled (i.e. pellet loading, skeleton welding), stored or handled.
   (ii) At each of these steps, the equipment and the operating conditions need to be prepared attentively and defined by the equipment engineer and the process engineer to be sure that the step contributes efficiently to the expected final quality of the product.
   (iii) The manufacturing phase needs to be prepared and it is necessary to identify some ranges in which the important operating conditions are to be contained to provide stable and repetitive manufacturing situations.
(b) Action 2 — manufacturing samples collected and testing of samples:
   (i) During manufacturing, at appropriate steps, representative samples need to be created, and destructive tests on these samples need to prove that the process is under control and generates correct products repetitively.
(c) Action 3 — direct examinations of the actual manufactured product on-line:
   (i) During manufacturing, direct non-destructive inspections are performed on the product itself to check conformity to requirements concerning dimensions, visual aspects and cleanliness.

REFERENCES TO ANNEX IV

[IV–1] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Specification for Zirconium Sponge and Other Forms of Virgin Metal for Nuclear Application, ASTM B349/B349M, ASTM, West Conshohocken, PA (2009).

[IV–2] BAUR, K., VON COLLANI, E., Nuclear Fuel Quality Management Handbook, Vol. I: Fabrication, Operation, Disposal and Transport of Nuclear Fuel, ANT International, Skultuna (2010).

[IV–3] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on Quality Control of Water Reactor Fuel, Technical Report Series No. 221, IAEA, Vienna (1983).

[IV–4] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Specification for Zirconium and Zirconium Alloy Ingots for Nuclear Application, ASTM B350, ASTM, West Conshohocken, PA (2011).

[IV–5] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Test Methods for Tension of Metallic Materials, ASTM E8/E8M, ASTM, West Conshohocken, PA (2009).

[IV–6] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Test Methods for Elevated Temperature Tension Tests of Metallic Materials, ASTM E21, ASTM, West Conshohocken, PA (2009).

[IV–7] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Test Method for Corrosion Testing of Products of Zirconium, Hafnium, and Their Alloys in Water at 680°F (360°C) or in Steam at 750°F (400°C), ASTM G2/G2M, ASTM, West Conshohocken, PA (2011).

ABBREVIATIONS

ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASNT	American Society for Nondestructive Testing
ASTM	American Society for Testing and Materials
ATF	accident tolerant fuel or advanced technology fuel
BOA	boron-induced offset anomaly risk assessment tool
BWR	boiling water reactor
CANDU	Canadian deuterium–uranium (Canadian PHWR)
CMA	Chemistry Monitoring and Assessment
CORAL	Crud Deposition Risk Assessment Model channel
DBA	design basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
EPRI	Electric Power Research Institute
FHD	hold down force
FHY	hydraulic force
FRED	Fuel Reliability Database
FSAR	final safety analysis report
GSR	General Safety Requirement
GTRF	grid to rod fretting
ICFM	in-core fuel management
IEC	International Electrochemical Commission
INPO	Institute of Nuclear Power Operations
ISO	International Organization for Standardization
LHR	linear heat rating
LOCA	loss of coolant accident
LWR	light water reactor
MOX	mixed oxide
NEA	Nuclear Energy Agency of the OECD
NFIF	nuclear steam supply system–fuel interface file
NRC	Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
PCI	pellet–cladding interaction
PCMI	pellet–cladding mechanical interaction
PCT	peak cladding temperature
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor

QA	quality assurance
QC	quality control
QMS	quality management system
RBMK	high power channel type reactor (Russian graphite moderated LWR)
RCCA	rod control cluster assembly
RIA	reactivity initiated accident
SALDNBR	safety analysis limit DNBR
SAR	safety analysis report
SCC	stress corrosion cracking
SDLDNBR	statistical design limit DNBR
SMR	small modular reactor
SS	stainless steel
SVVP	software verification and validation plan
37M	modified 37-element fuel bundle (for CANDU reactors)
37R	reference 37-element fuel bundle (for CANDU reactors)
V&V	verification and validation (of computer codes)
VERA	Virtual Environment for Reactor Applications
WANO	World Association of Nuclear Operators
WWER	water water energetic reactor (or water cooled, water moderated energy reactor) (Russian PWR); also called VVER
ZIRLO	Optimized high performance fuel cladding material (Westinghouse)

