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Technical Review of Acceptance Criteria for Pressurized Heavy Water Reactor Fuel

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TECHNICAL REVIEW OF ACCEPTANCE CRITERIA FOR PRESSURIZED HEAVY WATER REACTOR FUEL

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INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

Historically, pressurized heavy water reactor (PHWR) fuel bundles have been designed on the basis of descriptive requirements to ensure that the fuel bundles remain compatible with interfacing systems (e.g. primary heat transport system, fuel channel assembly, fuel handling system) and that they remain intact for all operational states of the reactor. Consideration is also given to ensuring that sufficient safety margins are embedded in the design for accident conditions. The qualification of the fuel bundle design for use in PHWRs has been demonstrated in a conservative manner where bounding or limiting conditions are taken into account.

As part of a general trend in fuel licensing, this historical approach to PHWR fuel design and safety analysis is gradually being altered to demonstrate quantified margins in the design. There is an increasing preference for the use of fuel acceptance criteria with numerical values and validated fuel computer codes in the demonstration of design safety margins.

The light water reactor (LWR) fuel acceptance criteria have been reviewed and updated several times since the late 1990s, to reflect current knowledge of fuel performance and the influence of design improvements to pellets, cladding materials and fuel assembly structures. In contrast, a systematic review of the PHWR fuel acceptance criteria used in IAEA Member States has not been performed, although States with operating PHWRs have invested significant effort into developing fuel acceptance criteria and methods that allow the use of fuel computer codes to assess design safety margins.

In response to a request by Member States with PHWRs for a review of progress in developing PHWR fuel acceptance criteria and in addressing associated issues, the IAEA organized a Technical Meeting on Pressurized Heavy Water Reactor Fuel Safety: Strategy and Path Forward, held 8–12 June 2015 in Ottawa, Canada.

The current publication describes the PHWR fuel acceptance criteria with numerical values reported by States with PHWRs and the technical basis of these criteria. The information provided is mainly based on the results of the technical meeting and Member State responses to a questionnaire on the use of the PHWR fuel acceptance criteria; information since the technical meeting in 2015 has also been included, as appropriate.

The IAEA is grateful to the meeting participants for their active involvement and presentations and to the subject matter experts who took part in the preparation of this publication for their valuable contributions and reviews. Special acknowledgement is given to M. Couture (Canada) for his leadership in compiling this publication.

The IAEA officers responsible for this publication were K. Sim of the Division of Nuclear Fuel Cycle and Waste Technology and F. Parada Iturria of the Division of Nuclear Installation Safety.

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1. INTRODUCTION

1.1. BACKGROUND

The light water reactor (LWR) fuel acceptance criteria used in States have been reviewed several times mainly by the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) Fuel Safety Working Group since the late 1990s, to reflect up-to-date knowledge on fuel performance and the influence of improved design of pellets, cladding materials and fuel assembly structures. Unlike LWR fuel, a systematic review of the pressurized heavy water reactor (PHWR) fuel acceptance criteria used in Member States of OECD/NEA or IAEA, respectively, has not been performed, though States with operating PHWRs have invested significant efforts into developing such criteria and methods that allow the use of computer codes in assessing the margins. Some of these States made significant progress in this area in terms of regulatory requirements and industrial efforts, while others have not been as progressive in their approach.

A brainstorming-type consultancy meeting was held at the IAEA from 2-4 June 2014, in response to the request of Member States with PHWRs to review the progress in developing fuel acceptance criteria and in addressing associated safety issues. The PHWR States have emphasized the need to share the fuel acceptance criteria developed among each other, in order to ensure the safe operation of all PHWRs in the world. With the support of the Canadian nuclear regulatory body (Canadian Nuclear Safety Commission (CNSC)), the nuclear industry and other representatives from PHWR States, the IAEA was able to provide a platform where all PHWR States could work together to perform a systematic review of PHWR fuel acceptance criteria. Subsequently, an IAEA Technical Meeting on PHWR Fuel Safety: Strategy and Path Forward was held in Ottawa, Canada, from 8-12 June 2015. Prior to this Technical Meeting, a survey on the use of fuel acceptance criteria in all PHWR States was conducted using a questionnaire. The purpose of the Technical Meeting was to review and assess the fuel acceptance criteria used in the PHWR community and collect input for this publication.

Although the topic covered in this publication is the same as in the OECD/NEA 2012 publication entitled “Nuclear Fuel Safety Criteria Technical Review” [1], the approach taken is quite different. Being a first attempt, it provides more background information regarding the role PHWR fuel acceptance criteria play in nuclear safety and details regarding the technical basis of the reported fuel acceptance criteria.

In the IAEA Safety Glossary [2], the term ‘acceptance criteria’ is defined as “specified bounds on the value of a functional indicator or condition indicator used to assess the ability of a structure, system or component to perform its design function”. The acceptance criteria[1] can be expressed in qualitative terms or as quantitative limits, and there are three categories of such criteria, as specified in IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [3]:

- Safety criteria: is defined as “[c]riteria that relate either directly to the radiological consequences of operational states or accident conditions, or to the integrity of barriers

[1] Acceptance criteria are established at two levels: high level (radiological) criteria and detailed (derived) technical criteria [3]. The radiological criteria are not covered by this publication.

against releases of radioactive material, with due consideration given to maintaining the safety functions”;

- Design criteria: is defined as “[d]esign limits for individual structures, systems and components, which are part of the design basis as important preconditions for meeting the safety criteria”;

- Operational criteria: is defined as “[r]ules 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”.

In IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants [4], ‘qualitative’ design limits (hereinafter design limits are referred to as ‘design criteria’) on relevant physical parameters for individual structures, systems and components of the reactor core for all operational states and accident conditions are described in accordance with Requirement 15 of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [5]. ‘Quantitative’ design criteria are used to provide a basis for the design of the fuel system for nuclear power plants. Generally, these quantitative design criteria need to be consistent with the qualitative design criteria described in SSG-52 [4]. Some operational criteria are also used for the same purpose as the quantitative design criteria.

The focus of this publication is given to acceptance (design) criteria for PHWR[2] fuel.

1.2. OBJECTIVE

This publication is intended to:

- Provide a collection of current practices in the fuel acceptance criteria used in States with operating PHWRs and technical justification available for such criteria;

- Identify topics where collaboration and further discussion between participating States would be beneficial.

1.3. SCOPE

This publication describes mainly the PHWR fuel acceptance criteria that are currently used, and it focuses primarily on horizontal channel-type operating PHWRs (e.g. CANDUs[3] and Indian PHWRs[4]) with current fuel bundles. Fuel acceptance criteria specific to Argentina’s

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[2] In this publication, operating PHWRs are grouped into two ‘horizontal channel-type PHWRs’ and ‘vertical channel-type PHWRs’. 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.

[3] 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.

[4] 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. The various discussions presented in Sections 3.1, 3.2 and Appendix I are of a general nature and therefore also apply, to a large extent, to Indian PHWRs. In cases where the statements are specific to CANDUs, the term ‘CANDUs’ is used instead of ‘horizontal channel-type PHWRs’.

‘vertical channel-type’ operating PHWRs (containing one single fuel assembly per fuel channel), Atucha-1 and Atucha-2, are also described as appropriate[5].

The fuel acceptance criteria reported in this publication are limited to those associated to fuel degradation mechanisms that can be activated in the reactor core. Degradation mechanisms of the fuel channel components that interface with the fuel bundles, and that could be activated by in-core fuel behaviour under operational states and accident conditions, are also addressed.

Regarding current practices in the fuel acceptance criteria, the following major sources of information have been used in this publication: responses to a questionnaire that was distributed by the IAEA to Member States with PHWRs; presentations at a Technical Meeting organized by the IAEA in Ottawa, Canada, from 8-12 June 2015; open literature documents related to PHWR fuel design; and additional information provided upon request by individual PHWR States.

It is recognized that, probably due to proprietary reasons, some responses to the questionnaire were brief or even missing. This publication compiles all responses submitted by the participating States.

Some main variants of PHWRs are currently in operation worldwide: CANDU 6, Pickering, Bruce/Darlington, India’s PHWRs, and Argentina’s Atucha-1 and Atucha-2 reactors. Under these main variants, various fuel design options are currently in use, under active development or under serious consideration. Examples include the following:

(1) Horizontal channel-type PHWRs:

- 19-natural uranium fuel element bundle (Indian PHWRs);
- 19-natural uranium fuel element bundle (KANUPP, Pakistan);
- 28-natural uranium fuel element bundle (Pickering, Canada);
- 37-natural uranium fuel element bundle for
  - CANDU 6 nuclear power plants (Point Lepreau (Canada), Wolsong (Korea), Qinshan (China), Cernavoda (Romania) and Embalse (Argentina)),
  - Bruce and Darlington nuclear power plants in Canada,
  - Indian 540 MWe PHWRs,
  - Modified 37-natural uranium fuel element element fuel bundle (Bruce and Darlington nuclear power plants in Canada);
- 43-natural uranium fuel element bundle (Korea, Canada);
- Thorium fuel pellets for use in 19-fuel element bundle (Indian PHWRs);
- Mixed-oxide fuel pellets for use in 19-fuel element bundle (Indian PHWRs).

(2) Vertical channel-type PHWRs:

[5] The fuel acceptance criteria for Argentina’s vertical channel-type operating PHWRs may not be comprehensively described due to limited information. Available source information is limited to questionnaire responses and presentations by Argentine representatives at the Technical Meeting in Ottawa.

- Slightly enriched uranium fuel pellets and natural uranium fuel pellets for use in 37-fuel element bundles (see footnote 6 for Argentina's Atucha-1 and Atucha-2 reactors).

Fuel acceptance criteria described in this publication are intended to cover fuel design options with uranium fuel[7] for use in PHWRs unless otherwise specified.

1.4. STRUCTURE

After briefly describing some general safety considerations in the design of horizontal channel-type PHWR core and pointing out the role that the specification of fuel acceptance criteria plays in their successful implementation, Section 2 discusses the design approach to meet safety objectives, key characteristics and regulatory requirements and practices regarding the horizontal channel-type PHWRs fuel acceptance criteria reported in this publication.

Section 3 describes the various fuel acceptance criteria used for currently operating reactors in States with PHWRs. Section 3.1 covers the fuel acceptance criteria used for operational states, while the fuel acceptance criteria used for accident conditions are reported in Section 3.2.

Section 4 describes conclusions and recommendations for future work.

Nuclear design criteria were among the topics covered in the IAEA questionnaire, and in several of the presentations made during the IAEA Technical Meeting in June 2015. Although not fuel acceptance criteria as such, they constitute key inputs in the verification that fuel acceptance criteria reported in Sections 3.1 and 3.2 are met. Criteria reported by participating PHWR States are provided in Appendix I.

Appendix II discusses the fuel and pressure tube fitness-for-service criteria for a specific fuel type (i.e. 37-natural uranium fuel element bundle) which the Canadian utilities propose to use in order to support the restart of their reactors after shutdown following certain slow design basis events (i.e. anticipated operational occurrences); this provides a good example of the operational support role of the fuel acceptance criteria reported in Section 3.1.

Appendix III identifies, for horizontal channel-type PHWRs, physical barriers and the safety functions they perform.

Annex I describes some key features of a criteria-based verification (CBV) approach to fuel design qualification currently being developed by the Canadian nuclear industry [6].

The Canadian approach to categorization of plant states for CANDUs, including design extension conditions (DECs), is presented in Annex II. The Indian approach to categorization of plant states for water cooled reactors, including DECs, is presented in Annex III.

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[6] Atucha-1 and Atucha-2 fuel elements have different design features from those of CANDU 37-element bundle, such as free-standing concept between pellets and the sheath, thick wall thickness, initial filling gas pressure, element length (approximately 6 m) and vertical position in the reactor.

[7] In principle, fuel acceptance criteria applicable to current natural uranium fuel are, with justification, applicable to advanced fuels for horizontal channel-type PHWRs. Numerical values of some criteria may be different from those for natural uranium fuel.

2. PHWR FUEL ACCEPTANCE CRITERIA: GENERAL CONSIDERATIONS

2.1. GENERAL SAFETY CONSIDERATIONS IN THE DESIGN OF THE REACTOR CORE

As stated in IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [7], the fundamental safety objective is "to protect people and the environment from harmful effects of ionizing radiation". To ensure the safety of nuclear power plants by avoiding the failure of barriers against the release of radioactive material, and by mitigating the consequences of their failure, the fulfilment of the three fundamental safety functions is required for all plant states, in accordance with Requirement 4 of SSR-2/1 (Rev. 1) [5]. The fundamental safety functions[8] as they apply specifically to the design of the reactor core[9] are as follows (Paragraph 2.2 of SSG-52 [4]):

- Control of reactivity;
- Removal of heat from the reactor core;
- Confinement of radioactive material.

The plant states typically considered for the design of the reactor core are normal operation, anticipated operational occurrences (AOOs), design basis accidents (DBAs) and design extension conditions (DECs) without significant fuel degradation (Paragraph 2.10 of SSG-52 [4]). The Canadian approach to the categorization of plants states for CANDUs, including DECs, is presented in Annex II. Table 1 presents typical examples of AOOs and DBAs considered for CANDUs. India also has an exhaustive classification of AOOs and DBAs considered for PHWRs, similar to the categorization given in Table 1.

Adequate design (i.e. capable, reliable and robust design) of the reactor core, including fuel bundles, based on the concept of defence in depth, will enable achievement of the fundamental safety functions, together with provision of associated reactor safety features (Paragraph 2.3 of SSG-52 [4]). This approach provides redundant means to ensure the fulfilment of the fundamental safety functions.

Accordingly, fuel elements are designed to ensure that their structural integrity and a leak tight barrier are maintained to prevent the transport of fission products into the coolant for normal operation and AOOs [4, 5]. Fuel elements are also designed to keep fuel cladding failures to a minimum for DBAs [4]. The reactor core is designed to maintain a configuration such that it can be shut down and remain coolable for design basis accidents and design extension conditions without significant fuel degradation [4, 5].

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[8] In Canada, they are often referred to as the 'Control, Cool and Contain' fundamental safety functions.

[9] The reactor core is the central part of a nuclear reactor where nuclear fission occurs. The reactor core consists fuel bundles, the coolant, the moderator, the reactor control systems, and fuel channels and related structures.

TABLE 1. EXAMPLES OF ANTICIPATED OPERATIONAL OCCURRENCES AND DESIGN BASIS ACCIDENTS CONSIDERED FOR CANDUs (CANADIAN PRACTICE, REPRODUCED COURTESY OF CNSC [8])

EVENT CATEGORY[a]	POSTULATED INITIATING EVENTS	
	ANTICIPATED OPERATIONAL OCCURRENCES	DESIGN BASIS ACCIDENTS
Increase in-reactor heat removal	— Inadvertent opening of steam relief valve — Secondary pressure control malfunctions leading to an increase in steam flow rate — Feedwater system malfunctions leading to an increase in the heat removal rate	— Steam line break
Decrease in-reactor heat removal	— Feedwater pump trips — Reduction in the steam flow rate for various reasons (e.g. control malfunctions, main steam valve closure, turbine trip, loss of external load, loss of power, loss of condenser vacuum)	— Feedwater line breaks
Change in-reactor coolant system flow rate	— Trip of one main coolant pump — Inadvertent isolation of one main coolant system loop (if applicable)	— Trip of more than one main coolant pump — Main coolant pump seizure or shaft break — Fuel channel flow blockage
Reactivity and power distribution anomalies	— Inadvertent single control rod withdrawal — Neutron poison concentration dilution due to a malfunction in the volume control system — Refuelling incorrect channel	— Uncontrolled control rod withdrawal
Increase in-reactor coolant inventory	— Malfunction of the chemical and inventory control system	— Inadvertent operation of emergency core cooling
Decrease in-reactor coolant inventory	— Very small loss of coolant accident (LOCA) due to the failure of an instrument line	— A spectrum of possible LOCAs — Inadvertent opening of the primary system relief valves — Leaks of primary coolant into the secondary system
Release of radioactive material from a subsystem or component	— Minor leakage from a radioactive waste system	— Overheating of, or damage to, used fuel in transit or storage — Break in a gaseous or liquid waste treatment system

[a] This Table provides a limited set of AOO and DBA case examples. More examples can be found in Appendix A of Ref. [8].

Paragraph 2.14 of SSR-2/1 (Rev. 1) [5] states:

"A relevant aspect of the implementation of defence in depth for a nuclear power plant is the provision in the design of a series of physical barriers, as well as a combination of active, passive and inherent safety features that contribute to the effectiveness of the physical barriers in confining radioactive material at specified locations."

Physical barriers considered as part of or affecting the design of the reactor core of a PHWR include: (i) the fuel matrix; (ii) the fuel sheath that prevents the release of radioactive material to the coolant; and (iii) the fuel channel that prevents the release of radioactive material to the containment.

The necessary steps in assuring that each of the physical barriers will remain intact with adequate margins are as follows:

- To identify, for operational states (normal operations and AOOs) and accident conditions, all the degradation mechanisms that could threaten the capacity of the physical barrier to perform its design function;
- To formulate, for each damage mechanism occurring under operational states, a design criterion which, if met, ensures that the mechanism will not prevent the barrier from remaining fit for service;
- To formulate, for each failure mechanism occurring under accident conditions, a design criterion which, if met, ensures that the mechanism is prevented from occurring.

The design criteria referred to in this publication as 'fuel acceptance criteria' have their foundation on experiments, operational evidence and analyses that identify the limitations of the material properties of the physical barriers when subjected to operational states and accident conditions.

2.2. DESIGN OBJECTIVES RELATED TO FUEL ACCEPTANCE CRITERIA AND KEY CHARACTERISTICS

In this publication, the physical barriers of interest are:

(a) The fuel matrix;

(b) The fuel sheath;

(c) The components of the primary heat transport system (PHTS) pressure boundary whose capacity to perform their design functions, under operational states and accident conditions, could be threatened by the fuel behaviour under those conditions.

The design criteria associated to those physical barriers are hereinafter referred to as 'fuel acceptance criteria'. The PHWR fuel acceptance criteria reported in this publication cover operational states and accident conditions, and reflect the criteria currently used for the PHWRs in operation worldwide.

Fuel acceptance criteria for operational states are criteria that need to be met in order to ensure that the fuel is capable of performing its design function without being damaged and without causing damage to the components of the interfacing systems that would render them no longer fit for service; they are reported in Section 3.1 and their design objectives and key characteristics are discussed in Section 2.2.1.

Fuel acceptance criteria for accident conditions are reported in Section 3.2 and their design objectives and key characteristics are discussed in Section 2.2.2.

2.2.1. Normal operations and anticipated operational occurrences

2.2.1.1. Design objectives

Damage to the fuel bundle is defined as a deviation of a design characteristic from its intended level with a severity that is sufficient to cause the fuel bundle design not to satisfy one or more of its design requirements. For each damage mechanism known to be active (which is also referred to as being a credible damage mechanism) under normal operations and AOOs, fuel acceptance criteria need to be specified with the objectives[10] of ensuring that [4, 9, 10]:

- The fuel is not damaged, where 'not damaged' means that:
  - (i) Fuel elements do not fail, where 'fuel element failure' means that the fuel element leaks and radioactive material is being released to the coolant;
  - (ii) Fuel (element/bundle) dimensions remain within operational tolerances;
  - (iii) The fuel bundle maintains its structural integrity;
  - (iv) The functional capabilities of the fuel are not reduced below those assumed in deterministic safety analysis.
- The damage that the fuel might cause to the fuel channel components is acceptable in the sense that these components remain fit for service[11].

In item (ii), the term 'operational tolerance' means the range of values of fuel element and bundle specific dimensions within which the fuel qualification is considered to be valid, in other words, within which the fuel bundle has been shown to meet its design and safety requirements. Examples of such dimensions are:

- Spacer and bearing pad wear (of fuel bundles for horizontal channel-type PHWRs);
- Fuel sheath strain;
- Fuel sheath corrosion thickness;
- Element bowing;
- End plate doming (of fuel bundles for horizontal channel-type PHWRs).

Maintenance of fuel bundle structural integrity does not preclude fuel sheath failures, but it does mean that the fuel bundle needs to have end plates free of cracks, that end cap-to-end plate welds remain intact, and that end plates are not permanently distorted (i.e. domed) by more than a prescribed limit.

Fuel is considered fit for service if it can continue to be operated in operational states with margins to damage.

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[10] The performance of the fundamental safety functions for normal operations and AOOs is considered successful, from a fuel design perspective, if those objectives are met.

[11] The term 'fitness-for-service' is used for CANDUs in Canada. For example, pressure tube fitness-for-service is determined according to rules set in CSA N285.4 [11] and N285.8 [12]. In other PHWR States, the term 'fitness-for-service' can be represented by other expressions.

Avoiding damage, as defined above, ensures that Requirements 43 and 44 of SSR-2/1 (Rev. 1) [5] regarding the maintenance, for operational states, of fuel structural integrity and coolable geometry, are met.

The specification of fuel acceptance criteria is among the defence in depth provisions taken so that (in accordance with Requirement 7 of SSR-2/1 (Rev. 1) [5]) the plant design ensures, as far as is practicable, that the first, or at most the second, level of defence is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the nuclear power plant.

2.2.1.2. Key characteristics

Each fuel damage mechanism, and each fuel channel component damage mechanism triggered by fuel behaviour, has one or more parameters related to relevant properties that uniquely describe (or directly govern) the damage mechanism process. Fuel acceptance criteria provide bounds (or limits) of these parameters which, if complied with, preclude with margin fuel (element/bundle) and fuel channel components being damaged. When material properties are not well characterized, it is a common engineering practice to formulate the fuel acceptance criteria in terms of surrogate parameters. Whether they are parameters that uniquely describe the damage mechanism or surrogates, these parameters are hereinafter referred to as the 'damage mechanism key parameters'.

Fuel acceptance criteria for operational states reported in Section 3.1 are categorized into three major groups [13]:

- Criteria to ensure the maintenance of fuel integrity under the degradation of thermal properties (hereinafter called 'thermal integrity of fuel');
- Criteria to ensure structural integrity of fuel;
- Criteria to ensure compatibility of fuel with interfacing systems.

They provide, for all credible damage mechanisms in operational states of currently operating PHWR, bounds (limits) on those damage mechanisms key parameters, which are either:

- Type 1 limits – limits beyond which damage occurs and consequently beyond which the fuel bundle design does not meet one or more of its design requirements; or
- Type 2 limits – limits which represent the maximum (or minimum) values of those parameters for which the fuel bundle design has been verified to satisfy its design requirements; in other words, limits beyond which the fuel may still not be damaged but for which the necessary analysis and validation tests have not been performed (referred to as 'fuel performance indicators').

These Type 1 and Type 2 limits[12] are associated with key parameters belonging to either the fuel bundle design envelope (examples of such key parameters – bundle specific dimensions – are provided in Section 2.2.1.1) or to the fuel bundle's operating envelope (i.e. parameters describing the environment of the fuel bundle). Examples of limits associated to the fuel bundle's operating envelope include:

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[12] Type 1 and Type 2 limits are based on Canadian practice. In other PHWR States, these limits can be sorted or represented in other expression methods.

- Limits related to bundle power and burnup;
- Stress corrosion cracking defect threshold curves, generally referred to as the FUELOGRAM defect threshold curves;
- Power cycling limit;
- Channel power limit;
- Limit on channel flow rate;
- Limits on axial and impact loads;
- Crossflow limits.

As discussed in Appendix II, the Type 1 and Type 2 limits have been used by the Canadian industry to derive fitness-for-service criteria for AOOs.

The fuel acceptance criteria reported in Section 3.1 were established:

- Through various fuel design qualification activities consisting of in-reactor (research reactors) and out-of-reactor tests and analyses aimed at demonstrating that, for the allowable range of normal operating conditions and AOOs, the fuel (element and bundle) design meets each design requirement imposed by plant design and operation without any damage to the fuel;
- From data collected through in-bay fuel inspections and post-irradiation examination over several decades of operating experience;
- With account taken of their impact on accident analysis results.

In Canada, fuel acceptance criteria for operational states (normal operations and AOOs) have an important operational support role. They are key inputs in the:

- Determination, for normal operations and AOOs, of the settings of the reactor control systems such as those for the core power/flux and the heat transport system parameters of pressure, flow, temperature [14];
- Determination of the plant's safe operating envelope [15];
- Derivation (see Appendix II) of a fuel fitness-for-service criteria that reactor operators propose to use after certain slow events with frequencies of occurrence $\geq 10^{-2}$ per reactor year [16, 17].

2.2.2. Accident conditions

2.2.2.1. Design objectives

For accident conditions, the following objectives[13] need to be met:

(a) Components of the reactor core and its associated structures should be designed with account taken of the safety functions to be achieved;

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[13] The performance of the fundamental safety functions is considered successful, from a fuel design perspective, if the objectives (b) and (c) above are met.

(b) The plant's fundamental safety function of removal of heat from the reactor core is performed to the degree of effectiveness required to ensure, for DBAs and DECs without significant fuel degradation, coolable core geometry and coolable bundle geometry;

(c) Fuel sheath failures are kept to a minimum[14].

Maintenance of coolable core geometry implies – for accident conditions – no more than one channel failure (commonly referred to as 'single channel events' in the accident analysis) in a horizontal channel-type PHWR [18], and coolable bundle geometry means that the channel decay heat can be removed from the bundles in a channel, in the long term without further fuel damage, and without reliance on the moderator.

These objectives are consistent with Requirement 44 of SSR-2/1 (Rev. 1) [5] on the structural capability of the reactor core, which states:

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

A key step in achieving the above objective (a) consists in identification of the safety functions performed by the reactor core and its associated structures. The safety functions performed by the fuel and the PHTS, the reactor core components of interest in this publication, are discussed in Section 3.2.1 and in Appendix III.

The fuel acceptance criteria for accident conditions reported in Section 3.2, whose key characteristics are described below, are among the defence in depth provisions introduced to ensure that the above objectives (b) and (c) are met with enough margins. These fuel acceptance criteria are related to the failure mechanisms that may challenge, under accident conditions, the integrity of the fuel matrix, fuel sheath and fuel channel. Regarding the fuel channel, only the pressure tube failure mechanisms that could be activated, as a result of fuel behaviour, are covered in this publication.

The failure mechanisms discussed in Section 3.2 and their associated fuel acceptance criteria are not organized according to accident scenarios since most of them are potentially active in more than one class of events. For instance, in CANDUs, fuel sheath embrittlement due to oxidation/hydriding is relevant to a large loss of coolant accident (LOCA) and other accident scenarios such as small LOCA, single channel events, LOCA with impairment of emergency core coolant injection, loss of regulation, and auxiliary system failures such as that of the shutdown cooling system.

For some of the failure mechanisms discussed in Section 3.2, there are no reported fuel acceptance criteria; this is not necessarily because none are used or being developed, but is probably due to the fact that, for most of them, no information about these criteria was specifically requested from the PHWR States. Given that significant research and development work has already been carried out to understand these mechanisms, and that some of those

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[14] The safety requirement regarding the maintenance of sheath integrity is usually more stringent for events with a higher frequency of occurrence. For instance, in Canada the current practice is that fuel sheath failures for all DBAs other than for large break loss of coolant accidents (LOCAs) and single channel events are not allowed.

failure mechanisms could also be of interest to DECs with core melt, discussions about those failure mechanisms have been included in Section 3.2.

2.2.2.2. Key characteristics

The value of each of the key parameter characterizing a specific barrier failure mechanism, and which demarcates the zone of essentially no failures from the zone where failure can occur with non-negligible likelihood, is referred to as the 'failure point' for that parameter. Determination, for a given key parameter, of its failure point need to be derived from experiments that identify the limitations of the material properties of the fuel (element/bundle) and, according to [19], need to:

(a) Be based on experimental data obtained under sufficiently representative conditions;

(b) Be set close to the values indicating non-failed states (rather than close to data for failed states), especially where the experimental data are limited;

(c) Account for measurement uncertainties and data scatter.

The fuel acceptance criterion associated with a barrier failure mechanism specifies bounds (limits) on the values of the key parameters that govern that failure mechanism process and which, if complied with during an accident sequence, prevent barrier failure due to that mechanism. These bounds are determined by establishing, for each of the key parameters, a margin to the key parameter's failure point to account for the following [19]:

- Data derived from experimental conditions which are not typical for reactor operating conditions (i.e. temperature, pressure, neutron flux, burnup);
- Data derived from an experimental set-up differing from the reactor geometry (i.e. scaling distortions);
- Effects from ageing or from differences in manufacturing of the plant components and of the experimental components;
- Incomplete knowledge ('unknown unknowns', i.e. unexpected or unforeseeable conditions).

These aspects, inevitably present to some extent in any experiment, cannot be quantified and the determination of the size of the margin to the failure point is by necessity based on engineering judgement.

As shown in Fig. 1, for each of the key parameters characterizing the failure mechanism of a physical barrier, there are two types of margins related to the prescribed limiting value of that key parameter:

- A so-called 'analysis margin' with respect to the value of this parameter calculated using deterministic safety analysis;
- A so-called 'margin to failure point' to the value that the key parameter has at the failure point.

The 'analysis margin' for a given key parameter is the difference between its prescribed limiting value and its calculated value using the design basis deterministic safety analysis methodology.

In cases where:

- The level of knowledge of a failure mechanism is low;
- The current fuel and fuel channel analysis codes cannot model the selected key parameter; or
- The codes are not validated against the range of conditions associated to the failure point,

it is a common engineering practice to make use of one or more surrogate parameters to express the fuel acceptance criterion which, if met, will prevent barrier failure due to that failure mechanism. These fuel surrogate criteria also need to be based on experimental evidence and are defined such that the margin to barrier's failure point is more conservative.

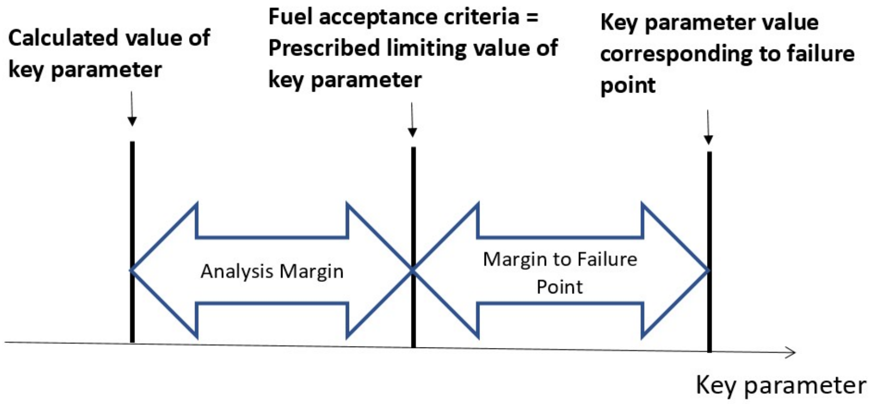

FIG. 1. Limits and margin model. The double arrows represent the two different types of margins for each key parameter characterizing a physical barrier failure mechanism.

2.3. REGULATORY REQUIREMENTS AND PRACTICES IN STATES WITH PRESSURIZED HEAVY WATER REACTORS

2.3.1. Canada

In order to gain assurance that fuel performs safely, in other words, that it remains fit for service in normal operation and following events such as those discussed in Appendix II, the CNSC [9, 20–22] has implemented a set of regulatory checks which include a requirement for licensees to implement the following:

- A systematic fuel design and qualification process leading to the determination of the fuel acceptance criteria for normal operating conditions, DBAs and DECs without significant fuel degradation;
- A multidisciplinary programme for assessment of fuel performance;
- An annual reporting to the CNSC on fuel monitoring and inspection results which include, among other things, information regarding significant deviations from the fuel acceptance criteria for normal operating conditions and proposed corrective actions;
- Quarterly report on safety performance indicators [20];

- Event reports and notifications;
- A submission, for regulatory review, of fitness-for-service assessment after certain slow events (see Appendix II).

In Canada, fuel acceptance criteria for operational states and accident conditions are part of the licensees licensing basis. They are proposed by the designer or licensee, and subject to detailed regulatory review. Requirements and guidance regarding these criteria are provided in the CNSC regulatory documents REGDOC-2.4.1 [8] and REGDOC-2.5.2 [9].

2.3.2. China

In China, fuel design is required to comply with the safety legislation HAF 102 [23]. Section 4.2 of HAF 102 establishes requirements for fuel element design, which are described below:

“The design of fuel elements shall be such that they satisfactorily withstand the intended irradiation in the reactor core after deterioration processes.

Fuel element design shall consider following deterioration factors: external pressure of coolant, additional internal pressure due to fission product within fuel element, irradiation effect of fuel and other materials in fuel assembly, pressure and temperature change induced by power change, chemical effect, static load, dynamic load by flow induced vibration and mechanics vibration, heat transfer behaviour change induced by deformation and chemical effect. Margins shall be taken account for uncertain factors of data, calculation, and fabrication.

Fuel design limit including permitted fission product release rate shall not be exceeded in normal operation, AOO transient shall not lead to significant deterioration of fuel element, fission product release shall be kept to a minimum. Design of fuel assembly shall provide convenience for inspection of fuel structure and component. During accident condition, fuel element shall remain in position, the deformation shall not reach the extent that would prevent sufficient post-accident core cooling, and specified limits shall not be exceeded.”

2.3.3. India

In India, fuel acceptance criteria for normal operation, AOOs and accident conditions are part of the licensing basis. They are proposed by the licensee, and subject to detailed regulatory review. Requirements and guidance regarding these criteria are provided in the AERB regulatory documents, for example the AERB Safety Guide on Fuel Design for Pressurized Heavy Water Reactors, AERB/SG/NPP-PHWR/D-6 (under revision) [24].

The design of fuel elements and bundles is required to be such that they satisfactorily withstand the intended irradiation and environmental exposure in the reactor core despite all processes of deterioration that can occur under all operational states.

The design of fuel bundles needs to consider their post-irradiation handling and storage including those damaged during usage or handling.

Specified fuel design limits are not allowed to be exceeded in normal operation, and conditions that may be transiently imposed on fuel bundle during anticipated operational occurrences are not allowed to cause significant additional deterioration. Fission product leakage needs to be restricted by design limits and kept to a minimum.

Design is required to ensure for timely detection and removal of any failed fuel from the core during nuclear power plant operation. In design basis accidents, the fuel bundles need to remain in position and not to suffer distortion to an extent that would render post-accident core cooling ineffective; and specified fuel integrity limits are not allowed to be exceeded.

The aforementioned requirements for reactor and fuel element design shall also be maintained in the event of changes in fuel management strategy or operational conditions during the plant life.

2.3.4. Korea

Korean regulation requires that a quality assurance programme needs to be established to guarantee the performance of the fuel during normal operation, anticipated operational transients and accident conditions. This quality assurance programme needs to cover design, manufacturing, transportation and storage of the fuel.

The "Regulations on Technical Standards for Nuclear Reactor Facilities" also state that "Specified Acceptable Fuel Design Limits (SAFDLs)" are defined, and they do not permit any failure of fuel rod during normal operation and AOOs.

There are no specific guidelines for the safety review of the CANDU fuel system in Korea. Instead, the Korea Institute of Nuclear Safety (KINS) staff selectively utilize the safety review guidelines of LWR fuel system (KINS/GE-N001) and Canadian regulations.

Korean practice for CANDU fuel regulation is descriptive; the regulatory body reviews applicant's methodologies using the relevant pressurized water reactor (PWR) fuel regulations and CANDU fuel experiences.

2.3.5. Romania

In Romania, the National Commission for Nuclear Activities Control (CNCAN) has licensed the nuclear fuel plant where the CANDU nuclear fuel bundles are manufactured, and also the research reactor where the nuclear fuel is tested to ensure that all the requirements for safe operation are met.

The regulatory framework for nuclear safety includes the nuclear safety regulations on the design and construction of nuclear power plants that established requirements for the nuclear fuel. These requirements aim to avoid failure of the fuel elements due to thermal or hydraulic causes during normal operation, during transients and in design basis accident conditions of moderate frequency.

In accordance with the regulations, nuclear fuel elements and assemblies are required to be designed to withstand satisfactorily the anticipated irradiation and environmental conditions in the reactor core in combination with all processes of deterioration that can occur in normal operation and in anticipated operational occurrences.

The fuel design analysis needs to account for all known degradation mechanisms of the fuel. Allowance needs to be made for uncertainties in data, calculations and in manufacture.

Specified fuel design limits are not allowed to be exceeded in operational states so as to ensure that AOOs do not cause significant further deterioration.

Leakage of fission products is not allowed to exceed the design limits; furthermore, it needs to be kept to a minimum. Fuel assemblies are designed to permit inspection of their structure and component parts as appropriate after irradiation.

These design requirements need to be met even for changes in fuel management strategy or in operational states over the operating lifetime of the plant.

In Romania, nuclear fuel acceptance criteria for operational states and accident conditions are part of the licensing basis for nuclear power plants. The technical specifications and the acceptance criteria are proposed by the designer or licensee, and subject to detailed regulatory review and acceptance. CNCAN also approves the quality assurance plans and inspection and test plans for the manufacturing of the nuclear fuel.

3. REVIEW OF FUEL ACCEPTANCE CRITERIA FOR OPERATING PRESSURIZED HEAVY WATER REACTORS

3.1. OPERATIONAL STATES

3.1.1. Technical basis

This section compiles, for currently operating PHWRs, the fuel acceptance criteria that are being used for operational states[15]. This information is based on responses to the IAEA questionnaire and supplementary information provided by some States.

Fuel acceptance criteria for operational states are intended for use to ensure that:

- The leaktightness and structural integrity of fuel elements are maintained to prevent fission product transport into the coolant;
- The structural integrity of fuel bundles is maintained mainly to prevent overheating of the fuel sheath;
- Fuel bundle compatibility with interfacing systems is maintained.

Fuel acceptance criteria applicable to fuel elements are targeted at preventing primary defects of the fuel sheath, based on credible fuel damage mechanisms. Secondary damage cannot occur in the absence of primary defects. Nevertheless, some primary defects are likely, and it would be wise to have capacity to limit secondary damage to reasonable levels. This is usually addressed through monitoring radioactivity in the coolant and taking steps to reduce it when excessive. On-line fuelling makes it relatively easier to do so in horizontal channel-type PHWRs while maintaining essentially full reactor power at normal flux shapes for significant durations.

Fuel acceptance criteria being reported in Section 3.1 address damage mechanisms known to be active in current operating reactors and are categorized into three major groups [13]:

[15] All fuel acceptance criteria reported by Canada apply to normal operations and AOOs. As discussed in Appendix II, the fuel acceptance criteria reported by Canada in Section 3.1 have been used to derive fitness-for-service criteria for AOOs. This work on AOOs is part of the Canadian industry's implementation of CNSC's regulatory document REGDOC 2.4.1 on Deterministic Safety Analysis [8].

- Criteria to ensure thermal integrity of fuel;
- Criteria to ensure structural integrity of fuel;
- Criteria to ensure compatibility of fuel with interfacing systems.

Thermal challenges, if excessive, can potentially cause melting in the pellet or in the structural materials (mainly Zircaloy). Pellet melting can result in (a) escape of additional fission gas from the UO_2 matrix into the pellet-to-sheath gap and (b) thermal expansion of the pellet, causing the sheath to rupture. This combination can send additional radioactive fission products into the coolant. Excessive amounts of radioactive fission products in the coolant could represent a radiological hazard to the plant staff. Fuel acceptance criteria to ensure thermal integrity of fuel are reported in Section 3.1.2.

Excessive mechanical damage can potentially cause holes, cracks, or breaks in structural materials, with consequences similar to those due to the thermal challenges described above. Criteria to ensure structural integrity of the fuel are reported in Section 3.1.3.

Incompatibility with the fuel channel can potentially cause the fuel bundle to jam inside the reactor or in its accessories such as fuel handling equipment, interfering with its proper insertion or eventual removal. Incompatibility can also potentially harm neighbouring or interfacing components through processes such as crevice corrosion. The criteria to ensure compatibility of fuel with interfacing systems are reported in Section 3.1.4.

Fuel acceptance criteria applicable to fuel bundles are targeted at maintaining geometry compatible with the design basis, i.e. to prevent loss of structural integrity or excessive deformation of the bundle.

Overall, the use of the criteria reported in this section have resulted in excellent fuel performance. For instance, in Canada, fuel bundle defect rates remain very low (less than 0.1% of a bundle basis [25]) and tend to be caused mostly by fretting due to debris.

As mentioned in Sections 2.2.1.2 and 2.2.1.3, some of the fuel acceptance criteria reported in this section have been used by the Canadian nuclear industry to derive fuel fitness-for-service criteria that Canadian reactor operators propose (see Appendix II) to use, following some events, to return to full power operation without any fuel and fuel channel inspection or analysis; such events, referred to as ‘slow events’, belong (in terms of frequency of occurrence) to the category of AOOs and include small break LOCA, electric failures causing a loss of flow, and slow loss of reactivity control.

It is noted that most of reported criteria from Argentina in Section 3.1.2 have not been verified yet for application to the Embalse nuclear power plant.

3.1.2. Criteria to ensure thermal integrity of the fuel

3.1.2.1. Overheating of the fuel pellet

PHWR fuel can potentially operate at high linear heat generation rates, in comparison with other reactors such as LWRs. Moreover, the heat generation rate can increase even further locally due to effects such as end flux peaking. For fuel pellets located near fuel bundle junction areas (where bundles are in contact with adjacent bundles), the fission rate increase may be as high as 10 to 15%. The increased heat generation, however, is balanced by the increased heat transfer by axial as well as radial heat flow from the end pellet. These conditions can lead to increased peak pellet temperatures, putting end pellets at greater risk of melting during an event.

Reported criteria

(a) During normal operations and during AOOs: The maximum temperature has to be less than the melting temperature of UO_2 for 90% of the pellet cross-section at the hot spot. Fuel melting temperature limit is approximated as 3073 K (2800ºC). The decrease of the melting point with burnup is neglected in the calculations (for vertical channel-type PHWRs, Argentina);

(b) The centreline pellet temperature has to be less than stoichiometric UO_2 melting temperature (e.g. 3120 K ± 30 K [26]) (Canada);

(c) The centreline temperature has to be below the melting temperature. The latter is given by:

$$T = 2805 - 0.133 \times \omega \qquad (1)$$

where T is the melting temperature in °C and ω is the fuel burnup in MW·h/kgU; this accounts for experimental uncertainties and burnup effects (China);

(d) During normal operations and during AOOs: The calculated maximum fuel pellet centreline temperature, with due allowance for irradiation, tolerances, uncertainties, etc., has to remain below the melting point (India);

(e) During normal operations and during AOOs: Centreline melting is used for fuel melting indication. It is evaluated for both ends of fuel rods at the outer ring which experience the highest neutron flux. Pellet melting temperature is 3078 K (2805°C) for fresh fuel and decrease at the rate of 32 K per 10 MW·d/kgU. Margin needs to exceed 20%. This evaluation needs to done at both ends of fuel rods in the outer ring which experience the highest neutron flux (Korea);

(f) The fuel centreline temperature needs to be much below the melting point of UO_2 which is 3033 K (2760ºC) (Pakistan);

(g) Criterion is based on centreline melting. Actual melting occurs at 3113 K (2840ºC); for conservatism, 3038.15 K (2765°C) is used for a burnup of 300 MW·h/kgU (Romania).

Discussion

The value of 3120 K ± 30 K [26] for the melting point of stoichiometric UO_2 is the value recommended by Rand et. Al. [27] from their analysis of fourteen experimental studies (over a period of 20 years) and has been accepted internationally. The value of 3120 K ± 30 K is used by MATPRO [28] as input in the calculation of the melting temperature of an uranium dioxide pellet as a function of fuel burnup.

For normal operations, pellet centreline melting is not permitted. If the pellet melts, the resulting volumetric expansion of the pellet may potentially push the sheath past breaking. Further, if the molten UO_2 flows, it can potentially reach the Zircaloy and melt it too.

For normal operations, the maximum pellet centreline temperature is usually considered to be below the pellet melting point with adequate margin which typically bounds the uncertainty of the calculation correlations.

3.1.2.2. Overheating of the fuel sheath: critical heat flux

Overheating of the fuel sheath can quickly degrade the strength of the sheath material. It can also lead to rapid oxidation, crevice corrosion, overstrain, creep and in worse case melting of the fuel sheath. Excessive bowing can occur if the overheating is localized and large. Excessive local bowing can potentially damage a neighbouring fuel element and even the pressure tube.

While liquid dryout on fuel sheath surface does not necessarily cause an abrupt rise in fuel surface temperature, a prolonged dryout may lead to a level of overheating that triggers the damage mechanisms mentioned above, resulting in the fuel (element/bundle) no longer meeting some of its design requirements, and therefore no longer being fit for service.

Reactor core design and operation apply limits on some of the thermohydraulic parameters to ensure adequate cooling to the fuel. In particular, the thermohydraulic design of the fuel assures that specified thermal safety limits are not exceeded in operational states. Although these thermal limits are not bounds on parameters directly related to properties describing the damage mechanisms mentioned above, their values are nevertheless fuel bundle design dependent, and their formulation are among the defence in depth provisions (measures) taken to address the damage mechanisms mentioned earlier, and ensure that the safety objectives formulated in Section 2.2.1.1 are met; for those reasons, these thermal limits are reported as fuel acceptance criteria for operational states.

A thermal safety limit is commonly specified as a margin to the critical heat flux (CHF), such as a minimum CHF ratio. The CHF is the thermal limit, beyond which heat transfer from the heated surface to the coolant reduces significantly, causing a rapid rise in surface temperature. The CHF ratio is the ratio of CHF over local heat flux. In general, empirical methods, such as empirical correlations, are employed to predict CHF values. These empirical CHF methods are bundle-design specific and are derived on the basis of fuel bundle simulation experiments, covering the range of thermohydraulic conditions (e.g. coolant pressure, temperature, flow) encountered in all the plant states (normal operations and accident conditions) considered in the design of the nuclear power plant. Sufficient margin to CHF is preserved to address uncertainties of various parameters affecting CHF or CHF ratio.

Reported criteria

(a) The departure of nucleate boiling ratio has to be 1.25 minimum. There are additional margins applied to account for the uncertainty in the initial normal operation conditions and also in case of AOO coolant pumps shut-off (Argentina);

(b) Under normal operating conditions, the fuel bundle has to maintain sufficient margin to dryout. No quantitative thermal safety limit is specified in terms of avoiding occurrence of dryout on fuel sheath surface. However, a sufficient margin to CHF is demonstrated to accommodate various uncertainties. In Canada, avoiding fuel sheath dryout, under normal operating conditions, is a fuel bundle design requirement (Canada);

(c) Default CATHENA heat transfer and critical heat flux (Groeneveld lookup tables) correlations are used. For post-dryout, the Groeneveld-Delorme correlation is used (China);

(d) The minimum CHF ratio has to be 1.3 under normal condition and 1.1 under anticipated operational occurrences (India);

(e) The critical power ratio has to be $\geq$1.0 and Xc-BL correlation is used (Korea);

(f) The maximum heat flux in the hot channel is limited to 1.08 MW/m^2 (108 W/cm^2) and the corresponding clad temperature of 322°C at normal coolant flow (Pakistan);

(g) The peak sheath temperature has to be less than 350ºC for normal operating conditions, with sufficient margin to the limit for onset of sheath dryout using Groeneveld critical heat flux lookup tables. Sheath oxidation is simulated using Urbanic-Heidrick correlations (Romania).

Discussion

Water cooled nuclear reactors (with the exception of supercritical pressure reactors) are limited in power to avoid occurrence of CHF under normal operations, AOOs and some event conditions. Critical heat flux is the thermal limit of a heated surface, reached due to phase change of liquid coolant, such as bubble formation, bubble coalescence, and vapour film formation, which abruptly decreases the heat transfer, causing localised overheating of the surface. The behaviour of CHF depends on flow conditions. Under subcooled or low quality conditions, such as those encountered in PWRs, CHF occurs at a relatively high heat flux and appears to be associated with the cloud of bubbles adjacent to the heated surface which reduces the amount of incoming water and acts like an insolation layer on the surface. When this behaviour occurs, surface temperature abruptly rises to a high value. This kind of CHF can be classified as departure from nucleate boiling, commonly encountered in PWRs. Under high quality conditions such as those in CANDU-type PHWRs, CHF occurs at a lower heat flux. The flow pattern prior to CHF is usually annular, and the fuel surface is normally covered by a liquid layer. When the evaporation rate is high enough, the liquid layer can no longer be sustained and dry patches will develop on the fuel surface, which reduces heat transfer from the surface to the coolant. Since the velocity in the vapour core is high, post-dryout heat transfer is much better than for low quality cases; wall temperature rises are moderate and less rapid. In general, the dryout-type CHF is of interest to PHWRs.

As mentioned earlier, a prolonged dryout may trigger various fuel sheath damage mechanisms including melting of the fuel sheath. For this reason, prevention of fuel sheath dryout (or dryout-type CHF) has been chosen by many States as a thermal safety limit for normal operation, some AOOs and some DBAs of horizontal channel-type PHWRs. In other scenarios where exceedance of CHF is permitted, post-CHF heat transfer calculations are performed to evaluate the fuel sheath temperatures (the influencing parameter for some other fuel criteria).

The thermohydraulic design of horizontal channel-type PHWRs is limited by the necessity to maintain an adequate safety margin between the operating heat flux and the CHF of a fuel bundle. Given that neither the operating heat flux nor the CHF is measurable in the reactor core, the critical channel power has been used as a surrogate parameter of CHF in maintaining the necessary margin to CHF. The critical channel power is the channel power at which the fuel string[16] starts experiencing fuel sheath dryout anywhere in the fuel string. It is simulated using a computer code, based on the knowledge of CHF, the axial power profile, and the thermohydraulic conditions of the reactor core.

In operation of horizontal channel-type PHWRs, channel powers are related to the reactor power and are measurable. Therefore, the necessary safety margin between the operating heat flux and the CHF of a fuel bundle is ensured by maintaining an equivalent margin between the channel power and the corresponding critical channel power, or by maintaining a minimum critical power ratio (the ratio of critical channel power over channel power). The minimum critical power ratio encompasses the margin of operating heat flux to CHF of fuel bundles. No information on this criterion was specifically requested from the PHWR States.

As discussed in Appendix II, which covers the Canadian approach to the formulation of fuel fitness-for-service criteria, CHF may be reached for a very short period of time, and for low sheath temperature (<450ºC), without causing damage ('not damaged' as defined in Section 2.2.1.1) to the fuel; this fact has been used to derive the fitness-for-service criteria for AOOs presented in that Appendix.

A summary of the current state of knowledge on fuel thermal hydraulics for CANDUs can be found in [29, 30].

3.1.3. Criteria to ensure structural integrity of the fuel

3.1.3.1. Element internal gas pressure

Nuclear fission generates, among other products, gases inside the oxide matrix. Over time, fission gases can reach the open gap inside the fuel element and increase the internal pressure.

Reported criteria

(a) During normal operations and operational incidences [31, 32] (Argentina):

  (i) Element internal pressure has to be less than the minimum coolant pressure (for horizontal channel-type PHWRs);

  (ii) Element internal pressure has not to increase fuel/cladding gap (for vertical channel-type PHWRs);

(b) For the majority of the reactors, the maximum bundle burnup is limited to ensure that, among other considerations, the amount of fission product release is bounded (Canada);

(c) During normal operations and during AOOs: From considerations of fission gas pressure, the linear heat generation rate is limited to a specific value, for example, to less than 61 kW/m [33]. This value applies to specific designs (e.g. 220 MW(e), 540 MW(e) or 700 MW(e)) and is dependent on coolant pressure (India);

(d) For normal operations and AOOs (Korea):

[16] The term 'fuel string' refers to the set of twelve or thirteen fuel bundles present in each fuel channel during normal operation.

(i) Element internal pressure has to be less than the PHTS pressure,

(ii) If element internal pressure is greater than the PHTS pressure,

- For short duration, tensile stress of the sheath has to be less than 80% of the ultimate tensile strength of the unirradiated sheath,
- For long duration, sheath strain has to remain below 0.3% to prevent local plastic deformation.

Discussion

Increases in fission gas release can potentially lead to excessive gas pressure in the fuel element above the coolant pressure. Because of deterioration in the heat transfer between the pellets and the sheath due to accumulated fission gases and increased gap size (i.e. reduced gap conductance), the pellet temperature can increase which can in turn further accelerate fission gas release. Excessive gas pressure may cause sheath strain which in turn can affect coolant flow and thus dryout. Excessive gas pressure can also threaten fuel integrity through local overstress/overstrain, especially at locations of stress concentrations and in conjunctions with corrosive internal environment in Zircaloy that has been embrittled through irradiation, oxidation, and/or deuteriding/hydriding.

From the above perspectives, the element internal gas pressure needs to be restricted to an acceptable magnitude. The criteria reported in this section have been formulated to avoid excessive gas pressure.

3.1.3.2. Stress corrosion cracking

Stress corrosion cracking (SCC) is a fuel sheath failure mechanism that occurs because of the combined effects of high stresses and high concentrations of corrosive fission products in the pellet-to-sheath gap. During irradiation, the sheath can be embrittled due to combined consequences of irradiation, oxides, and deuterides/hydrides. SCC in the embrittled fuel sheath occurs when the stresses on the inner surface of the sheath (as a result of pellet-cladding interaction) reach a certain limit under a corrosive environment.

The stresses can arise mainly due to power ramps and due to internal gas pressure; only the former situation is covered in this section. Power ramp stresses can be complicated by several situations, two of which are summarized here – ‘deconditioning’ and ‘residual stresses’ [34].

- Deconditioning – After a power reduction, the thermal contraction of the fuel pellets causes re-opening the pellet-sheath gap (or the gaps between the pellets fragments). If the reduced power operation is maintained long enough (i.e. extended reduced power operation), the fuel sheath will creep down and close the gaps again. The fuel element is then considered as re-conditioned at this lower power level. When the reactor core returns to full power at a later time, tensile stresses due to the power ramp will appear in the sheath. These residual stresses could increase the susceptibility to SCC driven by pellet-cladding interaction under corrosive fission product environments in the fuel element.
- Residual Stresses – These can be potentially significant during multiple ramps within a short period. Let’s say 2 ramps occur within, say 1 week, and the first high power is held constant during the period between the two ramps. The first ramp imposes stresses

on the sheath which may be less than the failure threshold. During the hold at high power, some of the stresses would relax, nevertheless, the relaxation may not be complete; therefore, some residual stresses would exist at the beginning of the second ramp. The second ramp would then add additional stresses of its own. The combined stresses – residual from first ramp plus incremental from the second ramp – may cause the sheath to fail even though individual stresses from either ramp may have been insufficient to do so by themselves.

Reported criteria

(a) Different Canadian utilities use different models as limits, for example the CANLUB FUELOGRAM model and the FUELOGRAM model [35] (Canada);

(b) During normal operations and during AOOs: Complex criterion is used based on operational parameters such as ramped power, change in power, burnup, and duration of stay at maximum power after the ramp (India);

(c) During normal operations (Korea):

- o Fuel has to withstand power variation caused by refuelling and other reactivity changes. Sheath failure due to pellet-cladding interaction is limited. Possibility of sheath failure is evaluated in terms of linear power and power increase versus burnup by utilizing some correlations;

- o SCC has to be limited also at the location of end plug weld;

(d) Fuel bundles has to withstand the power changes associated with fuelling (Romania).

Discussion

Irradiation prior to a power ramp can embrittle Zircaloy due to combined influences of fast neutrons, oxides, and deuterides/hydrides. During a subsequent power ramp, the embrittled Zircaloy can experience high tensile stresses due to pellet expansion. This would occur in the presence of a corrosive internal environment. The above combination can potentially crack the fuel element at locations of stress concentrations via SCC.

In PHWRs, SCC from power ramps has to date damaged fuel elements at circumferential ridges and at sheath/end cap junctions [34]. A PHWR fuel element also contains other locations of stress concentrations, such as junctions of sheath and appendages.

In addition, a typical PHWR sheath may potentially contain zones of significantly different microstructures, for example created by brazing and welding. They may potentially differ in their resistances to SCC.

The power ramp failure thresholds have been established by means of in-pile power ramp tests. These failure thresholds provide the lowest level of failures within the applicable burnup range. For CANDUs, the CANLUB FUELOGRAM model and the FUELOGRAM model [35] are typical threshold models.

In Canada, SCC has been effectively managed by operational procedures that reduce power increases during refuelling and by addition of CANLUB, a protective graphite layer coated on the inside surface of the fuel cladding tube [36]. The probability of a SCC sheath failure is estimated on the basis of power increase, final power and burnup diagrams with the codes mentioned above [35, 37]. The on-site fuel engineer has great flexibility in choosing which channels require fuelling. Power changes during refuelling may cause fuel defects if both a critical final power to which the fuel is ramped and a maximum change in power during the ramp are exceeded; both parameters are burnup dependent. There have been no power ramping failures using existing station fuelling procedures, so, during refuelling, the following parameters are limited [38]:

(a) Predicted bundle powers;

(b) Recycling bundle burnup or fuel bundle recycling;

(c) Fuel bundle burnup for movement of that bundle during fuelling.

3.1.3.3. Fuel failure due to static mechanical overstress and/or overstrain

During a variety of situations such as refuelling, structural components of the fuel bundle can potentially be exposed to relatively high loads and/or relatively sparse supports, leading to a potential for static mechanical overstress/overstrain.

Stress, strain, or hydraulic and fuelling machine loading limits for fuel sheath, fuel elements, end plate (of fuel bundles for horizontal channel-type PHWRs), and other fuel system structural features are reported below. The hydraulic and fuelling machine loads are supported by the column strength of the fuel element which is affected by the diameter, wall thickness and mechanical properties of the elements tubing. The fuel acceptance criteria reported are aimed at ensuring, for normal operation, that fuel system dimensions remain with operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.

Reported criteria for strain

(a) Sheath strain limit has to be (for vertical channel-type PHWRs, Argentina):

  (i) Less than 1% for fast power ramps,

  (ii) Less than 2.5% (elastic plus plastic) to avoid sheath damage due to long term pellet-cladding mechanical interaction;

(b) Power stations have, for normal operations, a fuel performance indicator (see definition in Section 2.2.1.2) that limits the sheath average diametral strain at pellet mid-plane (Canada);

(c) During normal operations and during AOOs (India):

  (i) Total, uniform, circumferential plastic strain in cladding has to be less than 1%,

  (ii) Sheath collapse is checked through collapse tests;

(d) As criteria to maintain structural integrity (Korea):

(i) For the sheath length supported by the pellets, sheath strain has to be below its ductility,

(ii) For the end cap, stress and strain have to be below their limits,

(iii) For the end plate, weld points have to remain intact,

(iv) For sheath length unsupported by the pellets, sheath deformation due to axial ridging, sheath collapse or sheath creep collapse needs to be below the limit,

(v) End plate has to be designed to withstand non-uniform axial expansion of fuel elements and to limit end flux peaking,

(vi) Thermomechanical loads needs to be reviewed in safety evaluation;

(e) The sheath would fail by low ductility if the strain due to athermal glide exceeds 0.4% before 95% of the sheath microstructure has annealed (Romania).

Reported criteria for stress

(a) Sheath stress limit (for vertical channel-type PHWRs, Argentina):

(i) Primary membrane stress (expressed by M) has to be below the minimum of 0.9×yield strength and 0.5×ultimate tensile strength,

(ii) Sum of primary membrane stress and bending stress (expressed by B), (M+B), has to be below the minimum of 1.35×yield strength and 0.7×ultimate tensile strength,

(iii) Sum of primary membrane stress, bending stress and secondary membrane stress (expressed by S), (M+B+S), has to be below the minimum of 2.70×yield strength and 1.0×ultimate tensile strength;

(b) During normal operations, no sheath stress limits are used (Canada);

(c) During normal operations, integrity of fuel bundle needs to be assured during refuelling, including consideration of hydraulic drag. Stress and strain are limited to maintain the integrity of fuel rod end cap and of the end plate at weld points (Korea).

Reported criteria for force, load and pressure

(a) Elastic buckling and plastic instability limit has to be: maximum differential pressure less than critical pressure for elastic buckling and also less than critical pressure for plastic deformation (for vertical channel-type PHWRs, Argentina);

(b) Axial loads from ram of the refuelling machine are kept below certain force value. This limit is confirmed adequate through demonstration of avoidance of bundle buckling under the imposed load. However, because differences in design, maximum forces differ for different power stations (Canada);

(c) During normal operations and during AOOs, axial compressive load limits are specified for the bundle (India);

(d) During normal operations (Korea):

   (i) Worst case hydraulic loads and impact loads for normal operation including refuelling have not to exceed fuel assembly integrity limits, 7300 N of compressive force,

   (ii) Deformation due to drag load needs to be analyzed;

(e) Total mechanical force during fuelling has to be below 166 N (Pakistan);

(f) Fuel bundles have to withstand axial loads caused by hydraulic drag and fuelling machine ram (Romania).

Discussion

To minimize parasitic absorption of neutrons, typical PHWR fuel uses relatively thin structural components; diametral collapse of the sheath is also allowed (specific to horizontal channel-type PHWRs). These features often result in relatively high local stresses and strains at several locations, frequently well into the plastic range. As well, local stresses/strains are often highly multiaxial.

The remaining text in Section 3.1.3.3 was contributed by India. However, to a large extent it also applies to other States with PHWRs. The following loads/effects need to be determined conservatively from the bounding operating parameters (such as coolant flowrate, pressure, temperature, reactor power level, neutron flux), with appropriate combinations thereof:

- *Bundle droop* – Bundle droop can take place due to self-weight of fuel elements, differential temperature across the cross-section of the bundle and axial compressive load. This may reduce the gap between adjacent elements or the gap between outer fuel elements and pressure tube. The limiting value for the gap is obtained by subchannel analysis and the compliance ensured by inspection after fabrication and analysis.

- *Hydraulic loads* that result in

  - Vibrational loads – Flow induced vibrations cause fretting on spacers and fatigue at the joint between the element and the end plate,

  - Impact loads on the bundle during refuelling operation when the coolant flow pushes the bundle downstream,

  - Crossflow vibration loads on the bundle during refuelling, while passing over the liner-tube holes provided for coolant entry.

  Resonance vibrations of the fuel element, fuel bundle and coolant channel assembly need to be avoided while designing the primary heat transport system. Currently, the fuel bundles are qualified for above mentioned hydraulic loads by type testing.

- *Fatigue due to power cycles* – Loads due to differential axial expansion of the elements cause fatigue on the end plates. The fatigue performance of the end plate has to satisfy the necessary power cycles. The end plates of fuel bundle were qualified by analysis for low cycle fatigue performance.

- *Compressive loads on the bundle* – This axial load is caused due to hydraulic drag from all the bundles/free components, like fuel locator in the channel during channel-closed condition and during PHT hydrostatic testing. During refuelling operation, in addition to the above loads, fuelling machine ram loads become significant. For example, after inserting four fresh bundles in a coolant channel from upstream end, when shield plug of downstream end is being removed, the fuel string experiences compressive force due to hydraulic drag force, upstream fuelling machine ram force, and friction force. Hydraulic drag force and friction force on the fuel string are also experienced by elements of the last bundle while being held against the side stops of the fuelling machine. Currently, the fuel bundles are qualified for the design compressive loads by type testing.

- *Seismic loads* – The fuel bundle needs to withstand loads generated due to the Operating Basis Earthquake (SL-1 earthquake level) and/or the Safe Shutdown Earthquake (SL-2 earthquake level) without exceeding deformation limits which jeopardise cooling, fragmentation or severance of any bundle/component. Since the fuel bundle is a free component in the coolant channel assembly, the fuel bundle is qualified for the Operating Basis Earthquake and/or the Safe Shutdown Earthquake by impact tests.

- *Thermohydraulic effects* – The pressure drop in the fuel bundles in a channel need to be within the design provision of the PHTS. This is checked by out-of-reactor type testing. The gap between the fuel elements and pressure tube has to remain acceptable, considering the fuel bundle and pressure tube wear, fretting and other dimensional changes due to irradiation (pressure tube growth, creep and swelling) during operation. The subchannel analysis and the element thermal analysis are carried out with the minimum gap and the element parameters with this gap are checked. Subchannel analysis is carried out with maximum expected coolant channel diametrical creep and the resulting thermal parameters are checked.

In Indian practice, the different weld joints in the fuel bundle assembly are addressed, taking into account the loads acting on the fuel element/bundle during its handling and movement in the reactor and outside the reactor by the fuelling machine and the fuel transfer system. The deterioration of the fuel element/bundle due to environment during its in-core residence is taken into account.

3.1.3.4. Hydriding

Fuel elements are fabricated with some amount of internal hydrogen. In addition, fuel can also pick up additional hydrogen/deuterium during operation. Under certain conditions, the hydrogen/deuterium can accumulate at a few preferred locations, for example at the relatively cooler locations or at locations of relatively higher stresses.

For convenience, the descriptions in this section have used ‘hydrogen’ and ‘deuterium’ interchangeably; likewise, for ‘hydride/deuteride’. The numeric limits, however, have to distinguish between them.

Reported criteria

(a) Deuteride/hydride concentration has to be less than 1000 ppm (for vertical channel-type PHWRs, Argentina);

(b) Due to the short in-reactor residence no lifetime limits, in addition to the fabrication specifications, are necessary (Canada);

(c) As acceptance criteria (Korea):

(i) Hydriding has to be limited in the fuel element and in the fuel system,

(ii) Total hydrogen content in an as-fabricated fuel sheath has to remain below 0.8 mg.

Discussion

Excessive local hydrides can reduce the local ductility of Zircaloy, rendering it less capable of carrying its loads during operation in the reactor.

Some PHWR States may not consider a lifetime criterion for hydriding because PHWR's natural uranium fuel achieves relatively short burnups. Nevertheless, it is still advisable to address the impact of hydrides on the ductility of the sheath in the design analysis, in particular, wherever sheath ductility is used as a safety criterion parameter.

3.1.3.5. Oxidation and crud

The sheath oxidizes during operation in the reactor. It can also collect deposits (interchangeably called 'crud') which are iron-based oxides circulating in the reactor core.

Reported criteria

(a) External oxide layer thickness has to be less than 70 µm (for vertical channel-type PHWRs, Argentina);

(b) No Type 1 limits (see definition in Section 2.2.1.2) for sheath oxidation thicknesses were reported. However, if fuel is discharged with sheath oxidation larger than the value of a predetermined fuel performance indicator (see definition in Section 2.2.1.2), power stations need to investigate the cause of this abnormal sheath oxidation thickness and apply any necessary corrective actions to ensure fuel coolability is maintained (Canada);

(c) Oxidation and buildup of oxidation products (crud) need to be limited (Korea);

(d) As acceptance criteria (Pakistan):

(i) Oxygen level has to remain below 0.05 ppm,

(ii) Crud level has to remain below 0.5 ppm.

Discussion

Natural uranium fuel has relatively low burnup. Also, PHWRs use separate fluids for coolant and for moderator. As the coolant chemistry control is maintained well within defined limits and the avoidance of certain alloying chemical elements in the PHTS (e.g. Cu), the amount of crud deposited on fuel surfaces is minor. Similarly, oxygen suppression system maintains the oxygen concentration in the coolant to very low values, and the short residence of fuel bundles in the PHTS has a significant effect on the oxidation phenomena. Sheath aqueous oxidation is low (few μm) [39]. In a low oxygen control environment, magnetite is one of the most commonly found iron-based compounds produced by the interaction between coolant and PHTS piping. As coolant chemistry control (Ph in particular) is maintained well below defined limits during full power operation, the iron-based compound amounts deposited on fuel surfaces or other bundle components are usually minor [40].

Decades of operational experience indicate that for current PHWRs with natural uranium, fuel oxidation and crud are usually insignificant and, therefore, it could be acceptable not to address the impact of oxides, crud and deposits on fuel surfaces in the design of natural uranium fuel for PHWR. However, recent experience in Canada indicates that coolant control (Ph in particular) issues during shutdowns can later result in crud formation when returning to full power operations. In 2008, black deposits were initially observed in some reactors during in-bay fuel inspections. Less than optimal Ph control of the PHTS during outages was identified as the root cause of the problem. Corrective actions were taken and have been successful in addressing the issue [41].

3.1.3.6. Fuel mechanical rupture due to impact loads such as refuelling and/or start/restart

A mechanical rupture refers to a defect in a fuel bundle or element caused by an externally applied force, such as significant axial impact loads during refuelling and/or start/restart. Depending on the refuelling scheme, this can potentially occur after some amount of irradiation embrittlement of Zircaloy.

Reported criteria

(a) During normal operations, the impact velocity of a bundle (inserted during refuelling with flow) on a fuel string in the fuel channel has to be less than a prescribed velocity; these limits depend of the reactor design (Canada);

(b) During normal operations, integrity of fuel bundle needs to be assured during refuelling impacts (Korea).

Discussion

Similar to static loads, impacts loads may also result in relatively high local stresses and strains during fuelling with flow operation. Impact tests have recorded local plastic deformations.

During on-power fuelling in horizontal channel-type PHWRs, the reactor coolant carries the new fuel bundle into the fuel channel until it hits an existing bundle in the channel. Therefore, the new as well as the old fuel bundles need to withstand normal refuelling impact loads. The impacts occur when the coolant sweeps the first bundle loaded downstream, allowing it to hit the stationary fuel bundle string. Impact loads on cold and hot irradiated bundles need to be minimized, particularly loads which are not applied squarely to the bundle ends. Also, the fuel channel needs to be capable of withstanding the effects of fuel bundle impacts (against the fuelling machine ram) that occur during fuelling.

3.1.3.7. Fatigue

Fatigue is defined as “a failure phenomenon associated with work-hardening of materials caused by fluctuating or repeated loads that result in increased brittleness and reduced service life” [42]. Fatigue is known as “characteristic of ductile materials but the final failure is rapid and characteristic of brittle fracture” [42].

The following two conditions are known to lead to fatigue:

(a) A relatively large, fluctuating applied stress;

(b) A sufficiently large number of stress cycles.

Fatigue failure is often caused at relatively low applied stress but is inactive within specified design loads. The stress can be generated due to mechanical, thermal or a combination of both loads, and can alternate between compression and tension, or simply alternate between high and low values.

Alternating stresses and strains are structural damage mechanisms of the fuel bundle which can be caused by a variety of situations, such as acoustics, flow, or power manoeuvring. If too large and/or too numerous, they may create potential for fatigue failures, sometimes aided by the corrosive environment inside a fuel element.

Reported criteria

(a) Cyclic stresses in the cladding due to dynamic loads has to be less than 50 MPa (50 N/mm^2) (Argentina);

(b) Time limits are established for residence in crossflow locations. Furthermore, some reactors in Canada also specify, for components subjected to cyclic strain during normal operating condition, that the sum of fatigue life-fractions consumed at different strain amplitudes shall be less than 1.0. The fatigue life shall be based on verified data and include safety factors (Canada);

(c) During normal operations and during AOOs: A low cycle fatigue limit is applied (India);

(d) Model of O'Donnell and Langer is used, with >50% margin to failure (applied to normal operations) (Korea).

Discussion

Fuel failures due to fatigue are not common in operating PHWRs.

In the past, fatigue failures have been reported in end plates due to excessive residence in crossflow regions, and due to acoustics in some reactors. During repeated changes in power, some fatigue failures in fuel sheaths have been reported at longitudinal ridges in Canada. Additional failures have been attributed to corrosion-assisted fatigue due to repeated changes in power, for example in Pakistan [34] and in Argentina [34]. Thus, experience to date suggests that the above locations are at credible risks of potential fatigue failures.

Fatigue failures are currently minimized through criteria established and met in the design, manufacture and qualification process.

3.1.4. Criteria to ensure compatibility of fuel with interfacing systems

3.1.4.1. Failure due to excessive interaction loads along the fuel string

This mode of failure refers to the fuel string potentially expanding to become longer than the available length of the cavity in the fuel channel of a horizontal channel-type PHWR. If excessive, this could damage the fuel channel components that interface with the fuel bundle. For instance, such an expansion of the fuel string could result in deforming the fuel bundle into an unknown geometry which could result in fuel element-to-pressure tube contact and potential pressure tube damage/failure.

Reported criteria

(a) Constrained expansion of the fuel string is not allowed in any reactor. This is implemented by ensuring that the fuel bundle string expansion is less than the available gap between the fuel strings and supporting structures (Canada);

(b) Including the effects of temperature and channel creep, string length has to be less than available gap during normal operation (India);

(c) During normal operations and during AOOs, total axial length of fuel bundles in a pressure tube is limited within the length of each shield plug. In this evaluation, string expansion due to the thermal expansion is considered (Korea);

(d) Expanded length of the fuel string has to be less than available gap (Pakistan);

(e) Fuel string length has to be less than the distance between shield plugs (Romania).

Discussion

Failure due to excessive interaction loads along the fuel string is primarily a consideration of dimensional compatibility and is more of a concern during accident conditions than during normal operations. In the latter, it is usually addressed by providing sufficient length of the fuel cavity. String lengths are sometimes measured during operation (e.g. during fuelling). Design calculations would consider on-power dimensions of key mating components, including time dependent effects such as creep and irradiation growth.

3.1.4.2. Bundle geometry change

Element bowing or sagging and element diametral straining could affect heat transfer between the fuel element sheath and the coolant (in horizontal channel-type PHWRs), while the current approach is based on experimental measurements that had used reasonably straight tubes. Excessive deformations of the sheath would pose a question mark on the use of such heat transfer coefficients in accident analyses, or even for determining pre-accident conditions such as fission gas release during normal operation. Therefore, it is prudent to limit the extent of sheath deformations within reasonable limits.

Reported criteria

(a) For normal operating conditions, unirradiated bundles which pass the bent tube gauge test will not experience in-reactor difficulties during refuelling or discharge. Also, possible elastic-plastic deformation, element spacing and element-pressure tube spacing requirements will not affect bundle compatibility with interfacing systems (Canada);

(b) Provision for bundle axial and radial growth are provided in the channel assembly design (India);

(c) Geometry change of fuel bundle in the core is limited as described in criteria for fuel string expansion; for fuel elements; and for compatibility with the PHTS, pressure tube integrity and fuel handling system (Korea);

(d) Bundle geometry has to meet limits of as-build bundle specifications (Romania).

Discussion

Operational experience indicates that fuel bundles which pass through the bent tube gauge maintain sufficient dimensional stability during irradiation that dimensional compatibility with interfacing equipment is retained. Dimensional changes during irradiation can be covered through calculations that account for on-power dimensions and geometries of key mating components, including time dependent effects such as creep and irradiation growth.

3.1.4.3. Failure due to excessive interference with interfacing equipment

This mode of failure refers to in-service dimensions of the fuel bundle potentially interfering with the dimensions of interfacing equipment, through processes such as interlocking of spacers, creep, thermal expansion, etc.

Reported criteria

(a) Fuel geometric compatibility is ensured through adherence to fuel design parameters, bent tube gauge testing of as-fabricated bundles and limits during irradiation on pad wear (Canada);

(b) Bundles are checked for interlocking; interlocked bundles are not loaded into the reactor (India);

(c) Dimensional changes such as rod bowing or irradiation growth of fuel rods and fuel assemblies have to be limited to prevent fuel failures or a situation in which thermohydraulic limits are exceeded. Further, fuel bundle deformation shall be limited as required for compatibility with fuel handling system (Korea);

(d) Spacer interlocking is not allowed. Each bundle is checked for interlocked spacers and pass a gauge test before it is loaded in the core. Further, fuel bundle design must be compatible with operation of the fuelling machine. Bundle geometry has to satisfy the limits imposed by the technical specifications (Romania).

Discussion

Similar to failure due to excessive interaction loads along the fuel string (see Section 3.1.4.1), failure due to excessive interference with interfacing equipment is primarily a consideration of dimensional compatibility and is usually addressed by checking against the bent tube gauge. Calculations and/or measurements of on-power dimensions of key mating components are very complex and would need to include effects such as bowing, parallelogramming, sag, droop, and time dependent effects such as creep and irradiation growth.

3.1.4.4. Wear

Parts of fuel can wear due to certain mechanisms such as fretting wear and sliding wear. For example, wear can occur during the following situations: (a) when the bundle is exposed to crossflow; (b) when the bundle is slid into, or out of, the pressure tube; (c) during acoustic vibrations; and (d) during flow induced vibrations.

Reported criteria

(a) Limits are set to prevent excessive wear of the fuel bundle. A time limit is imposed for the bundle in crossflow; a limit on maximum spacer pad and bearing pads wear is also imposed (Canada);

(b) As acceptance criteria (China):

  (i) Fretting damage caused by coolant flow has not to reduce inter-element spacings (i.e. spacer height) below an acceptable level,

  (ii) During refuelling, a fuel bundle has to withstand the loads imposed by radial flows as the bundle passes through the end fitting. While in the liner tube hole regions, a bundle is subjected to very high fretting rates due to crossflow. The bundle needs not to be subjected to crossflow longer than necessary for refuelling of a channel (bundle exposure time in crossflow is about 10 minutes);

(c) Interlocking of bundles is checked and removed before loading into the core (India);

(d) During normal operations (Korea):

  (i) Fretting wear at contact pins on structural members should be limited,

  (ii) Excessive wear of bearing pads should be limited and a gap shall be maintained between the fuel rod and the pressure tube,

  (iii) Wear of spacer shall not exceed the minimum height of one spacer,

  (iv) To prevent fuel damage due to crossflow in end fitting, wear of spacer and bearing pads is limited,

  (v) Flow rate should be less than the upper limit, below which vibration of fuel bundles and wear of pressure tubes are acceptable. (Upper limit of flow rate is 30 kg/s for single phase flow and two phase flow.)

(e) As acceptance criteria (Romania):

  (i) Fuel bundle design must be able to withstand (for a short duration) the crossflow in the liner region and higher channel flows during fuelling,

  (ii) Interlocking of spacers is not allowed. Each bundle is checked for interlocking of spacers and has to pass through a gauge test before it is loaded in the core.

Discussion

Corners of excessively worn spacer pads can rub against the surface of an adjacent sheath and potentially damage it, for example when the bundle is exposed to crossflow. Due to on-power refuelling in horizontal channel-type PHWRs, bearing pads that support the bundle on the inside surface of pressure tube can be worn out.

Excessive wear of the bearing pads and/or spacer pads can potentially reduce the separation between adjacent sheaths or between a sheath and the pressure tube, altering the assumed heat transfer coefficient between the sheath and the coolant. In the limit, if a pad is fully worn off, a sheath can contact its neighbour (another sheath or a pressure tube) and create a local hot spot.

All fuel bundles in horizontal channel-type PHWRs are confirmed to be 'not interlocked' immediately prior to loading into the reactor. This eliminates the potential for a bundle becoming interlocked during transport being loaded into the reactor. During irradiation, wear between spacer pairs occurs; however, based on in-bay and post irradiation inspection of bundles, no concerns have been raised with respect to susceptibility of the bundles to interlocking. This is consistent with out-of-reactor endurance testing and qualification of several fuel bundle designs. As such, no design limits on in-reactor spacer wear have been considered to preclude fuel bundle interlocking.

3.1.4.5. Pressure tube wear and oxide thickness

Reported Criteria

(a) There are no limits on bearing pad wear in pressure tube. Pressure tube bearing pad wear are assessed as flaws. Flaws are addressed and limited when fitness-for-service cannot be demonstrated, either due to failing crack initiation criteria or plastic collapse. These limits are based on flaw dimensions and calculated stresses. They are dispositioned using the rules N285.4 [11] and N285.8 [12] of the Canadian Standards Association (Canada);

(b) Limit on wear and oxide thickness of pressure tube. Excessive wear of pressure tube has to be limited and a gap must be maintained between the fuel rod and the pressure tube (Korea);

(c) The integrity of the pressure tube under an interaction with fuel bundles is considered out of the scope for fuel design qualification (China).

Discussion

Pressure tubes can wear due to interactions with the fuel bundle, for example due to sliding of a fuel bundle through the channel, or due to fretting from vibrations of the fuel element/bundle (specific to horizontal channel-type PHWRs).

Sliding wear tests show wear in pressure tubes [43]. Nevertheless, to date such wear has been sufficiently small in operating horizontal channel-type PHWRs. The worn surfaces frequently exhibit higher oxidation compared to unworn surfaces.

Criteria for pressure tube wear have been reported in different formats by various States with PHWRs. Some have reported criteria exclusively for wear, others have combined criterion for wear with that of oxidation, and yet others have combined crevice corrosion with wear. Therefore, this compilation had to necessarily make arbitrary decisions about the format, content and placements of reported 'mixed' criteria.

3.1.4.6. Pressure tube crevice corrosion

Reported Criteria

(a) There are no design limits on crevice corrosion in pressure tubes. Pressure tube crevice corrosion marks are assessed as flaws. Flaws are addressed and limited when fitness-for-service cannot be demonstrated, either due to failing crack initiation criteria or plastic collapse. These limits are based on flaw dimensions and calculated stresses. They are dispositioned using the rules N285.4 [11] and N285.8 [12] of the Canadian Standards Association (Canada).

Discussion

Shallow marks due to crevice corrosion have indeed been observed in pressure tubes of horizontal channel-type operating PHWRs. In some cases, depth of crevice corrosion has increased with time.

Depending on the shape of the crevice corrosion, local stress concentration can be introduced in the pressure tube. In addition, loss of load bearing material would increase the overall stress in the vicinity of the corroded area.

Crevice corrosion to date has not led to safety of the pressure tubes being compromised in horizontal channel-type operating PHWRs.

3.2.ACCIDENT CONDITIONS

3.2.1. Technical basis

Physical barriers to be considered under accident conditions are identified in Appendix III. In this section, these barriers are assessed in relation to their safety functions.

3.2.1.1. Fuel matrix

The fuel matrix material used in the PHWRs is uranium dioxide (UO_2) in the form of sintered pellets. UO_2 was selected due to its excellent chemical stability and its high melting temperature.

Under normal operations, the polycrystalline UO_2 solid has close to 100% retention of the majority of (less volatile) fission products generated by fission [44]. The remaining fission products (e.g. noble gases and some volatiles) are gradually released to the pellet-to-sheath gap by athermal and thermally activated mechanisms and, consequently, their retention depends on the temperature evolution during the accident sequence. The effectiveness of this retention/delay can be reduced if the fuel matrix temperature reaches values close to its melting point and if the oxidation state of the fuel matrix changes from $UO_{2.00}$ to $UO_{2\pm x}$, where the affected properties are its thermal conductivity and the migration rate of certain fission products (e.g. noble gases and volatiles)[17]. The release characteristics of $UO_{2.00}$ and $UO_{2\pm x}$ have been studied for the last 20 years in international laboratories. The relationship between oxidation state of the fuel matrix and its fission product retention characteristics are well known.

[17] Mechanical damage can also degrade this barrier.

After fuel-to-sheath gap closure, the fuel matrix is in contact with the sheath and closer to the in-core pressure boundary (the pressure tube). Thus, potential interactions among these barriers may be important to consider.

For example, during a large break LOCA in a horizontal channel-type PHWR, if the power increase following the increase in reactivity during the core voiding is large, the energy deposition in the fuel matrix can be fast (1 to 2 seconds), although not nearly as fast as in an LWR control rod ejection scenario (tenths of milliseconds). In principle, depending on the amount and rate of the energy insertion and fuel burnup, a large power pulse can result in the ejection, after sheath failure, of round particles from previously molten urania or small broken in fine pieces (particles) from the 'rim' region in high burnup fuel. This phenomenon is called 'fuel dispersal' (see Section 3.2.1.4). If the urania particles are at high temperature, the integrity of the pressure tube may be threatened.

Mechanical interaction between fuel pellets and sheath can occur due to thermal expansion of the pellet and fission product swelling. In certain cases, this interaction can affect sheath integrity. For certain values of slow energy insertion, it is also possible that the temperature of the fuel matrix can reach its melting temperature. If this occurs, the molten material can relocate inside the fuel pellets and contact the fuel sheath. As the melting temperature of the sheath is lower than the melting temperature of the fuel matrix, the sheath integrity may be lost. If there is a significant amount of molten material, similar threat could affect the pressure tube. Furthermore, the melting of the fuel matrix has a strong influence on the pellet expansion and on the migration properties of fission products.

In addition, other chemical interactions may occur, depending on the particular characteristics of the accident sequence. These may include interactions between fission products (e.g. iodine and caesium), UO_2 — Zircaloy (reduction of UO_2), and the previously mentioned UO_2 — O_2 (UO_{2+x} formation). The most important interactions with respect to the performance of UO_2 are the oxidation and reduction reactions. The first ones are affecting the UO_2 thermal conductivity, melting temperature and fission product release rate, and the second ones the production of several metallic and ceramic faces.

3.2.1.2. Fuel sheath

The fuel sheaths used in the PHWR reactors are made of Zircaloy-4. This zirconium alloy was selected on the basis of its low neutron cross-section, its mechanical properties, and its excellent water-corrosion behaviour. The fuel sheath retains fission products and provides support to the geometrical structure of the fuel bundle, and therefore assists in maintaining coolable bundle geometry in conjunction with the pressure tube[18].

Under normal operating conditions, the effectiveness of this barrier is close to 100% for the associated safety functions (see Appendix III). The sheath is impermeable to migration of the fission products stored in the fuel-to-sheath gap outside of the fuel pellets and its good thermal properties ensure sufficient heat removal from the fuel matrix.

Several mechanisms may challenge this barrier. Under significant axial forces or circumferential temperature differences, the fuel sheath may induce bowing in the fuel element. In certain cases, many fuel elements may bow in a consistent manner resulting in barrelling

[18] A coolable, fuel bundle geometry means that the channel decay heat can be removed from the fuel bundles in a channel, in the long term, by the emergency core cooling system, without further fuel damage, and without reliance on the moderator.

(increase of bundle mid-length diameter) or hour-glassing (decrease of bundle mid-length diameter). Furthermore, if the temperature of the sheath is sufficiently high, the sagging of fuel elements can result in contact between neighbouring elements or even the pressure tube and could induce the formation of hot spots. Also, if the heat removal conditions degrade in a small region of the sheath, significant local sheath strains can result. These deformations may reduce the effectiveness of the fuel sheath to ensure fuel coolability.

The effectiveness of the fuel sheath to retain fission products can be affected by several sheath failure mechanisms. These mechanisms can progress, resulting in sheath failures and allowing the release of some of the fission products stored in the fuel-to-sheath gap. The understanding of these sheath failure mechanisms is of sufficient level to facilitate prediction of sheath failures during accident sequences.

The oxidation of Zircaloy-4 is an exothermic reaction; the reaction rate depends highly on the sheath temperature, the oxide metallographic phase of the surface oxide layer, local heat removal and coolant composition. In certain accident scenarios, this reaction may represent an important heat source.

Fuel sheath behaviour during accident conditions, including elastic-plastic deformation and oxidation characteristics, has been extensively studied in laboratories of several States. The knowledge accumulated by this experimental work is extensive and the sheath fission product retention characteristics and deformation behaviour are well understood.

It is conceivable that, during some accident sequences, the fuel sheath temperature could exceed its melting temperature. If this occurs, the molten material can relocate and contact the pressure tube, producing a hot spot. If conditions of differential pressure and deformation state of the pressure tube are in the sensitive ranges, the pressure tube could fail and affect the heat removal conditions of the bundles in the failed pressure tube. Similar but less severe conditions may occur when a fuel element contacts the pressure tube.

3.2.1.3. Primary heat transport system

The safety functions performed by the PHTS are to ensure that the removal of the heat generated by fission and fission product decay is not impeded during the short term and long term phases of an accident sequence. In addressing this barrier, only failure mechanisms influenced by the interactions with the fuel are included in this publication; consequently, pressure tube failure mechanisms independent of fuel behaviour and the performance of the calandria tube are excluded.

The pressure tubes used in horizontal channel-type PHWRs are made of an alloy of zirconium-niobium. This alloy (Zr-2.5wt% Nb) was selected on the basis of its low neutron cross-section, its good mechanical properties at high neutron fluence (better than Zircaloy), and its excellent water-corrosion behaviour. The pressure tubes convey the circulating coolant along strings of fuel bundles, enabling the removal of the heat generated by fission and fission products decay. Under normal operating conditions, the effectiveness of the pressure tube to perform the identified safety function is very high (~100%). Even in the case of an impaired PHTS, the remaining coolant will be able to remove at least some of the generated heat if the pressure tubes and/or calandria tubes are intact.

Degradation mechanisms that could be active during normal operation conditions, such as delayed hydride cracking, irradiation enhanced deformation and changes of the pressure tube material properties, may affect pressure tube failure susceptibility during certain accident sequences. Because of strict operating limits imposed on the pressure tube by operating procedures, these mechanisms have a minor impact.

During accident sequences, if the pressure tube temperature increases to the Zr-2.5wt% Nb transformation temperature and the differential pressure across its thickness is sufficient, the pressure tube will deform radially (ballooning), and may contact the calandria tube (in a horizontal channel-type PHWR). This contact significantly increases the heat transfer from the pressure tube to the moderator, inhibiting further degradation of the pressure tube. However, before contact is established, the pressure tube is susceptible to rupture if, for example, local contact between fuel element and pressure tube occurs (i.e. a hot spot) while the internal pressure is high. Furthermore, in situations where axial forces are applied during an accident (compressed fuel string expansion), fuel element/pressure tube contact can threaten the integrity of the pressure tube.

If the pressure tube ruptures, coolant leakage may occur. This can affect the flow of coolant along the channel and affect the fuel cooling capability within the channel. The pressure tube can also sag or balloon into contact with the calandria tube, thus providing an additional path for heat to be removed by the moderator.

The effectiveness of the pressure tube to perform the indicated safety functions can be affected by several degradation mechanisms. These mechanisms are well understood, and the knowledge level is sufficient for the prediction of pressure tube failure during accidents. It is noted that, even if a pressure tube fails, an intact surrounding calandria tube may still perform these safety functions.

During the early stages of certain events (e.g. large break LOCA), the fuel bundle string could accelerate by flow reversal and acquire kinetic energy along the channel. Above a certain value of the kinetic energy, the impact of the fuel string with some of the structures of the end fitting can damage the pressure tube rolled joint.

The pressure tube behaviour during accident conditions, including elastic-plastic deformation and oxidation characteristics, has been extensively studied in Canadian laboratories. The knowledge accumulated by this experimental work is extensive and the pressure tube deformation behaviour is well understood.

3.2.1.4. Barrier failure mechanisms

Failure of a barrier is a safety concern; the extent of concern depends on its consequences [45]. Of particular interest for this publication are barrier failures leading to radiological releases to the public or the environment beyond authorized limits and significant degradation of heat removal.

In determining barrier failure mechanisms, it is important to distinguish between a failure mechanism key process and other influential processes. For example, the key process on the melting of the fuel matrix (stoichiometric UO_2) is the loss of long range crystallographic order due to lattice oscillation when its melting temperature is reached. On the other hand, influential or enabling processes that contribute to this failure mechanism are changes in thermal conductivity due to pore formation, evolution of the heat capacity during temperature increases, or imbalance between heat generation and heat removal from the fuel matrix. This publication focuses on the key process of barrier failures.

As mentioned in Section 3.2.1.3, only pressure tube failures trigged by the interaction between fuel and the pressure tube are covered by the scope of this publication. Other failure mechanisms (e.g. deuteriding/hydriding, Inconel interaction) will not be discussed.

Each barrier failure mechanism has one or more key parameter(s) related to the relevant mechanism that uniquely describes the failure process. Fuel acceptance criteria can be established by selecting specific value(s) of these parameters that, if they are complied with during an accident sequence, barrier failures would be prevented.

A brief description of these barrier failure mechanisms is provided below.

Fuel matrix

- 'Fuel dispersal' is the ejection (spread) of fine fuel particles and/or molten materials through a rupture or opening in the cladding which may occur at high energy deposition. There are key differences between the fuel dispersal processes occurring at low and high burnup:

    - At low burnup, significant amounts of molten urania can be ejected from the failed rod and the evidence of prior melting is apparent from the spherical shape of the solidified fuel "particles" collected after the test [46–48];

    - At high burnup (greater than 42 GW·d/tU), the rim region of the pellet (thickness of ~100 μm) is characterized by defect clusters of extremely tangled dislocations, loss of as-fabricated grain boundary, fine recrystallized grains of 50–200 nm size and coarsened pores surrounded by recrystallized grains. Following a significant energy deposition and sheath failure, the fuel in the rim region could be ejected and dispersed as fine fuel particles that do not present the recognizable spherical shape of prior melted fuel particles [46–50];

- 'Fuel melting' is the loss of long range crystallographic order of the UO_2 or $UO_{2\pm x}$. The onset of fuel melting is characterized by the appearance of the first liquid on the fuel matrix (solidus line).

Fuel sheath

- 'High plastic strain rate' is the fast deformation of the sheath. After the onset of non-uniform strain (very large strain rates leading to sheath ballooning), if the driving force can be maintained, the sheath will rupture;

- 'Athermal strain' is a low ductility sheath failure mechanism. Under certain conditions, sheath failures can occur at relatively low strains;

- 'Beryllium braze penetration' is a beryllium induced sheath failure (specific to CANDU fuel bundles). Fuel sheaths with beryllium-brazed appendages exhibit failures in the braze alloy region;

- 'Embrittlement' due to oxidation/deuteriding (hydriding) is the deterioration of material properties due to increases in oxygen/deuterium concentration;

- 'Mechanical damage' of fuel is the damage of fuel bundle(s) by impact with other fuel channel structures;

- ‘Sheath melting’ is the appearance of the first liquid (solidus line);

- ‘Plastic strain’ is a permanent sheath deformation that can result in failure [51].

Pressure tube in a horizontal channel-type PHWR

- ‘Local plastic strain due to contact with the molten material’ is the local deformation due to high contact temperature with molten material;

- ‘Local plastic strain due to fuel element contact’ is a local deformation on the contact area. Fuel element deformation can result, for example, from differential axial thermal expansion across the fuel bundle cross-section, very low flexural rigidity due to high sheath temperature, or mechanical loading bending moments (e.g. bundle impact, constrained axial expansion);

- ‘Rolled joint failure due to fuel string impact’ is the damage of the rolled joint by mechanical impact with the fuel string.

3.2.1.5. Failure mechanisms and fuel acceptance criteria

Each of the barrier failure mechanisms listed above can be described by a set of parameters that can comprehensively describe the degradation and failure phenomenon. From this set, it is always possible to select one or a few parameters related to the specific mechanism that regulates the progression of the failure process. If limits imposed on these key parameters are complied with during an accident sequence, the corresponding barrier failures would be prevented. These parameters and their limits are known as fuel acceptance criteria (see Sections 1 and 2).

The selection of parameters to implement specific fuel acceptance criteria may not be unique, and different States or even different utilities in the same State can select different sets to implement a criterion that prevent barrier failure. Furthermore, it is plausible that limits in some influencing parameters can also be effective in precluding barrier failures in certain accident sequences. However, an effective selection of acceptance criteria depends on how well the failure mechanism is known, the conservatism that the State (or utility) considers necessary to comply with regulatory requirements, operational practices, and new findings from research and development activities.

As stated in Section 1, the IAEA has initiated a process by which States with PHWRs can share their fuel acceptance criteria used in order to ensure safe operation of the plants. As the initial step, the IAEA requested the interested States to complete a survey on the use of fuel acceptance criteria.

In the following Sections 3.2.2 to 3.2.4, the criteria for each of the listed barrier failure mechanisms reported by the participating States are summarized, and technical information considered important to define a set of fuel failure criteria for PHWRs is discussed.

3.2.2. Fuel matrix

3.2.2.1. Dispersal

Reported criteria

(a) Avoidance of fuel dispersal if the total enthalpy in the hottest fuel element is less than 965 kJ/kg UO_2 (or 840 kJ/kg UO_2 for single units) or no fuel melting (Canada, China, India, Romania);

(b) Total stored energy has to be <756 kJ/kg UO_2 (for vertical channel-type PHWRs, Argentina);

(c) Fuel pellet radial average enthalpy of the hottest fuel element has not to exceed 837 kJ/kg UO_2 (200 cal/g) (India);

(d) Stored bundle enthalpy shall be <838 kJ/kg UO_2 (Pakistan).

Discussion

Fuel dispersal can occur during reactivity initiated accident (RIA) scenarios where rapid and significant power excursions are possible. To avoid the loss of coolable geometry and the generation of coolant pressure pulses, peak fuel enthalpy criteria are used as limits for RIA. Reference [48] describes low burnup fuel dispersal (for burnups lower than 1000 MW·h/kgU or 42 GW·d/tU) [48, 49]) as process involving the ejection of previously molten urania particles and significant amount of molten material outside of the failed fuel element sheaths that can challenge pressure tube integrity. Reference [48] also describes high burnup fuel dispersal as a phenomenon that results in the ejections of fine particles from the ‘rim’ region. It is evident that high burnup fuel dispersal is not an active mechanism for typical PHWR natural uranium fuel discharge burnups.

Reference [1] assessed the current U.S. NRC Regulatory Guide 1.77 that states a value of 1170 kJ/kg UO_2 (280 cal/g) to maintain minimum core damage and ensure coolable geometry. After a brief analysis of the newer test results, the authors suggested that this value for the dispersal limit is a bit high for low burnup fuel and should be considered to decrease it to about 962 kJ/kg (230 cal/g).

The most limiting event of this type, in horizontal channel-type PHWRs, is the RIA scenario that develops during a large break LOCA. Similar scenarios are assessed for LWRs (e.g. control road ejection); however, the generated power pulse in PHWRs would subject the fuel to a longer duration, lower amplitude power excursion than in an LWR. In the latter, RIAs are characterized by power pulses that occur over a timespan of <100 ms compared to about 2 s in a horizontal channel-type PHWR.

The integrated energy generated in the fuel during the power pulse is a key parameter that governs the fuel response [46]. Representative tests [47–49] have been performed on different types of fuel in various research reactors over the past three decades [52–70]. For LWR fuel, important results were obtained from Special Power Excursion Reactor Test, Capsule Driver Core Facility, Power Burst Facility, Nuclear Safety Research Reactor and French CABRI reactor. In the case of horizontal channel-type PHWR fuel; only two sets were performed with CANDU fuels on the Power Burst Facility and TRIGA-ACPR (Annular Core Pulse Reactor). Finally, for the VVER fuel type, experiments were performed in the Impulse Graphite Reactor.

Reference [49] summarizes the CANDU fuel behaviour characteristics for events with significant energy deposition based on the experimental data mentioned above:

- Threshold values of fuel enthalpy for clad melting, low burnup fuel dispersion failures are dependent upon the power pulse width and increase for longer pulse durations due to the increased heat removed by the coolant;

- Clad melt failure occurs at similar, but slightly higher enthalpy values than oxygen embrittlement. Threshold values of fuel enthalpy for clad are not strongly dependent upon the power pulse width;

- Threshold values of fuel enthalpy for oxygen embrittlement failure tend to increase with increasing pulse width due to the larger sensitivity of fuel sheath temperature to energy removed by the coolant;

- Pellet-cladding mechanical interaction (PCMI) failure has a strong sensitivity to pulse width and the threshold fuel enthalpy for failure increases rapidly with increasing half height pulse width. Additionally, for this mode of failure very small gaps between fuel pellet and sheath are necessary. It is for this reason that PCMI failures have been observed in RIA tests with short pulse widths (4 ms) and irradiated fuel, while unirradiated fuel tends not to exhibit this failure mode. PCMI is an unlikely failure mode for CANDU fuel because of the long power pulse (~2 s);

- Low burnup fuel dispersion for fresh and irradiated rods (burnup less than 1000 MW·h/kgU) with fuel sheaths that maintain ductility (i.e. sheaths without the excessive corrosion observed on LWR high burnup fuel cladding) is associated with gross melting of the fuel.

As indicated above, RIA tests on fresh and irradiated fuel with burnup less than 1000 MW·h/kgU show that fuel is associated with ejection of molten material or previously molten urania from the fuel. References [48, 49] show that low burnup fuel dispersal with ductile cladding is associated with melting at the periphery of the fuel pellet which is strongly influenced by the radial fission power distribution (enrichment). In essence, at such low burnups fuel melting occurs prior to fuel dispersal. Furthermore, based on the significant wider power pulse (1 to 2 seconds) estimated during a large break LOCA in a CANDU reactor, the adoption of enthalpy thresholds derived from experimental results from tests with very short power excursions (two orders of magnitude shorter) are very conservative. At low burnups, prevention of fuel melting is an effective criterion for preventing fuel dispersal.

No experimental results using quantified hyper-stoichiometric fuel ($UO_{2\pm x}$) have been reported. Because the property changes associated with this type of fuel, some impact on fuel dispersal is possible. However, because the stoichiometric deviations in failed fuel prior to the RIA event at a CANDU reactor are local to the failure location, their impact is likely not significant. Current approach is that the RIA threshold is not affected by the presence of failed fuel. RIA experiments with the typical stoichiometric deviations and distribution observed on irradiated failed fuel could support the current approach.

Participating States reported different energy deposition limits as fuel acceptance criteria based on limiting the fuel total enthalpy (965 to 756 kJ/kg UO_2) and/or avoidance of fuel melting. Enthalpy values were derived from experiments or calculated, under adiabatic conditions, to reach fuel melting. Tests with sheath thicknesses similar to the use for horizontal channel-type

PHWRs may be better suited for selecting a fuel acceptance criterion for this failure mode [47–49]. The use of a temperature limit is based on the experimental evidence that low burnup fuel dispersal occurs only after melting of the fuel matrix.

3.2.2.2. Fuel melting

Reported criteria

(a) Melting temperature of UO_2 (approximately 3073 K) for 90% of the pellet cross-section at the hot spot (for vertical channel-type PHWRs, Argentina);

(b) Centreline pellet temperature less than stoichiometric UO_2 melting temperature (3120 K ± 30 K) (Canada);

(c) Centreline temperature less than 3113 K with correction for burnup (China);

(d) Fuel melting is considered at temperature of 2840ºC(India);

(e) Centreline temperature less than 3078 K with correction for burnup (Korea);

(f) Fuel central temperature below UO_2 melting point during fuel irradiation (Pakistan);

(g) Centreline temperature limit of 3038.15 K (2765ºC) (Romania).

Discussion

The onset of fuel melting is characterized by the appearance of the first liquid on the fuel matrix (solidus line). Stoichiometric uranium dioxide (UO_2) melts at 3120 K ± 30 K [26]. Upon melting, the formed liquid has a volume expansion of 10.4%.

Stoichiometric UO_2 melts congruently at a unique temperature [71], while hypo- or hyper-stoichiometric UO_2 does not and has separate solidus and liquidus temperatures which are lower than the congruent melting point of $UO_{2.00}$. In [72], it has been determined the solidus and liquidus lines for UO_{2+x} experimentally. In addition, concentration of fission and activation products can also affect these temperatures.

The fission and activation products generated in the UO_2 as burnup increases can be grouped into three broad classes:

- Insoluble fission products – These fission products, insoluble in either solid or liquid UO_2, do not significantly interact with the UO_2 and their impact with the melting point is non-observable. Examples of these fission products are Kr and Xe (inert gases) as well as transition metals such as Mo, Rh and Ru, as metallic precipitates which do not interact with the fuel matrix;

- Soluble fission products – Fission products soluble in the liquid phase (e.g. Sr, Ba) may be able to form eutectic alloys. Assessments for PHWRs conditions estimate a decrease in the UO_2 melting temperature of ~12 K at 700 MW·h/kgU. Accordingly, these eutectic-forming fission products have a very small impact on the fuel melting temperature;

- Other fission products – Actinide oxides, rare earth oxides, and Zr. The most important of these elements in PHWRs, based on the amount generated, is Pu. This dependency was reported by Carbajo et al. [73]:

$$T_L(y) = 3120.0 - 388.1y - 30.4y^2 \quad (2)$$

$$T_S(y) = 3120.0 - 655.3y + 336.4y^2 - 99.9y^3 \quad (3)$$

where $T_L(y)$ and $T_S(y)$ are the liquidus and solidus temperatures, respectively (in Kelvin), and y is the mole fraction of PuO_2. For Pu generated in a fuel with burnup for CANDUs, a decrease of <5 K in the melting temperature will result at a burnup of 700 MW·h/kgU [74].

In summary, fission and activation products generated during irradiation can decrease the melting temperatures by <17 K for a burnup of 700 MW·h/kgU. At typical PHWR fuel discharge burnups, the estimated decrease is <3 K.

After sheath failure, the steam can react with the fuel pellets, increasing its stoichiometry. In fact, phase diagrams of the uranium-oxygen system [72, 75] provide the solidus and liquidus lines for hyper-stoichiometric fuel. The melting temperature decreases when stoichiometry increases, about 200 K for $UO_{2.10}$, which is more significant than that resulting from fission and activation products [71, 72, 76].

In defective fuel elements with oxidized fuel and centreline melting, the centreline melting is “a self-regulating phenomenon in that the formation of a liquid reduces the potential for more liquid to form due to the non-congruent phase change” [77]. In [77], it is also clarified that “specifically, with the difference in oxygen concentration between the solidus and liquidus lines on the O-U binary phase diagram, there is an enrichment of oxygen in the liquid so that the remaining solid is reduced in oxygen content that raises the local melting temperature and increases the thermal conductivity of the solid”. However, the urania liquid volume expansion could result in the liquid relocating out of the centreline or through the cracks of the solid annulus. If liquid fuel attempts to move out of the molten fuel region, which is at the melting temperature, then the liquid fuel would solidify quickly [46]. The results of in-reactor experiments indicate that significant amounts of molten UO_2 can form and still be retained within the central region of the fuel [78–82]. Fuel melting does not necessarily result in failure of the fuel cladding [1]; if melting is limited, the geometry of the fuel remains intact and, therefore, coolable.

All participating States have a safety criterion for preventing $UO_{2.00}$ melting during accidents. The stoichiometric UO_2 melting temperature and uncertainty values were reported in IAEA-TECDOC-1496 [26]. Some States have implemented corrections as function of fuel burnup (fission and/or activation products) or use conservative criteria below the reported value.

3.2.3. Fuel sheath

3.2.3.1. Strain rate

Reported criteria

(a) Limits on the strain rate are imposed ($<10\ s^{-1}$) to preclude sheath failure (Canada, China, Korea, Romania);

(b) Limits are imposed to sheath strain rather than strain rate (for vertical channel-type PHWRs, Argentina, India).

Discussion

After the onset of non-uniform strain, if the driving force can be maintained, the sheath will rupture. The strain rates associated with the mechanisms could be very large. The data from experiment shows that, even under well-controlled conditions of a laboratory ballooning experiment, the measurements of strain at failure, under nominally identical conditions, show considerable scatter. This is due to the high sensitivity of the failure strain to both external conditions and small variations in the as-manufactured microstructure of the material. Most importantly, due to the very rapid nature of the ballooning mechanism, it is not possible to accurately measure the strain rate associated with the later stages of ballooning.

The choice of 10 s^{-1} strain rate is somewhat arbitrary and is based on considerations of numerical stability of fuel simulation software, as strain rates in excess of this value would require unsuitably small time steps to ensure convergence of the sheath strain calculation. In addition, the concept of the 'limit of uniform strain' is also employed in one dimensional simulation codes that cannot model such phenomenon as localized sheath ballooning. Even two dimensional codes that could more accurately model the ballooning phenomenon would have to be validated against experimental data that are extremely difficult to obtain for the site at which rupture occurs. However, since rupture is a local event governed by local non-uniformities (e.g. temperature, material properties, geometry), variations in late stage rupture have negligible impact on overall strain to failure or failure timing.

It is considered that this criterion is redundant since the uniform hoop strain criterion (see Section 3.2.3.7) ensures sheath integrity.

3.2.3.2. Athermal (low ductility) strain for horizontal channel-type PHWRs

Reported criteria

(a) Low ductility failure is avoided if athermal sheath plastic strain is below 0.4% before 95% of the sheath material has recrystallized [<773 K] (Canada);

(b) For temperatures below 700 K, sheath plastic strain of <0.4% (Korea).

Discussion

Several creep mechanisms (e.g. athermal glide, dislocation creep, grain boundary sliding) occur in Zircaloy [83, 84]. The dominant mechanism depends upon the internal stress level from work hardening during manufacture, sheath stress level, temperature, and irradiation hardening.

For most accident transients, the temperature of the sheath rises, and annealing restores the ductility before significant loads are imposed on the sheath. However, if the fuel experiences a rapid temperature increase, the fuel pellet may expand thermally before the temperature of the sheath has risen sufficiently for annealing to occur. In this case, the sheath may fail at an average strain substantially below the generic overstrain criteria of 5% [85] (see Section 3.2.3.7).

Experiments performed in Canada's National Research Universal (NRU) reactor studied the effects of heat treatment by testing samples of Zircaloy-2, Zircaloy-4, and Zr-2.5wt% Nb alloy in the as-received, α annealed and β heat-treated states subjected to axial tension, transverse ring tension and biaxial burst tests [86, 87]. The results showed considerable variation with the amount of cold work, and the circumferential strain at bursting varied between 0.4% and 29%.

The value of 0.4% is used in some PHWR States as a conservative lower limit for this failure mechanism. Expert analysis of the existing data suggests that this value may be conservative and that a more appropriate value would be 3% plastic strain; biaxial tests on irradiated sheath would be needed to confirm this value.

Currently, States with operating PHWRs use two consistent forms of this fuel acceptance criterion by assuming sheath failure if the athermal sheath strain exceeds 0.4% before 95% of the sheath material has recrystallized or the equivalent 0.4% strain is exceeded for temperatures below 700 K.

3.2.3.3. Beryllium braze penetration in fuel bundles for horizontal-channel type PHWRs

Reported criteria

(a) A life fraction and failure probability model with correction factors for oxide thickness, sheath thickness and appendage arrangements; probability of beryllium assisted cracking less than 1% (Canada, China, Korea, Romania);

(b) There are no criteria of beryllium braze penetration in the Indian PHWR fuel bundles and Argentina's Atucha fuel assembly because of absence of beryllium in fuel fabrication (Argentina, India);

(c) Criterion not necessary because of low fuel temperatures (Pakistan).

Discussion

Bearing and spacer pads are beryllium-brazed to the fuel bundle elements to separate neighbouring elements both from one another and from interfacing systems such as fuel handling systems and fuel channels. In some States, operating PHWRs did not use this technique (e.g. Argentina, India).

During manufacturing, these pads are vacuum coated with beryllium metal and brazed to the fuel sheaths using an induction heating process. During heating, when the target temperature is reached, the beryllium metal deposited on the pads rapidly reacts with the available zirconium to form a braze alloy forming a eutectic with a melting temperature of 1243 K [88].

During postulated accident conditions, where the fuel sheath can be subject to elevated temperatures and high hoop stress, fuel sheaths with beryllium-brazed appendages may fail by this mechanism in the braze alloy region. These failures are attributed to a liquid metal embrittlement mechanism. The result is, once cracks incubate in the surface of the fuel sheath, the presence of beryllium, and stress can subsequently cause the crack to propagate through the sheath.

Initially, the experimental work on this mechanism [88] was conducted on Zircaloy-4 specimens subjected to various temperatures and internal pressures in inert environments and correlations developed for the incubation and crack propagation times associated with beryllium assisted crack penetration on the basis of data from isothermal and isobaric tests. The behaviour of the mechanism was observed to change abruptly at the braze alloy melting temperature. Separate correlations were developed for each process above and below 1243 K for sheath hoop stresses ranging from 1 MPa to 50 MPa. For practical use, these correlations were converted into a failure probability coupled with a life fraction approach to account for pressure and temperature transients. These correlations provide an estimate of the probability of failure during the course of the accident. Adjustments to the failure correlations are necessary for sheaths of different thickness and appendage arrangements than those used in the tests. These correlations were extended to oxidizing environments, and a correction factor was included to the life fraction methodology as a function of oxide thickness [89, 90].

The ranges for sheath hoop stresses used in these experiments cover the range of operating conditions and upset conditions. This mechanism has not been observed to occur for sheath temperatures below 1023 K and stresses up to 30 MPa maintained up to 19 minutes (liquid metal embrittlement) [80, 88].

All States that include beryllium in their fuel fabrication process use the methodology described above for assessing the probability of this failure mechanism and a very similar safety criterion. No information was provided by States using different fabrication methods on fuel acceptance criteria applied to prevent sheath failures under bearing/spacer pads.

3.2.3.4. Embrittlement due to oxidation/hydriding

Reported criteria

(a) Sawatzky embrittlement criterion where the sheath could fail on rewet if the oxygen concentration within the sheath exceeds 0.7 wt% over more than half the sheath thickness (Canada, China, Romania);

(b) U.S. criterion: Appendix K to 10CFR50.46 US criterion (for vertical channel-type PHWRs, Argentina);

(c) Limits based on irradiation experience -5 µm outside clad oxide thickness (India);

(d) Prevention of excessive sheath oxidation and hydrogen content <0.8 mg (Korea);

(e) Oxygen concentration <0.05 ppm and D_2 between 0 and 20 cm^3 D_2/liter D_2O (Pakistan).

Discussion

Oxidation and hydriding are two phenomena that can affect fuel integrity under accident conditions. The oxidation process degrades the sheath material properties, decreases the load bearing cross-section, and could also affect the heat transfer from fuel elements to coolant. Furthermore, hydrogen pick-up and hydride formation could lead to the reduction of sheath ductility and, under certain conditions, subsequent loss of integrity during the quenching phase of events.

These phenomena have been extensively studied and have shown that their impact on sheath performance is more significant for longer exposures to oxidizing environments (air, steam). This is the case for LWRs where the fuel is irradiated for several years, accumulating burnups greater than 1000 MW·h/kgU. For horizontal channel-type PHWRs, where the discharge burnups are in the order of 250 MW·h/kgU and average exposure periods are shorter than a year, the impact of the above phenomena on sheath performance is less severe. Consequently, during the PHWR pre-accident phase, the sheath has significant less concentration of oxygen and hydrogen/deuterium than their LWRs counterpart. Also, fabrication limits on oxygen/hydrogen content further limits their concentration prior to an event.

However, during accidents, fuel temperature increases and exposure to environments with high concentrations of steam or air could increase the amount of sheath oxidation and heat generation by this reaction in a relative short period of time.

The Zircaloy oxidation will release heat and generate hydrogen or deuterium gas by the following chemical reactions:

- $Zr + O_2 \rightarrow ZrO_2$ + heat; (4)
- $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$ + heat; or (5)
- $Zr + 2D_2O \rightarrow ZrO_2 + 2D_2$ + heat. (6)

With an unlimited amount of heavy water vapour, the sheathing can be fully oxidized to ZrO_2 during an accident. On the other hand, as a significant amount of deuterium gas can be generated with the large mass of zirconium in the PHWRs fuel channels, the gas phase can become depleted in heavy water vapour in the downstream locations of the fuel channels. In this case, sheath may not be completely oxidized. The presence of ZrO_2 on the sheaths has a very important effect of inhibiting or delaying the uptake of hydrogen by the metal.

The rate of oxide formation and diffusion of oxygen into the metallic sheath is a strong function of temperature. As long as the temperature remains high (>1073 K) and there is an adequate supply of oxygen, the oxide and oxygen stabilized α layer will grow at the expense of the β layer.

The oxidized sheathing has a complex morphology. For instance, for temperatures above the α to β transformation, two different contiguous metallic phases (α and β) of zirconium can co-exist for the partially oxidized Zircaloy sheathing, in addition to the zirconia layer. Since the mechanical properties of these layers are strongly influenced by oxygen distribution, accurate predictions of the layer thicknesses, oxygen profile, and reaction rates are needed. Simple parabolic rate kinetics can be used to estimate layer thicknesses for 'thin' layers in comparison with the initial sheath thickness and unlimited steam supply. However, a diffusion-based calculation is required with the presence of thick layers and oxygen depleted environments. Such more detailed treatment is needed to accurately predict sheath deformation behaviour and failure. For oxidation in environments containing air, the precipitation of zirconium nitride (ZrN) in the oxide phase can alter the morphology and some properties of the oxide layer [91].

As indicated above, the steam oxidation reaction releases deuterium (hydrogen), part of which diffuse into the metal phases and cause embrittlement of the sheath. The reacted oxygen stabilizes α-Zircaloy while diffused hydrogen stabilizes β-Zircaloy [92]. During the cooling phase of an event, the β-phase transforms into the α-phase and the dissolved hydrogen can precipitate forming hydrides. Fuel element failure is associated with rapid quench (750 K – 870 K) from the film boiling regime when there is insufficient remaining metallic substrate to withstand the thermal shock [93].

Above 1073 K, the oxidized sheathing is sufficiently ductile to prevent such failure. However, at low temperatures, the embrittled sheath may not have sufficient ductility to withstand thermally induced stresses during rewet. Experiments conducted by Sawatzky that were used to assess the ultimate tensile strength of Zircaloy-2 samples with varying degrees of oxidation revealed that the majority of the stress in the sheath is carried by the prior β phase following the temperature transient [94]. Below 1073 K, the ultimate tensile strength of prior β-Zircaloy with less than or equal to 0.7 wt% oxygen is at least twice that of the as-received Zircaloy. This observation led to the Sawatzky oxygen embrittlement criterion where the sheath could fail on rewet if the oxygen concentration within the sheath exceeds 0.7 wt% over more than half the sheath thickness. Earlier work presented in [95] and in [96] supported the development of the Sawatzky criterion. Later experiments [97] also confirmed this criterion.

This criterion is also consistent with international practice. Several criteria for preventing fuel failure due to oxygen and hydrogen embrittlement with water quench by the emergency core cooling system, have been proposed [98]. For example, the U.S. criterion is as follows:

- The calculated total oxidation of the cladding using the Baker and Just oxidation model [99], shall nowhere exceed 0.17 times the total cladding thickness before oxidation;
- The maximum fuel cladding temperature shall not exceed 1477 K.

Since the oxide thickness formed during a transient is not directly correlated to embrittlement caused by diffusion into the β phase of Zircaloy, the Sawatzky criterion is probably more representative [100] and, thus, the U.S. is revisiting their criteria [101].

It is important to note that fuel failure by this mechanism does not necessarily imply loss of coolable geometry or loss of heat transfer area [92].

The fuel acceptance criterion associated with this failure mechanism is not uniform in the States with operating PHWRs. Canada, China and Romania have implemented the Sawatzky criterion, Argentina the U.S. criterion (self-standing cladding) and India, Korea and Pakistan use other criteria based on experimental data on oxygen/hydrogen concentrations.

3.2.3.5. Mechanical damage

Reported criteria

No information on this fuel failure criterion was obtained from the PHWR States. (This criterion is not applicable to vertical channel-type PHWRs, e.g. Atucha-1 and Atucha-2.)

Discussion

In a horizontal channel-type PHWR during the blowdown phase of some events (i.e. large break LOCA), the coolant flow could reverse and the fuel string (composed of 12 or 13 bundles) may be displaced along the channel and finally impact against the channel-end structures (i.e. shield plug). The potential for bundle or channel damage depends on the bundle kinetic energy at impact. Until terminated by impact with the shield plug, the fuel string accelerates quickly for several milliseconds and then the velocity approaches an asymptotic value [102].

During impact, the fuel bundles could deform plastically and be reconfigured, resulting in an increase in resistance to coolant flow. A consequence is potential flow blockage in the shield plugs or end fitting following impact. If sufficiently severe, the resulting flow blockage could lead to early melting of the fuel or sheath, which could lead to fuel channel failure. Furthermore, as several fuel strings could impact almost simultaneously their respective shield plugs, the occurrence of fuel string impact with the shield plug in many channels raises concerns for the calandria pressure boundary. The channels with minimum outlet feeder resistance, largest inlet feeder diameter, and lower inlet header temperature give the largest impact velocities [102].

An extensive test programme involving six well-characterized test series was carried out [103]. In particular, the high-strain-rate tests were designed to demonstrate the acceptability of alternate fuel management solutions. For example, concerns are pertinent to the 12-bundle concept and fuelling-with-flow concepts. For impact energies greater than 11.31 kJ (with a 0.24 m/s velocity standard error), fuel damage (broken end plates, dislodged fuel elements, fuel elements lodged in end fitting flow holes) could occur [103]. Possible concurrent failure of the rolled joint is addressed in Section 3.2.4.3.

No information on this fuel failure criterion was specifically requested from the PHWR States. If mechanical damage of a fuel bundle is a credible result of any postulated event, prevention of the degradation of related barriers may be necessary.

3.2.3.6. Sheath melting

Reported criteria

No information on this fuel failure criterion was specifically requested from the PHWR States.

For Argentina's Atucha-1 and Atucha-2 fuels, the sheath temperature should be below 1200ºC.

Discussion

If, during an accident, there is a mismatch between heat generation and heat removal from a fuel element, the sheath temperature will increase and could reach the solidus temperature of Zircaloy-4 and start to melt. The solidus line temperature monotonically increases with oxygen concentration [104]. However, an unoxidized fuel sheath will likely fail by overstraining before melting while an oxidized sheath will likely fail by cracking of the oxide and overstrain under the oxide cracks before melting is reached (see Section 3.2.3.7).

The presence of oxygen in the fuel sheath has the effect of increasing its melting temperature [104, 105]. The melting temperature of Zircaloy-4 is well known; as-received Zircaloy-4 melts at 2033 K, the oxygen stabilized α-phase at 2243 K and ZrO_2 at 2963 K.

It has been well known in the industry that the rate of oxidation of Zircaloy-4 sheath material depends on temperature and oxygen availability. Temperature affects the number of phases present in the sheath and the oxygen diffusion coefficients of all of them. The ZrO_2 protective surface layer has three possible phases: monoclinic, tetragonal and cubic. The monoclinic phase is stable below 1323 K [107], the tetragonal phase between 1273 K and 2553 K [106, 107], and the cubic phase between 1763 K and the ZrO_2 melting temperature of 2963 K [107]. The cubic phase has a significantly higher oxygen diffusion coefficient than the other two phases. When the temperature increases, the transition between the ZrO_2 phases is characterized by increases in oxidation rates and consequential heat generation, provided that sufficient oxygen is available. Otherwise, the reaction rate is controlled by the oxygen flux at the sheath surface.

Notable increases in the sheath oxidation rate are observed for temperatures above 1773 K (tetragonal → cubic) for unlimited oxygen supplies. The increase is gradual since both phases coexist for temperatures between 1763 K and 2553 K. However, if the heat removal from the sheath surface is impaired, the faster heat generation can trigger sheath temperature escalation.

The relatively larger masses of Zircaloy in LWR fuel produce a much larger inventory of melt available for relocation than is possible with fuel cladding in a horizontal channel-type PHWR. Since the thin-walled cladding is distributed horizontally, there is not enough Zircaloy mass to accumulate so that liquid/solid interfacial forces (i.e. as opposed to gravity) dominate the relocation patterns. Movement of melt is determined by the adhesive/cohesive properties of the melt. If the adhesion forces are greater than cohesion forces, the melt will spread on the adjacent solid surface, otherwise the melt will tend to run.

In the contact area between elements, the sheath oxide layer stops growing due to localized steam deficiency. The ZrO_2 thickness will become even thinner; as the heat up progresses the underlying metal depletes the oxygen content. Following any melting of the metal substrate, the ZrO_2 shell will be perforated at its thinnest cross-section. Capillary forces are also capable of moving the molten material into inter-element spaces. However, the surface area of the relocated melt will be smaller than that of the original uniformly distributed cladding; in fact, the reduction in Zircaloy surface area will limit the subsequent exothermic oxidation [108].

Molten sheath in contact with solid UO_2 may partially dissolve and liquefy the pellets at ~1000 K below UO_2 melting temperature [105]. The driving force for the reaction is diffusion of oxygen from the UO_2 into the sheathing. The kinetics of this process has been extensively studied in single-effect laboratory crucible experiments [109–114]. On melting of the cladding, the liquid metal contacts the solid UO_2 fuel and dissolution of the fuel begins. The endothermic reaction of UO_2 dissolution in U-Zr-O melts and the melting of α-Zr(O) sheathing is affected by the supply of heat. Dissolution continues until the melt is saturated in both oxygen and uranium. The fuel dissolution process has been studied in detail and it is shown that diffusion in the growing molten (Zr,U,O)-alloy phase is rapid, and the liquid phase concentration remains at saturation [113]. Reduction of the fuel by oxygen diffusion affects the amount of oxygen in the melt.

The (Zr,U,O)-alloy phase reacted with solid UO_2 to pick up additional uranium and oxygen. The UO_2 is consequently reduced to UO_{2-x}. When the UO_{2-x} is further depleted in oxygen, the UO_{2-x}/U solvus boundary of the U-O phase diagram is reached. At this point, liquid uranium forms in the UO_{2-x}. On cooling to room temperature, the UO_{2-x} and the liquid uranium form stoichiometric UO_2 and α-U along the UO_2 grain boundaries and in the UO_2 grains.

The amount of UO_2 dissolved by the cladding is a function of the cladding volume relative to the UO_2 volume. Consequently, the fraction of the fuel pellet that is dissolved by liquefaction is small [113]. The effect of a thick oxide layer will also reduce the extent of fuel liquefaction because less metal is available to dissolve uranium when a melt forms.

As all the processes discussed above can be prevented by avoiding sheath melting, it is likely that all States with PHWRs have a fuel acceptance criterion based on sheath temperature. However, it is recognized that temperature escalation could be fast for temperatures above the ZrO_2 cubic phase formation, and that the analysis tools may have difficulty in simulating the temperature increases within the accuracy required for severe cases. To simplify the analysis tool validation requirements, a sheath temperature limit value, based on the appearance of the oxide cubic phase [107], is a possible alternative.

3.2.3.7. Plastic strain

Reported criteria

(a) Long term sheath strain <2.5% and <1% during power ramps (for vertical channel-type PHWRs, Argentina);

(b) Uniform sheath hoop strain <5% (Canada);

(c) Uniform sheath hoop strain <5% for sheath temperature <1273 K, reduced to 2% at higher temperatures (China, Romania);

(d) Total uniform circumferential strain <1% (India);

(e) Criterion not necessary due to low power operation and fuel design (Pakistan).

Discussion

Plastic hoop strain is the permanent increase in the sheath circumference from its initial value. The driving force for deformation is the stress caused by a combination of external pressure, pellet-cladding mechanical interaction, and internal gas pressure.

Sheath failure due to plastic overstrain under inert conditions can arise when the fuel element internal gas pressure causes the sheath to strain [83]. When thin-walled sheaths are subject to increases in internal overpressure, they will uniformly strain along their length because work hardening ensures that the hoop strain of the fuel sheath remains approximately uniform. After a certain (critical) strain is reached and the overpressure is maintained, sheaths will undergo a rapid localized ballooning and, at a large enough strain, the sheaths will fail.

A fuel sheath failure in this way generally shows a large degree of sheath strain localized around the failure point. The strain at rupture is dependent on a number of external parameters: driving pressure, sheath temperature, sheath temperature ramp rate, the presence of dissolved oxygen in the sheath, α/β phase fractions, and both axial and circumferential temperature gradients. At high sheath stresses, prior to annealing, burnup also influences rupture.

The data from experiment shows that even under well-controlled conditions of a laboratory ballooning experiment, the measurements of strain at failure, under nominally identical conditions, show considerable scatter. This is due to the high sensitivity of the failure strain to both external conditions and small variations in the as-manufactured microstructure of the material. Most importantly, due to the very rapid nature of the ballooning mechanism, it is not possible to accurately measure the strain rate associated with the later stages of ballooning.

Therefore, a fuel failure criterion has been developed based on the concept of the limit of plastic instability rather than the sheath strain at failure. An analysis found in [85], based on data compiled in [115], showed that the minimum critical strain for this threshold is 7% ± 1.5% [85] and that the average sheath strain at the onset of the instability for fuel sheathing in CANDUs is temperature dependent but has a minimum value of 5%, corresponding to a sheath temperature of 650°C.

Ballooning and burst type failures do not lead to significant fuel dispersal [92] and fuel failure does not necessarily mean the loss of coolable geometry or loss of heat transfer area. However, a fuel bundle in a CANDU reactor has certain features that could potentially lead to flow blockage via coplanar blockage due to the production of “long balloons” by subchannel space restrictions [116]. This can only occur for strains larger than the 5% uniform strain value.

Sheath failure by a local plastic overstrain can occur in an oxidizing environment, the formation of oxide layers on the sheath surface and diffusion of oxygen into the metallic portion of the Zircaloy sheath can significantly strengthen the sheath and reduce its plastic creep rate [116].

The ZrO_2 layer is formed in compression but exhibits cracking due to the crystallographic mismatch between the tetragonal and monoclinic phases (breakaway) that coexist at temperatures below 1273 K. The tetragonal phase is stabilized by compression forces at the metal oxide interface below its phase transition temperature. Also, at temperatures above 1273 K, additional cracks may develop by tensile strain (e.g. 2% [117]). These cracks provide a path for oxygen to penetrate into the crack tip. Moreover, a high tensile strain at the oxide tip may cause the new oxide to propagate into the substrate. The crack propagation rate is a function of the hoop stress and the growth rate of the oxide. The strain rate of the oxygen stabilized α layer is strongly dependent on the oxygen profile [118] in the layer making it substantially more creep resistant. Oxide layers less than ~5 µm thick have a negligible impact on the high temperature creep of Zircaloy-4 sheaths [116].

Two criteria were developed for this failure mechanism under oxidizing conditions. One states that a limit of about 15% local strain under the deepest crack is a sufficient limit for prevention of this failure mechanism. The other reported criterion is 5% uniform strain at or below 1273 K, which is reduced to 2% at higher temperatures [119]. However, no documentation describing the technical basis for the later criterion is available.

Due to differences in fuel design, the PHWR States did not report a common fuel acceptance criterion on sheath plastic strain.

3.2.4. Pressure tube in a horizontal channel-type PHWR

The following Sections 3.2.4.1 to 3.2.4.3 are based on practices from Canada and other States that operate horizontal channel-type PHWRs. In China, the integrity of the pressure tube under an interaction with fuel bundles is considered out of the scope for fuel design qualifciation.

3.2.4.1. Local plastic strain due to contact with molten material

Reported criteria

No information on this fuel failure criterion was specifically requested from the PHWR States. Local plastic strain due to contact with molten material is not considered as a safety criterion in some States. This criterion is also not applicable to vertical channel-type PHWRs, e.g. Atucha-1 and Atucha-2.

Discussion

A concern during the early phase of events involving power increases is the potential for the power pulse to increase fuel temperatures sufficiently high that molten $UO_{2\pm x}$/Zircaloy could form. In the unlikely event of this melt relocating outside the fuel bundle contacting the pressure tube, the local thermal interaction may cause failure of the pressure tube.

An important factor influencing the extent of fuel channel deformation due to melt contact is the internal pressure in the pressure tube at the time of melt contact [120]. Tests performed with high internal pressure induced greater pressure tube deformations than tests performed at low pressure with similar masses of melt relocating on them. The mass of melt that contacted the pressure tube also played a role in fuel channel integrity. Test programmes demonstrated the ability of an as-received ballooned pressure tube in contact with the calandria tube to withstand a very intense, relatively short lived hot spot induced by molten Zircaloy-4, provided the calandria tube was well cooled.

Local melt heat transfer to the pressure tube is a major phenomenon for large break LOCAs and flow blockage in single channel events. Molten fuel sheaths probably would not relocate onto the pressure tube as Zircaloy with dissolved oxygen wets uranium dioxide decreasing the driving force for relocation. However, for fast heat-up transients, there is a potential for some melt from end plates and end caps.

The most important factor in determining fuel channel deformation caused by melts is the cooling condition on the outside of the calandria tube. As mentioned above, another important factor is the internal pressure tube pressure at the time of melt contact [120]. The key phenomena occurring after a channel flow blockage would be:

- Coolant boil-off;
- Cladding heat-up and melting;
- Dripping of molten Zircaloy from the fuel elements;
- Thermal interaction between the molten Zircaloy and the pressure tube;
- Localized deformation of the pressure tube or calandria tube;
- Potential channel failure.

If the coolant surrounding the fuel is in nucleate boiling, the channel will remain cool, limiting the area over which fuel channel deformation occurs, providing a good heat conduction path from the melt to the surrounding water. In addition, failure may be inhibited if a rewet front moves from the cooler calandria tube as heat from the melt dissipates. Extensive experimental data was acquired on this phenomenon [120] and very conservative criteria were derived from these experiments. The pressure tube failure mechanism by contact with molten material in horizontal channel-type PHWRs is studied with limits of the amount of melt the pressure tube can sustain. This is done for cases where melt occurs before or after the pressure tube contact with the calandria tube, and in relation to pressure tube internal pressure and calandria tube outside surface heat transfer.

As in the previous section, this mechanism was not included in the IAEA questionnaire and, thus, no information was provided by the participating States.

3.2.4.2. Local plastic strain due to fuel element contact

Reported criteria

No information on this fuel failure criterion was specifically requested from the PHWR States. Local plastic strain due to fuel element contact is not considered as a safety criterion in some States. This criterion is also not applicable to vertical channel-type PHWRs, e.g. Atucha-1 and Atucha-2.

Discussion

Channel integrity could be threatened by the potential formation of local hot spots on the pressure tube by forced fuel element contact before pressure tube ballooning ensure a hard contact with the calandria tube. If such a hot spot was to occur, there is a potential for sufficient local strain to rupture the pressure tube. Failure occurs when a local region of the pressure tube necks down to a knife-edge. The Canadian industry is aware of this potential pressure tube failure mechanism for large break LOCA and is conducting joint research and development work to investigate the criteria that, if met, would prevent pressure tube failure.

For a ballooned pressure tube, the hot spot is expected to quench with minimum or no local deformation due to the enhanced heat transfer to the moderator.

Fuel elements can contact the pressure tube from several types of bundle deformation: (a) fuel element bowing; (b) bundle sagging/slumping, and (c) bundle settling.

Fuel element bowing can be due to differential axial thermal expansion across the fuel bundle cross-section, hydraulic drag, gravity or mechanical loading bending moments (e.g. bundle impact, constrained axial expansion). Due to the design features of the fuel elements for horizontal channel-type PHWRs, there are two possible types of contact between the pressure tube and the fuel element. The first type is contact with the fuel elements bearing pad and the other is contact with the fuel element sheath.

Experiments that study the first type of contact have shown that this process is self-limiting, where the heat transfer to the pressure tube is arrested by the local deformation of the pressure tube under the bearing pads once a hot spot has developed. Due to the plasticity of the fuel elements, the second type of contact could be maintained during the pressure tube local deformation. In this case, the integrity of the pressure tube depends on the contact load, the internal pressure inside the pressure tube, and the extent of the hot spot. The pellet-to-sheath contact forces and their effects on fuel element rigidity are not easy to quantify in most experiments [121]. The most thoroughly characterized tests were conducted without pellets inside the sheath, so the stiffness of the fuel elements may have been lower than expected under actual conditions [121].

Bundle sagging and slumping are deformation due to gravitational loading for temperatures above 973 K. In particular, sagging is a rapid deformation in the α/β transformation temperature range (1081 K – 1281 K) [122]. A few experiments have been done that involve fuel bundle slumping, but many of the tests in these experiments had high uncertainties [121].

Bundle settling refers to changes in the end plate geometry due to applied loads and temperature. While theoretical aspects of the deformation are known, the complexity of the bundle end plate geometries is high.

The information available on fuel bundle sagging, slumping, and settling is inadequate to characterize the onset of bundle deformation. Also, the large temperature differences in the tests between the top and bottom of the bundle, and axially along the bundle, may have caused some non-characteristic deformations. However, due to the large temperature sensitivity of the creep rate at elevated temperatures, the time of failure, during heating transients, are expected to be predicted accurately [123].

As in the previous section, this mechanism was not included in the IAEA questionnaire and, thus, no information was provided by the participating States.

3.2.4.3. Rolled joint failure due to fuel string relocation

Reported criteria

No information on this fuel failure criterion was specifically requested from the PHWR States. Rolled joint failure due to fuel string relocation is not considered as a safety criterion in some States. This criterion is also not applicable to vertical channel-type PHWRs, e.g. Atucha-1 and Atucha-2.

Discussion

As described in Section 3.2.3.5, during events like a large break LOCA, the coolant flow can change direction (flow reversal) and accelerate the fuel strings (or bundles) in the channels. If the bundles acquire sufficient kinetic energy during the acceleration time, the impact of the bundles against the channel-end structures could damage the channels. In Canada, the industry is aware of this potential fuel channel failure mechanism for large break LOCA and is conducting joint research and development work to investigate the criteria which, if met, would prevent fuel channel failure.

It is argued that this mechanism is physically impossible for the estimated conditions during accidents of horizontal channel-type PHWRs. However, it is considered prudent to establish a fuel acceptance criterion until this proof is obtained.

The pressure tube and end fittings in the fuel channel of a horizontal channel-type PHWR are joined together by roll-expanded joints. The pressure tube is rolled into the end fitting to a specified reduction in the wall thickness of the pressure tube. During roll expansion, the pressure tube material is forced into three circumferential grooves in end fitting to provide a strong and leak-tight joint. During the flow reversal during a large break LOCA, the rolled joint is expected to experience high strain-rate produced by the fuel string impact loads. At sufficient load, the pressure tube will fail near the corner of the inboard groove before the joint slips with a fracture surface at about 45º to the tube axis [103].

To assess the fuel string kinetic energy necessary for this failure mechanism, a set of experiments were performed for various kinetic energies. In these impact tests, the maximum value of the kinetic energy to prevent rolled joint failure was assessed. In tests with a 12-bundle fuel string, the maximum value was determined to be 14.6 kJ at a velocity of 6.3 m/s, with a standard error of 0.24 m/s in the velocity measurement [103].

Because it is unclear if this failure mechanism could be active during certain accident sequences, it is considered prudent to have a safety criterion for this pressure tube failure mechanism.

4. CONCLUSIONS

In response to the request of Member States with operating PHWRs, the IAEA, with the support of States' representatives, provided a platform where all PHWR States worked together to perform a systematic review of fuel acceptance criteria for PHWRs. This review was based on:

- The responses to a questionnaire circulated among Member States by the IAEA whose objective was to survey the PHWR fuel acceptance criteria established for normal operation, AOOs and postulated accident conditions in States with PHWRs;
- Presentations at a Technical Meeting organized by the IAEA and held at the CNSC offices in Ottawa, Canada (8-12 June 2015);
- Open literature documents related to PHWR fuel design;
- Additional information provided upon request by individual PHWR States.

This publication presented the results of the systematic review of the PHWR fuel acceptance criteria that are mainly used in horizontal channel-type operating PHWRs.

As mentioned earlier, the establishment of fuel acceptance criteria is an important step in the implementation of defence in depth. Once established, fuel acceptance criteria can be used, among other things, as key inputs in the:

- Determination, for normal operations and AOOs, of operating limits for the reactor control system;
- Derivation (a Canadian industry practice), for certain DBAs, of fitness-for-service criteria used for restart following the events;
- Formulation, for accident conditions, of special safety system trip set points acceptance criteria;
- Determination, through deterministic safety analysis, of the special safety system settings and the plant's safe operating envelope.

For some fuel acceptance criteria, different approaches or different values are used among participating States. This publication, as the first systematic review of PHWR fuel acceptance criteria, provides technical information considered important to better understand the PHWR degradation mechanisms and associated fuel acceptance criteria (addressed in Section 3).

Given the significant role that fuel acceptance criteria play in ensuring nuclear safety, further collaboration and discussion on fuel acceptance criteria topics between participating States would be beneficial. Examples include the following:

- The failure mechanisms in Section 3.2 for which no criteria were reported would probably be a good starting point.
- In Section 3.2.2.1, participating States reported limits for the different energy deposition (in the fuel matrix) as fuel acceptance criteria based on limiting the fuel total enthalpy (965 to 756 kJ/kg UO_2) and/or avoidance of fuel melting. It would be informative to compare the technical basis of these various reported limits.
- The development of fuel acceptance criteria which, if met, ensure maintenance of coolable fuel bundle geometry under accident conditions. Although bundle deformation is not a barrier failure mechanism as such, the loss of coolable fuel bundle geometry could trigger a number of the failure mechanisms discussed in this publication. The establishment of criteria which, if met, ensure maintenance of coolable fuel bundle geometry would be useful for the analysis of LOCAs. The development of criteria (which could be different than the criteria used for DBA analysis) for coolable bundle geometry could be of interest in DECs with no severe fuel degradation; the use of such criteria would increase one's confidence that pressure tube integrity is maintained in such accident scenarions.
- Following the Fukushima Daiichi nuclear accident, there was an international recognition that efforts have to be made to enhance protection and safety, for both existing and new nuclear power plants. One area that could benefit from international collaboration is the establishment, to the extent possible, of fuel acceptance criteria for certain DEC events. For example, criteria such as those discussed in Section 3.2.4.2 on pressure tube local plastic strain due to fuel element contact could be used to

demonstrate that fuel channel integrity is maintained under DECs with no severe fuel degradation.

APPENDIX I. REVIEW OF NUCLEAR DESIGN CRITERIA FOR PHWRS

I.1. TECHNCIAL BASIS

The nuclear design criteria discussed in this Appendix are related to performance parameters for the design of the reactor core. The core design refers to flux and power distribution within the reactor core, the design and use of reactivity control systems for normal operation and for shutting down the reactor, to stability, to the various reactivity feedback characteristics, and to the physics of the fuel.

The defence in depth and related principles are applied in the design of the reactivity control safety function such that the fission chain reaction is controlled and, when necessary, terminated under all credible circumstances, i.e. normal operation, AOOs and accident conditions. To achieve this, the core design provides for effective provisions to ensure success of the following safety functions:

(a) To prevent unacceptable reactivity transients;

(b) To prevent progression of AOOs into DBAs and to shut down the reactor to mitigate the consequences of DBAs;

(c) To shut down the reactor as necessary;

(d) To maintain the reactor in a safe shutdown state.

The reactor power is controlled by a combination of the inherent neutronic characteristics of the reactor core and its thermohydraulic characteristics, and the capability of the control and shutdown systems to actuate in all plant states, including normal operation, AOOs, DBAs and DECs without core melting.

Key core nuclear performance parameters include:

- The reactivity feedback to changes in the temperature of fuel, coolant and moderator;
- The shutdown margin;
- The maximum reactivity insertion rates;
- The control rod and control bank worth;
- The radial and axial power peaking factors;
- The maximum linear heat generation rate;
- The coolant density change induced reactivity;
- The kinetic parameters (delayed neutron fraction and neutron lifetime).

The performance parameters of the reactor core are typically used in safety analyses of control and shutdown systems' effectiveness to verify that fuel design and safety criteria limits are not exceeded for operational states and accident conditions, and that postulated reactivity accidents neither cause a significant damage to the coolant pressure boundary nor impair the coolable capability to the core.

This Appendix focuses mainly on the natural uranium core in horizontal channel-type PHWRs.

I.2. REACTIVITY FEEDBACK

Reported criteria

(a) No specific criteria are prescribed for reactivity coefficients (Canada);

(b) Negative power coefficient of reactivity under all operating conditions. Neutronic overpower trip set point is used as a guideline (India);

(c) No specific criteria, but reactivity coefficients are taken into account in safety analysis (Korea).

Discussion

The reactivity in a PHWR is affected by the following local parameters:

- Fuel temperature;
- Coolant temperature;
- Coolant density;
- Coolant purity;
- Moderator temperature;
- Moderator density;
- Moderator purity;
- Moderator poisons;
- Burnup.

A change in local parameters can impact the reactivity characteristics, and either mitigate or worsen reactivity transients. Whether the reactivity feedback is positive or negative, the design of a horizontal channel-type PHWR has established a balanced set of inherent reactor physics characteristics (small magnitude of the reactivity coefficients and long neutron lifetime), combined with an engineered control system (reactor regulating system) and safety systems (dual, redundant, highly reliable, fast acting independent shutdown systems), to manage the reactor response to the range of reactor conditions, namely for normal operation, AOOs, DBAs and severe accident scenarios.

A change in the fuel temperature in a PHWR is accompanied by a change in reactivity due to immediate heating of the fuel by the fission heat. The fuel temperature coefficient responds promptly to the energy deposited in the fuel, while the other coefficients like the moderator temperature coefficient have a delayed response. The reactivity effect of fuel temperature depends on the fuel composition and, therefore, on the burnup. The temperature effect is greater at lower fuel temperatures, for the same temperature change; however, the effect becomes minimal at high fuel temperatures due to thermal expansion.

A change in the coolant temperature affects the coolant density and the neutron spectrum. An increase of the coolant temperature decreases the density of the water which is similar to the effect of void increase. As coolant density decreases, the resonance absorption in the fuel decreases. In addition, an increased coolant temperature affects the fuel temperature because the coolant acts as the boundary condition for heat transfer from the fuel through the sheath.

The coefficient of reactivity due to the coolant density effect is positive, and the coefficient due to the spectrum effect is negative for fresh fuel and positive for equilibrium fuel. The overall reactivity effect is the net of these two phenomena.

The reactivity due to void generation occurs in the event of a break or a fault in the PHTS. The pressurized coolant in the channel may partially void and, therefore, add positive reactivity. The reactivity increase for partial coolant voiding would be smaller than that for full voiding. This applies also in a postulated loss of coolant from the primary circuit, as channels would void only partially. Both the amount and rate of voiding depend on the break size. The void reactivity is higher for fresh fuel. In general, the addition of absorbing material in the form of boron in the moderator, ordinary water as an impurity in the heavy water coolant, or parasitic absorbers in the structural materials (e.g. pressure tubes) increases the void reactivity. The shutdown systems are designed to limit the maximum magnitude of power pulse due to positive void reactivity such as not to exceed the maximum enthalpy specified for fuel dispersal in Section 3.2.2.1.

An increase in the moderator temperature decreases the moderator density and increases the moderator neutron temperature. Both effects make the moderator temperature reactivity negative for fresh fuel. With increasing fuel burnup, a shift from a negative to a positive coefficient is expected due to the accumulation of Pu-239. A change in moderator density also changes the poison concentration, if moderator poison is present. A postulated pressure tube/calandria tube rupture can release high pressure steam in the form of a bubble in the moderator. This will lead to a reactivity decrease, depending on the bubble size.

There are no specific criteria that explicitly establish acceptable ranges of values or preclude the acceptability of a positive reactivity feedback. The acceptability of the core reactivity feedback is determined in the reviews of the analyses in which they are used (e.g. control requirement analyses, stability analyses, and transient and accident analyses). The judgment to be made is whether the magnitude of reactivity feedbacks have been assigned suitably conservative values. The basis for that judgment includes the scenario in which it is important, the state of the art for calculation of the reactivity feedback, the uncertainty associated with such calculations, experimental checks in operating reactors, and any required checks in the start-up programme of the reactor.

I.3. MAXIMUM REACTIVITY WORTH AND MAXIMUM REACTIVITY INSERTION RATE

Reported criteria

No information on this criterion was specifically requested from the PHWR States.

Discussion

The reactivity of horizontal channel-type PHWRs is controlled by a set of mechanisms used for regulation (i.e. control) and protection (i.e. safety). These reactivity control mechanisms make changes to neutron absorption and, thus, control reactor power. For continuous short term reactivity control, a small amount of excess reactivity is necessary. This level of reactivity control is provided by the light-water liquid zone compartments. The purpose of the reactor regulating system is: (a) to maintain the reactor critical for normal operation; (b) to allow power manoeuvres; (c) to monitor and control the reactor flux shape; and (d) to monitor and reduce the reactor power at an appropriate rate (including controlled shutdown) if any parameter is outside specific limits. The reactor protection system, on the other hand, is designed for rapid insertion of a large amount of negative reactivity to shut down the reactor.

The complex physical (e.g. refuelling) and nuclear (e.g. fuel burnup, xenon transient) changes occurring in the core during reactor operation require more than one type of reactivity mechanism. The core reactivity changes require compensating and regulating controls. The mechanism designed to control or compensate the reactivity change in the core needs to have somewhat larger reactivity worth and operational time interval. The operational time period during which a mechanism is required to increase or suppress core excess reactivity determines the reactivity insertion rate ($\Delta k/s$). The reactivity insertion rate of a control device is limited so that control action at its maximum possible velocity does not challenge either the pressure boundary integrity or core coolability.

Design limits are placed on the maximum reactivity worth of control devices and on rates at which reactivity can be changed to ensure that the potential effects of sudden or large reactivity increase cannot challenge the pressure boundary or the coolability of the core.

Various control mechanisms in the reactor regulating system of a horizontal channel-type PHWR have different total reactivity worth. For example, in the standard CANDU 6 reactor design, the following control mechanisms are used: (a) 14 liquid zone control compartments (H_2O filled); (b) 21 adjuster rods; (c) 4 mechanical control absorbers; and (d) moderator poison. The total reactivity worth for this reactor design includes: (a) ~7 mk for the liquid zone controllers; (b) ~15 mk for the adjuster rod system; and (c) ~10 mk for the mechanical control absorbers. The maximum rate of change of reactivity (i.e. insertion rate) for a bank of adjuster rods is 0.1 mk/s. The maximum rate for liquid zone controllers is ±0.14 mk/s, and for mechanical control absorbers it is ±0.075 mk/s (driving) and -3.5 mk/s (dropping, de-energized clutches). It is important to note that these values, like the others reported in this section, represent theoretical CANDU 6 values. The actual values may vary between the individual CANDU 6 plants and reflect an unaged equilibrium (time average) core.

The total reactivity worth of shutdown system #1 (shutoff rods) is ~80 mk, while it is more than 300 mk for shutdown system #2 (poison tanks).

I.4. CONTROL OF LOCAL AND GLOBAL POWER

Reported criteria

No information on this criterion was specifically requested from the PHWR States.

Discussion

Local neutron flux disturbances can develop while the bulk power is held constant, due to the large core dimensions relative to the average distance travelled by a neutron. In an equilibrium horizontal channel-type PHWR, the overall channel-flux shape is nearly constant during operation, but local maxima and minima develop in the neutron flux, with changes to its amplitude and location because of changes in xenon concentration, fuel burnup, on-power fuelling, and the configuration of the reactivity devices.

The factors affecting core criticality and flux and power distribution during normal operation fall into three main categories:

- Fuel burnup;

- Xenon transients;
- Local parameter effects.

Fuel burnup (or irradiation) has a major effect on the fuel properties (and hence the reactivity), because of evolution of fuel nuclide composition with time such as fission in U-235, conversion of U-238 to Pu-239, fission in Pu-239, and generation of fission products. Eventually, reactivity decreases as fuel is irradiated since fission products accumulate and total fissile content decreases and refuelling with new fuel is necessary.

The local flux tilts needs to be controlled so that safe operating limits on fuel bundle power and channel power are not exceeded. In a cylindrical reactor like horizontal channel-type PHWRs, the ratio of average thermal flux in a homogeneous core (i.e. no flattening with adjusters or differential fuelling) to the maximum thermal flux is about 0.275. The power output of the reactor is proportional to the average thermal flux of a reactor, and a low average thermal flux creates a challenge to the economics of plant operation; fuel located away from the central peak in a channel contributes significantly less than its potential share of power. An upper limit on thermal flux is set by safety considerations based on a maximum heating rate to avoid fuel damage. The upper limit on the thermal flux is usually the maximum thermal flux.

In normal operation, a good fuelling strategy coupled with effective power regulation limits the local flux tilts. The reactor regulating system can automatically change the water level in each liquid zone controller. In this way, the neutron flux distribution is shaped towards design reference shape (known as ‘spatial control’), while keeping the reactor critical for steady operation or providing small positive or negative reactivity to increase or decrease power in a controlled manner (known as ‘bulk control’).

The improvement made to the ratio of average thermal flux to the maximum thermal flux is called ‘flux flattening’. There are four different methods in which the average flux is increased:

- Addition of a reflector to flatten flux in the radial direction;
- Adjustor rods for axial and radial flux flattening;
- Bi-directional refuelling for axial flux flattening;
- Differential exit burnup in the radial zones for radial flux flattening.

The first two methods are incorporated in the reactor design while the last two methods are achieved through fuel management. Use of all of these methods allows the ratio of average thermal flux to maximum thermal flux to be increased to about 0.55. This level of increase doubles the thermal output of the reactor without increasing the channel or bundle power above the peak set for safety considerations. Full spatial control is available when power reaches the threshold for onset of xenon oscillations at a certain low power. Xenon disturbances below this reactor power do not cause self-sustaining oscillations.

I.5. SHUTDOWN STATE

Reported criteria

(a) Each shutdown system should be capable to render the reactor subcritical. Guaranteed Shutdown States (GSS) for outages are typically achieved by overpoisoning the moderator (Canada);

(b) Shutdown margin of 10 mk. For GSS, the amount of boron has to be added on the basis of the assumption that the core is in its most reactive condition (e.g. no burnup credit, no fission products) (India);

(c) At least 26 of the 28 shutoff rods (shutdown system #1) have to be operable, and at least five of the six poison tanks (shutdown system #2) have to be operable. GSS and related procedures are specified in the technical specification. Negative reactivity worth of absorber rods and boron addition in moderator system shall provide compensation of absence of xenon (Korea);

(d) Each shutdown system, acting alone, needs to be capable of maintaining the reactor in a subcritical state indefinitely or long enough (i.e. 15 min.) to permit operator action to supplement the shutdown capability. GSS are accomplished by poisoning the moderator with a concentration determined assuming a safety factor of 2 (i.e. 100% uncertainty) (Romania).

Discussion

Adequate reactivity worth of shutoff rods or neutron absorbing liquid poison should be available to terminate the fission process. The liquid poison system used in horizontal channel-type PHWRs adds soluble chemicals with large neutron capture cross-sections. Boron trioxide (B_2O_3), gadolinium nitrate hexahydrate ($Gd(NO_3)_3 \cdot 6H_2O$), and lithium pentaborate decahydrate ($Li_2O \cdot 5B_2O_3 \cdot 10H_2O$) are the chemicals used.

Boron may be used to compensate for the lack of neutron absorbing fission products in the fuel at initial plant start-up with fresh fuel. Boron may also be used "as a poison shim during refuelling for long-term shutdowns and for any other situation which requires poison on a long-term basis" [42]. According to [42], "Gadolinium is used on reactor start-up following a long shut down, since the reduction in its neutron capture cross-section occurs at approximately the same rate as the fission product xenon is produced. A further advantage of using gadolinium is that the capacity of ion exchange resin for removing gadolinium is approximately 14 times greater than for boron."

A GSS is a state where the reactor remains in a stable, subcritical state independent of reactivity perturbations caused by any possible changes in core configuration, core properties, or process system failures. The following three different types of GSS are currently being practised in operating horizontal channel-type PHWRs: overpoisoned GSS, moderator-drained GSS, and rod-based GSS.

During planned maintenance outages, the overpoisoned GSS is commonly used to keep the reactor deeply subcritical. To ensure that the reactor is kept subcritical for a postulated in-core break event, in which moderator poison can be diluted, an adequate poison concentration is maintained. In addition, at least one shutdown system should be available in automated mode. The use of the GSS therefore ensures adequate protection against possible loss of reactivity control events or in-core breaks.

The process of maintaining the overpoisoned GSS is complex as well as labour and time intensive. This type of GSS is unique to the horizontal channel-type PHWR design, whereas other nuclear reactor designs use control rods and shutoff rods to establish a GSS. For horizontal channel-type PHWR designs, an adequate subcriticality margin can also be maintained by guaranteeing all solid shutoff absorbers, control absorbers, and adjuster absorbers are in the core, along with a small amount of poison in the moderator. This configuration is referred to as rod-based GSS [124]. A minimal amount of poison in the moderator is maintained as a double contingency measure; and the shutdown system #2 is poised and available to provide additional protection and satisfy the defence in depth requirement. The rod-based GSS has been currently used at few horizontal channel-type PHWR units for unplanned and short duration shutdowns.

I.6. FUEL ENRICHMENT

Reported criteria

(a) In Atucha-2, natural uranium fuel bundles are used. Atucha-1 utilizes slightly enriched uranium (0.85 wt% U-235). Although both have slightly different U-235 content, the safety limits applied to Atucha-1 and Atucha-2 fuels are the same (Argentina);

(b) Fuel bundles in Canadian nuclear power plants use natural uranium as fissile material (Canada);

(c) Fuel enrichment is not used (China);

(d) Fuel enrichment is not used (India);

(e) Enrichment is not used (natural and depleted uranium are used only) (Korea);

(f) Fuel enrichment is not used. (Romania).

Discussion

Typical horizontal channel-type PHWRs use natural uranium, which contains 0.71 wt% U-235. The main energy source in natural or enriched uranium is the initial fissile content of U-235. When the fuel is burned in the reactor, plutonium is produced from neutron absorption in the fertile isotope U-238. Two fissile isotopes, Pu-239 and Pu-241, are produced in the fuel and burned to release energy. While U-235 content is depleted in PHWRs, there is also a net increase in plutonium with time. Fissioning of Pu-239 provides up to about 50% of the total energy produced in a horizontal channel-type PHWR. When the fuel is discharged, the total fissile content is still 0.5 wt%, consisting of 0.23 wt% U-235 and 0.27 wt% Pu-239 and Pu-241.

The use of slightly enriched uranium fuel with 0.9 to 1.2 wt% U-235 has been considered for new designs, such as the ACR-1000 reactor design, and for reduction of the void reactivity magnitude in operating CANDU reactors (low void reactivity fuel).

During initial start-up, the central channels (i.e. about 80 channels) usually contain depleted uranium fuel bundles in order to limit maximum bundle and channel powers by shaping the power distribution. Fresh (unirradiated) depleted fuel (0.52 wt% U-235, two bundles per channel) is used as a substitute for irradiated fuel. The outer channels (about 300 channels) have fresh natural uranium (0.71 wt% U-235) fuel. The higher reactivity of the fresh fuel will be suppressed by adding a soluble neutron absorber (boron) to the moderator.

I.7. FUEL MANAGEMENT

Reported criteria

No information on this criterion was specifically requested from the PHWR States, except burnup. The following are the reported criteria for maximum bundle burnup at discharge from the core:

(a) 360 MW·h/kgU for Atucha-1, 305 MW·h/kgU for Atucha-2 (Argentina);

(b) 450 MW·h/kgU (Canada);

(c) No specific burnup limit (China);

(d) 360 MW·h/kgU (India);

(e) No specific burnup limit (Korea);

(f) 360 MW·h/kgU (Pakistan);

(g) 290 MW·h/kgU (Romania).

Discussion

The objective of fuel management is to establish strategies for fuel loading and fuel replacement to operate the reactor in a safe and reliable mode while keeping the total energy costs low. To achieve this objective, the following in-core fuel management practices are implemented:

- Adjust refuelling rate to maintain the reactor critical;
- Control core power to comply with safety and operational limits and conditions on fuel power;
- Maximize fuel burnup within operational limits and conditions to minimize fuel replacement costs;
- Avoid fuel defects to minimize potential radiological consequences and fuel replacement costs.

A specific design feature of CANDU reactors is on-power refuelling, which is used for long term reactivity control to maintain near constant core reactivity while fuel is irradiated. The flux and the power distribution are well controlled by conducting refuelling with different frequencies in different zones of the reactor core.

Core calculations are typically performed two to three times a week to track and monitor the in-core flux, power, burnup distributions, and discharge burnup of individual bundles. The selection of channels for refuelling is based on the current core state, power and burnup distributions. Normally eight bundles are shifted during one channel refuelling. It typically takes two and a half hours to refuel one fuel channel, which indicates that the refuelling is almost continuously processed during normal working hours. Fresh fuel is loaded directly into the magazine of the fuelling machine, whilst irradiated fuel is remotely discharged under shielded conditions into the irradiated fuel bay and stored under water.

The fuel management analysis typically includes the entire spectrum of operating states:

- Normal operation at full power;
- Load following (as applicable);
- Approach to criticality and power increase;
- Power maneuvering;
- Startup;
- Refuelling;
- Shutdown;
- Anticipated operational occurrences.

In the analysis, fuel element performance is assessed and fuel integrity is confirmed by demonstrating that the respective fuel acceptance criteria (in Section 3.1) are met for all operational states. The same approach also applies where different fuel bundle designs or different fuel compositions co-exist in the reactor core. Therefore, no explicit criteria for fuel management are required.

I.8. REGULATORY REQUIREMENTS AND PRACTICES

I.8.1. Canada

In Canada, the regulatory document REGDOC-2.5.2 on Design of Reactor Facilities: Nuclear Power Plants [9] establishes requirements and guidance for the design of new water-cooled nuclear power plants. To a large degree, it represents the CNSC's adoption of the principles established by SSR-2/1 (Rev. 1) [4], and the adaptation of those principles to align with Canadian practices. The guiding principle is the promotion of implementation of multiple levels of defence in the design rather than the prescription of specific limits for individual performance parameters for the design of the reactor core. Section 8.1 of REGDOC-2.5.2 sets out the following requirement: "The design of the core shall be such that:

(a) the fission chain reaction is controlled during operational states,

(b) the maximum degree of positive reactivity and its maximum rate of increase by insertion in operational states and DBAs are limited by a combination of the inherent neutronic characteristics of the core, its thermal-hydraulic characteristics, and the capabilities of the control system and means of shutdown, so that no resultant failure of the reactor pressure boundary will occur, cooling capability will be maintained, and no significant damage will occur to the reactor core". [19]

The guidance provided in REGDOC-2.5.2 [9] includes the following:

- "The nuclear design should incorporate inherently safe features to reduce the reliance on engineered safety systems or operational procedures. Defence in depth and related principles should be applied in the design of the reactivity control safety function, such that the fission chain reaction is controlled during operational states, and, when necessary, terminated for DBAs and DECs";

[19] Canadian Standards Association (CSA) Standard N290.4-11 on Requirements for reactor control systems of nuclear power plants (2011; reaffirmed 2016) [14], and CSA Standard CAN3-N290.1-80 on Requirements for the Shutdown Systems of CANDU Nuclear Power Plants (1980; reaffirmed 2011) [125].

- "The design of the reactor core and associated coolant and fuel systems should take into account all practical means so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity and power. The consequences of those accidents that would be aggravated by a positive reactivity feedback should be either acceptable, or be satisfactorily mitigated by other design features";
- "It is recognized that reactivity coefficients of the design are important in determining the reactor behaviour and safety characteristics. This document does not have specific requirements on the sign or magnitude of the reactivity coefficients including the power coefficient of reactivity. Instead, this document requires a number of design provisions related to the nuclear design to ensure that the design is acceptable for reactor control, stability and plant safety";
- "If a reactor design has a positive power coefficient of reactivity for any operating state, the design authority should demonstrate that operation with a positive power coefficient is acceptable, by showing:
    - a bounding value of power coefficient of reactivity has been calculated for all permitted operating states and used in control, stability, and safety analyses,
    - measurements of the power coefficient of reactivity are conducted at start-up and periodically for certain operating limiting core conditions to demonstrate that measured values are bounded by calculated values with adequate margin,
    - the reactor control system is designed with adequate reliability and has the capability to automatically accommodate for a positive power coefficient of reactivity for a wide range of AOOs".

I.8.2. Korea

Regulatory requirements and practices in Korea are described in Section 2.3.4 of this publication.

APPENDIX II. REVIEW OF THE CANADIAN APPROACH TO FITNESS FOR SERVICE CRITERIA FOR CANDU FUEL

II.1. TECHNICAL BASIS

Anticipated operational occurrence (AOO) events have “the potential to challenge the safety of the reactor and might be reasonably expected to happen during the lifetime of a plant, with frequencies of occurrence equal to or greater than 10^{-2} per reactor year” [9]. It needs to be ensured that the fuel remains fit for service when the reactor comes back to normal operation following an AOO. For this purpose, in Canadian nuclear industry, so-called ‘fitness-for-service criteria’ are used.

The concept of fitness-for-service criteria for fuel is an integral approach to ensure that the fuel and fuel channel maintain their structural integrity under the environment that several fuel degradation mechanisms are stimulated by an AOO. Therefore, the fitness-for-service criteria are derived on the basis of assessments of associated fuel degradation mechanisms using a stylized conservative analysis approach. In the stylized analysis, the fuel thermomechanical behaviour is simulated taking into account coolant pressures, bundle powers and power ramps typical of AOOs, while keeping the fuel sheath temperature constant throughout the 60 seconds simulation time of the event. The contents of this Appendix are primarily taken from COG-12-2049 [16].

Fitness-for-service criteria are usually used by reactor operators as a simple and reliable measure to provide assurance that the performance of the fuel returned to normal operation following an AOO still remains within its acceptable range. Therefore, fitness-for-service criteria are specific to fuel design and also to transients considered.

Fitness-for-service criteria in this Appendix are derived for a specific fuel type (i.e. 37-element bundle) and specific AOOs. However, the approach used is stylized and, as long as any other accident conditions (rather than AOOs) fall within the conditions on which these criteria are derived, the criteria have also to be applicable to these accident conditions. It is agreed that demonstration of such applicability would be required in this case.

This Appendix is intended to demonstrate how fuel acceptance criteria (such as those reported by Canada in Section 3.1 and referred to as ‘acceptance criteria/operating limits’ in Table II.1) established for individual physical (surrogate) parameters are used through fitness-for-service criteria by reactor operators. As example, fitness-for-service criteria developed for a 37-element fuel bundle covering bundle power and thermohydraulic conditions bounding those of Bruce Power, Ontario Power Generation, and New Brunswick Power reactors are as follows:

- The fuel sheath temperature remains equal to or below 450°C during an AOO;
- The incremental increase in the sheath strain as a result of the AOO remains equal or below 0.5%.

These were derived by considering sheath temperatures up to 450°C and a maximum initial bundle power of 969 kW with low and high bundle burnups. The analysis covered a coolant pressure range from 9 to 11 MPa and power ramps of 0.3%/s, 0.6%/s and 1.0%/s, respectively.

The above fitness-for-service criteria were derived on the basis of assessments of the following fuel degradation mechanisms (in Section 3.1, these mechanisms are referred to as ‘fuel damage mechanisms'), i.e. processes that cause, or have the potential to cause, fuel performance degradation:

- Fuel sheath strain;
- Fuel pellet-cladding mechanical interaction;
- Axial relocation of fuel pellet fragments;
- CANLUB degradation;
- Stress corrosion cracking of the sheath;
- Fuel element bowing and bundle deformation (including end plate deformation);
- Fuel sheath collapse into axial gaps;
- Lobe collapse or fuel sheath longitudinal ridging;
- Sheath oxidation;
- Hydride formation;
- Hydride migration;
- Pre-defected fuel element degradation (fuel oxidation, sheath oxidation and embrittlement).

In the assessment of individual fuel degradation mechanisms, criteria for determining acceptable performance of the fuel need to be used; these can be selected from fuel design specifications and/or operational experience including experiments, to ensure that the fuel after return to service from an AOO will remain in similar conditions under normal operation before the transient or will remain within the acceptable performance range. As such, operating limits were taken into account as criteria for determining acceptable performance of the fuel in deriving the above-mentioned fitness-for-service criteria. These criteria (i.e. operating limits) are summarized in Table II.1.

In Sections II.2 through II.13, fuel degradation assessments are discussed using operating limits as the acceptance criteria.

The results of fuel degradation assessments based on fuel operating limits reveal that sheath strain is the most limiting degradation mechanism for sheath temperatures up to 450°C (calculations were done for temperatures between 350°C and 450°C in steps of 25°C). The maximum increase of sheath strain, 0.5%, during a transient is used as the acceptance criterion, as it is supported by CANDU fuel operational experience.

Based on the stylized analysis and fuel degradation assessment results, fitness-for-service time versus power ramp maps were developed for different peak sheath temperatures, as illustrated in Fig. II.1. These fitness-for-service maps were applied to demonstrate fuel fitness-for-service for three loss-of-regulation control cases at the Darlington nuclear power plant. Detailed simulations of fuel thermomechanical behaviour were also performed for the same AOO cases; the analysis results confirmed the conservatism embedded in the fitness-for-service maps.

In general, the fitness-for-service maps can be used for any AOO whose conditions fall within the above analysis range on which the maps are based. If the fuel fitness-for-service cannot be ascertained from the maps, detailed thermal and mechanical assessments of the fuel using the fuel acceptance criteria in Table II.1 would be required.

TABLE II.1. ACCEPTANCE CRITERIA (OPERATING LIMITS) USED FOR CANDU FUEL PERFORMANCE ANALYSIS DURING ANTICIPATED OPERATIONAL OCCURRENCES (CANADIAN PRACTICE)

DEGRADATION MECHANISM	ACCEPTANCE CRITERIA (OPERATING LIMITS)
Fuel sheath strain	– Sheath strain increase due to an AOO is less than 0.5%.
Fuel pellet-cladding mechanical interaction	– Fuel defect frequency does not increase due to fragments relocation in the gap if the fuel is subjected to vibration or shaking; and – There is no incremental increase in fuel temperature following an AOO.
Axial relocation of fuel pellet fragments	– Fuel defect frequency does not increase due to axial fragments relocation in the gap if the fuel is subjected to vibration or shaking as a result of an AOO.
CANLUB degradation	– Bundle power ramp capability is not adversely affected by AOOs.
Stress corrosion cracking of the sheath	– Sheath failure by stress corrosion cracking does not occur during AOOs; and – There is no degradation of the fuel sheath tolerance to power ramps.
Fuel element bowing and bundle deformation Fuel sheath collapse into axial gaps	– Fuel element bowing is less than 0.47 mm. – End plate deformation/distortion is less than 0.5 mm. – There is no significant elastic or creep collapse of the sheath that is not supported by pellets.
Lobe collapse or fuel sheath longitudinal ridging	– The stress in a ridge of the sheath is limited below 0.2% yield strength where creep effects will not increase the height of the ridge.
Sheath oxidation	– Fuel sheath oxidation is less than 10% of the wall thickness (based on normal fuel sheath manufacturing specifications). This is equivalent to ~64 µm of ZrO_2 uniform thickness (averaged over the sheath circumference). [local or limited oxidation] – The additional oxide layer causes no more than 5°C temperature increase on the inside surface of the sheath at 100% full power. This is equivalent to ZrO_2 thickness of ~6.8 µm for peak power bundle. [global or reactor-wide oxidation]
Hydride formation	– There is no increase in hydride concentration beyond the maximum observed in CANDU fuel normally. A maximum hydrogen concentration of 200 µg/g (hydrogen equivalent) has been observed in fuel where operation or fuel handling has not been impaired. The normal hydrogen concentration is less than 100 µg/g.
Hydride migration	– There is local precipitation of hydride as a result of AOOs.
Pre-defected fuel sheath degradation	– Average O/U atomic ratio in pellets remains below 2.231. – There is no significant increase in fuel centreline temperature due to thermal conductivity changes in the pellet. – Increase in oxidation thickness is less than 0.5 µm for the outer surface and 1.2 µm for the inner surface. – Increase in sheath hydride concentration remains below that of the intact sheath. Equivalently, no more than an additional 20 µg/g, i.e. 10% of the maximum concentration of hydrogen observed in fuel where operation or fuel handling has not been impaired.

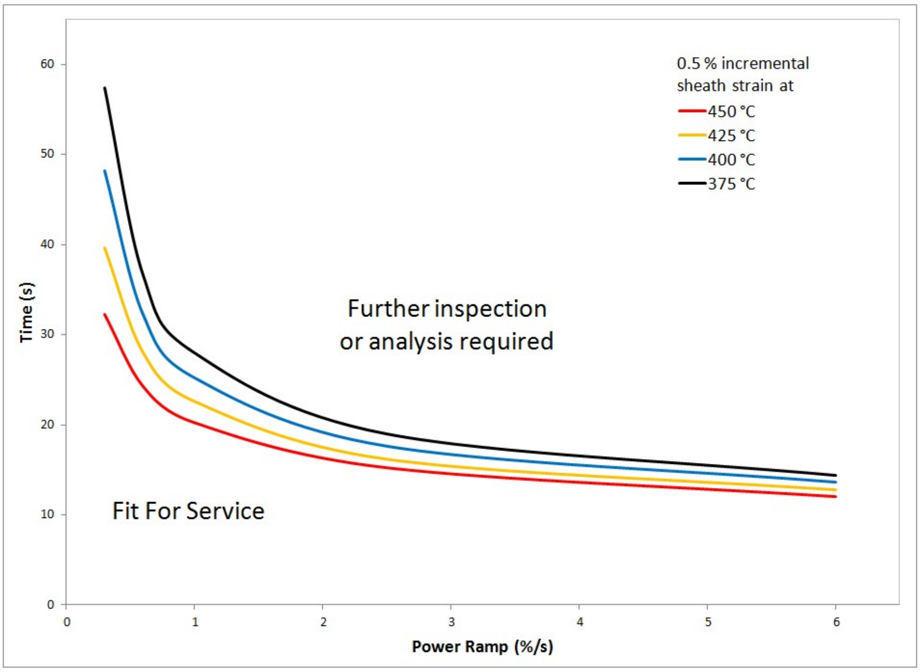

FIG. II.1. Fitness-for-service map at different peak sheath temperature (reproduced courtesy of CANDU Owners Group [16]).

II.2. FUEL SHEATH STRAIN

The fuel element contains a mixture of fill gases, added during manufacture, and fission gases, generated during normal operation. Under accident conditions, as the fuel element temperature increases, the internal gas pressure increases and the yield stress of the sheath decreases. If the element internal gas pressure exceeds the coolant pressure on the outside of the fuel element such that the hoop stress exceeds the yield stress, the sheath will undergo elastic/plastic strain. If the straining of the sheath continues unrestricted, the sheath would eventually fail. Also, under accident power ramp conditions, the fuel expands faster than the sheath such that the hoop stress may exceed the yield stress, the sheath will undergo elastic/plastic strain.

Reported criteria (operating limit)

Acceptance criterion for sheath strain based on operating limits is that the sheath strain increase due to an AOO is less than 0.5%. This limit is consistent with the acceptance criterion used for hot cell examinations of irradiated bundles[20] [126–128] and ensures that the increase in fuel centreline temperature is insignificant upon return to service under normal operating conditions as a result of the AOO. In addition, small changes in fuel element diameter (<0.5%) do not significantly affect the subchannels area and the margin to dryout under normal operating conditions. Also, the 0.5% sheath strain is much less than the strain necessary to cause fuel element-to-fuel element or fuel element-to-pressure tube contact because this strain limit will cause an increase in the sheath outer diameter that is less than the spacer height or bearing pad height.

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[20] Post irradiation examination has shown that normal operation can result in fuel sheath average strain of up to 1% on the outer elements with only a gradual increase to significant defect probability under normal operating condition.

Discussion

For constant bundle powers up to 969 kW at constant sheath temperatures up to 450°C, the increase in sheath strain is not significant for the short duration of the AOO. For power-ramped bundles, the time to reach the incremental sheath strain of 0.5% decreases with increasing power ramp rate or increasing sheath temperature. The fitness-for-service map in Fig. II.1 depicts the behaviour.

II.3. FUEL PELLET-CLADDING MECHANICAL INTERACTION

At high power, the fuel pellets expand and provide good contact with the externally pressurized sheath. During high power operation, small cracks are developed in the fuel pellet which could link and develop into fragments. During shutdown, if the system is depressurized, the sheath may become free standing due to the internal gas pressure resulting in a formation of a radial gap between the sheath and the pellet. Any loose fragments from the cracked pellets could relocate into this radial gap, particularly if the fuel is subjected to vibration or shaking. The relocated fuel fragments could provide local stress raisers when the reactor is returned to normal operating conditions. This could result in local stress in the sheath which can lead to SCC failure.

Reported criteria (operating limit)

Acceptance criteria for pellet-cladding mechanical interaction based on operating limits are that the fuel defect frequency (i.e. the number of defects per year per station) does not increase due to fragments relocation in the gap if the fuel is subjected to vibration or shaking; and that there is no incremental increase in fuel temperature as a result of an AOO.

Discussion

Fuel pellets manufacturing specification guidelines are provided to ensure satisfactory interaction between the pellets and the sheath. Changes in pellet surface finish and diameter are expected as the fuel operates under hot conditions. An increase in operating power causes a corresponding increase in fuel pellet temperature and diameter. At a linear heat generation rating >7 kW/m, the ceramic UO_2 fuel pellets start to develop radial cracks because of internal thermal stresses [129]. During high power operation, these small cracks link and develop into fuel fragments.

Radial fuel relocation and its impact on fuel sheath strain were reviewed by the Canadian industry and practices for avoiding excessive sheath strain during the return to power were recommended. As a result, reactor power operating restrictions are provided for the return to power following a prolonged period at reduced power. For example, to minimize the effect due to creep-down during shutdown, an empirical procedure to restrict the rate of power rise (and hence give time for stresses to relax), was devised on the basis of U.S. work performed in support of LWR fuel [130].

Experimental observation

Observations from experiments [131, 132] showed that the fuel pellets expanded axially and diametrically during irradiation and after half of the sheath circumference was removed and the element shaken, the fuel pellets broke apart at the crack sites that had developed during irradiation. Thus, fuel fragment relocation was thought to be a possible mechanism for ratcheting fuel strain during successive power cycles.

Experiments were performed [133] to investigate the response of sheath strain to element operating power. These tests measured the in-reactor change in sheath diameter as a function of time at power. The tests showed that the sheath strain increased to 0.94% at the pellet interfaces at the start-up (at an-operating power of 61 kW/m) and decreased to about 0.30% after four to five days at full power. During another irradiation test, the sheath strain was found to return to a higher level following a shutdown. The higher sheath strain dissipated rapidly to pre-shutdown levels after less than five full power days. Tests using previously irradiated fuel showed a similar response in sheath strain to reactor shutdowns, but generally required about ten days at full power to return to pre-shutdown strain levels. The sheath strain at 55% full power also responded to reactor shutdowns but remained considerably lower than sheath strains at 100 percent full power (sheath strains ranged from about 0.40% to 0.10% at 55% full power, compared to about 0.70% to 0.45% at 100% full power).

Power ramp rate experiments show that LWR fuel can be successfully increased in power at a maximum rate equivalent to 0.33 kW/m per hour, up to a maximum power of approximately 52 kW/m [130] which is comparable to representative CANDU fuel linear power values of 50 kW/m to 56 kW/m. This represents an increase in fuel rating of about 0.6% full power per hour. Power cycling tests following this ramp rate confirmed satisfactory fuel performance [130].

Operational guidelines

As sheath strain was found to increase during a return to full power from a shutdown, restrictions on operating power are necessary to avoid excessive sheath stress. The sheath stress is reduced if cracks are allowed to heal and fragments to refit within the pellets by fuel creep.

The reactor power can be increased to a level of about 70% full power which will initiate grain growth in the high power elements. At this power level, the sheath strain returns to a level similar to that prior to shutdown. The reactor power can then be increased at 0.5% full power per hour up to 100% full power. This slow ramp rate allows progressive fuel creep and crack healing, and a return to pre-shutdown operating conditions.

Industry experience with the operational guidelines shows that slow power ramp rates allow progressive fuel creep and crack healing, and a return to pre-shutdown operating conditions.

It is noted that the coolant pressure for AOO events remains high and the sheath is collapsed on pellets, so no fragment relocation is possible. In addition, since the expected maximum sheath strain due to an AOO transient is within the fuel design limits and the observed inspection results, this mechanism is not specific to AOO transients but generally applicable to shutdown and start-up in general and is to be avoided by following operational guidelines.

II.4. AXIAL RELOCATION OF FUEL PELLET FRAGMENTS

After a reactor shutdown, the pellets contract due to lower fuel temperatures, resulting in the reappearance of an axial gap distributed between the pellets. If the fuel is subjected to vibration, shaking during a shutdown, movement during discharge from one channel and reloading into another channel in the reactor or during fuel channel inspections, then fuel fragments may relocate axially and reduce the overall axial gap during the return to full power. This may lead to new, excessive loading of the end cap and stress related failure of the end cap weld.

Reported criteria (operating limit)

Acceptance criterion for the axial relocation of pellet fragments in the fuel element based on operating limits is that the fuel defect frequency does not increase due to axial fragments relocation in the gap if the fuel is subjected to vibration or shaking as a result of an AOO. It is noted that an axial gap in the fuel element between the UO_2 stack and the end caps is designed to accommodate the expansion of the fuel pellets during high power operation, thereby reducing the likelihood of element failures from axial loading of the fuel stack on the end cap.

Discussion

CANDU fuel operation experience includes many occasions where the fuel has been moved without leading to fuel failure by axial interaction between the fuel stack and the end caps. Examples include:

- ‘Swing’ fuelling schemes for first charge fuel: where low burnup fuel is removed from one channel and re-inserted into another to improve fuel utilisation;
- Fuel channel inspections: whereby two or more bundles are temporarily removed from a channel to allow inspection tooling to record pressure tube data;
- Spacer Location And Repositioning (or called SLAR) programmes: where fuel is temporarily removed from the channel to allow tooling to relocate spacer rings. A core-wide, SLAR programme was recently completed at Point Lepreau, without affecting fuel reliability.

Many of these experiences have become routine practice throughout reactor operations without leading to fuel failures.

The coolant pressure for the AOO events remains high and the sheath is collapsed on pellets, so no fragment relocation is possible. The AOO transient is not expected to increase the fuel defect frequency due to axial fragments relocation in the axial gap if the fuel is subjected to vibration or shaking during normal operation following the AOO event. Therefore, this mechanism is not expected to be a concern for AOO transients and is qualitatively addressed in light of the above rationale and experiences observed in fuel handling during operation and maintenance activities.

II.5. CANLUB DEGRADATION

During the fuel manufacturing process, the internal sheath surfaces are coated with a graphite-based liquid that is cured in an oven at a specific temperature for a given time. The resulting coating on the sheath surface is, on average, about 5 μm thick and is referred to as ‘CANLUB’. CANLUB is added to the fuel because it is effective in preventing iodine SCC of the sheath under power ramp conditions [134]. Outer elements are at the greatest risk of failure by SCC because of their higher operating powers and greater iodine inventories.

Although graphite is effective as a pellet lubricant in reducing sheath stress, the CANLUB layer has been shown to play a more important role as a fission-product chemical barrier and perhaps enhancing the gap heat transfer coefficient. There is also considerable evidence that it controls reaction of fission-freed oxygen with the sheath, which in turn may have a profound impact on the chemical speciation and migration of several key fission products [135–139]. Progressive degradation of the integrity of the CANLUB layer with increasing burnup has been reported in CANDU operational experience [128]. Some progress has been achieved in understanding the mechanisms involved in the function and degradation of the CANLUB layer, which could lead to the development of a more robust barrier system [138, 140]. However, a full understanding of how CANLUB works has not been reached by the industry.

The concern is that subjecting some fuel sheaths to temperatures that are higher than normal operating temperatures could lead to deterioration of the CANLUB protective layer effectiveness and may compromise the performance of the bundles returned to service when subjected to power ramp scenario, as discussed below.

Reported criteria (operating limit)

Acceptance criterion for the CANLUB degradation in the fuel element based on operating limits is that the bundle power ramp capability is not adversely affected by AOOs.

Discussion

During the bundle manufacturing process, the sheath temperatures are increased above room temperature on up to two occasions (excluding local heating effects during brazing of appendages). Following initial coating of the inside sheath surface by CANLUB, the sheaths are baked in a furnace, at up to about 390°C, to remove excess hydrogen from the CANLUB mixture. Additionally, as described in the fuel manufacturer's design specification, one bundle per month is autoclaved as a method of routine fuel inspection. The autoclaving process involves exposing the bundle to steam at 400°C for over 16 hours.

Autoclaved bundles have been randomly inserted into the reactor core for years. Fuel manufacturers, who are informed of the serial numbers of all bundles with unassigned defect causes, indicate that the performance of autoclaved bundles is no different from non-autoclaved bundles. Additionally, a review of fuel performance has shown that SCC failures in a 15 years period are the result of power excursions and are not related to normal reactor operations [141]. Thus, the power ramp performance of autoclaved bundles is not degraded by the increased sheath temperatures during the inspection process. The analyzed AOO transient duration (a few seconds/minutes) is a small fraction of the autoclave hold-time.

Additional manufacture information based on thermo graphic analysis of CANLUB lots showed, in controlled temperature degradation tests, that the majority of hydrocarbons outgassing occurs below 400°C. No chemical reactions take place until 550°C, at which point the ash is developed. The manufacture vacuum bakes the tubes below 400°C and essentially degrades and outgases the content, leaving almost no hydrogenous compounds. The tube surface consists of nearly pure carbon. As long as there is no carbon reactive content in the element annulus (20% residual air, 80% helium plus fission gases), the carbon is stable until 550°C.

A fuel maximum sheath temperature of 450°C is assumed in this assessment for the entire surface of the element. Under AOO conditions, only a small part of the element would be in dryout while the rest will be wet almost at the coolant temperature. Hence, it is not expected that a patchy increase of temperature up to 450°C on the fuel surface for a short duration impairs the CANLUB performance. In addition, a partial loss of CANLUB was observed with no fuel performance issues. Progressive displacement of the CANLUB layer as a function of burnup has been well documented through post-irradiation examinations by optical and scanning electron microscopy, and it is known that the CANLUB begins to adhere to pellets rather than the sheath during operation [136, 142].

Based on the above information, the maximum sheath temperature of 450°C would not have any adverse impact on the CANLUB performance during the short duration of an AOO transient.

II.6. STRESS CORROSION CRACKING OF THE SHEATH

Iodine is a highly reactive fission product that is released to the sheath-to-pellet gap during normal fuel operation. At fuel operating temperatures up to about 700°C, iodine can contribute to sheath deterioration by SCC. Sheath failures by SCC require a critical combination of iodine concentration and sheath stress. The sheath stress associated with past SCC fuel failures is usually the result of expanding fuel pellets under power ramp conditions.

Reported criteria (operating limit)

Acceptance criteria for the SCC of the sheath based on operating limits are that sheath failure by SCC does not occur during AOOs, and that there is no degradation of the fuel sheath tolerance to power ramps.

Discussion

Sheath failure by SCC requires a critical combination of iodine concentration and sheath stress. This combination may be reached during power ramps or due to pressure difference across the sheath. Several experiments were performed to evaluate the time necessary for sheath SCC in an iodine environment [143–147]. Based on these experiments, correlations were developed to predict the failure time of the sheath at a given hoop stress for temperatures between 300°C and 700°C. The predictions are characterized by a minimum hoop stress, below which, sheath failure will not occur. The tests were completed with iodine surface concentrations above 1 mg/cm^2 and with no protective CANLUB coating. Increases in the iodine concentration beyond this level were found to have no further effect on sheath corrosion.

The SCC failure threshold of the sheath is a function of time, temperature, internal gas pressure and coolant pressure. It has a maximum value at the beginning up to two hours and decrease linearly in about 200 hours of exposure to a minimum value. The failure threshold minimum value is 15% higher than the hoop stress for the most limiting bundle power and internal gas pressure at normal operation. The SCC failure threshold at 400°C is in the order of 500 MPa [144] for approximately the first two hours, decreases linearly to 170 MPa in 200 hours, and stays constant for the duration of the sheath exposure to the conditions.

For AOO transients, the sheath exposure time to the conditions is limited only to a few seconds/minutes, and the failure threshold for 450°C is expected to decrease slightly below 500 MPa but remains high. As a result, no sheath failure by SCC is expected since hoop stresses during the short duration of AOO transients for sheath temperatures up to 450°C are significantly lower than SCC failure threshold limit. In addition, the concentration of the iodine is far from the critical values necessary for corrosion and failure of the sheath over the short time that characterizes the AOO transients, which is far shorter than that of the experiments.

II.7. FUEL ELEMENT BOWING AND BUNDLE DEFORMATION

Bowing of a fuel element occurs as a result of an asymmetric temperature distribution on the fuel element. When there is a temperature difference on the fuel element (one side of the fuel element is hotter than the other), as