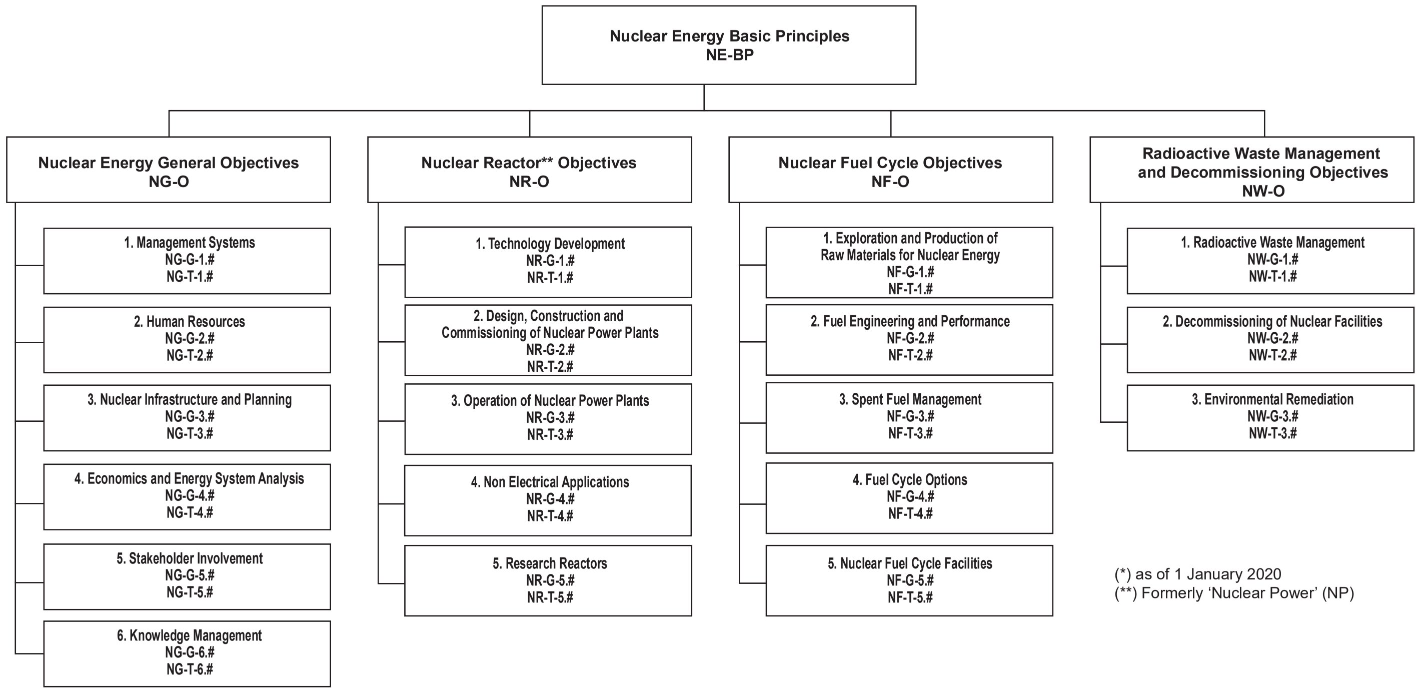
CONTRIBUTORS TO DRAFTING AND REVIEW

Alonso Pacheco, J.	ENUSA, Spain
Armstrong, J.	Canada Nuclear Laboratories, Canada
Aullo Chaves, M.	ENUSA, Spain
Gorokhov, I.	OKB GIDROPRESS, Russian Federation
Gusev, A.	TVEL, Russian Federation
Inagaki, K.	Central Research Institute of Electric Power Industry (CRIEPI), Japan
Judah, J.	J. Judah and Associates, Inc., Canada
Mader, E.	Electric Power Research Institute (EPRI), United States of America
Sim, K.	International Atomic Energy Agency
Waeckel, N.	Électricité de France (EDF), consultant France
Zhang, J.	TRACTEBEL (ENGIE), Belgium

Consultancy meetings

Virtual: 29–30 March 2021, 6–8 December 2021
Vienna, Austria: 7–10 June 2022

Structure of the IAEA Nuclear Energy Series*



(*) as of 1 January 2020
(**) Formerly 'Nuclear Power' (NP)

Key

BP: Basic Principles
O: Objectives
G: Guides and Methodologies
T: Technical Reports
Nos 1–6: Topic designations
#: Guide or Report number

Examples

NG-G-3.1: Nuclear Energy General (**NG**), Guides and Methodologies (**G**), Nuclear Infrastructure and Planning (topic **3**), **#1**
NR-T-5.4: Nuclear Reactors (**NR**), Technical Report (**T**), Research Reactors (topic **5**), **#4**
NF-T-3.6: Nuclear Fuel (**NF**), Technical Report (**T**), Spent Fuel Management (topic **3**), **#6**
NW-G-1.1: Radioactive Waste Management and Decommissioning (**NW**), Guides and Methodologies (**G**), Radioactive Waste Management (topic **1**) **#1**

IAEA
International Atomic Energy Agency

No. 26

ORDERING LOCALLY

IAEA priced publications may be purchased from the sources listed below or from major local booksellers.

Orders for unpriced publications should be made directly to the IAEA. The contact details are given at the end of this list.

NORTH AMERICA

Bernan / Rowman & Littlefield
15250 NBN Way, Blue Ridge Summit, PA 17214, USA
Telephone: +1 800 462 6420 • Fax: +1 800 338 4550
Email: orders@rowman.com • Web site: www.rowman.com/bernan

REST OF WORLD

Please contact your preferred local supplier, or our lead distributor:

Eurospan Group
Gray's Inn House
127 Clerkenwell Road
London EC1R 5DB
United Kingdom

Trade orders and enquiries:
Telephone: +44 (0)176 760 4972 • Fax: +44 (0)176 760 1640
Email: eurospan@turpin-distribution.com

Individual orders:
www.eurospanbookstore.com/iaea

For further information:
Telephone: +44 (0)207 240 0856 • Fax: +44 (0)207 379 0609
Email: info@eurospangroup.com • Web site: www.eurospangroup.com

Orders for both priced and unpriced publications may be addressed directly to:

Marketing and Sales Unit
International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22530 • Fax: +43 1 26007 22529
Email: sales.publications@iaea.org • Web site: www.iaea.org/publications

23-05072E-T

Improved fuel reliability means reduced fuel failures in reactor operation. Fuel failures, with their consequent adverse impact on the environment and requirements for additional waste management, result in costs for remediation, 'failed core' operation and maintenance. Therefore, poor performance of fuel can lead to uncompetitive operational conditions for a nuclear power plant. A revision of the earlier edition, this publication has been significantly extended to support nuclear fuel designers, manufacturers, reactor operators, and fuel engineers and managers on fuel design and design changes, fuel manufacturing, qualification, in-reactor operation, and on-site services to achieve excellence in fuel reliability and performance and safe operation of nuclear fuel under all applicable plant states.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA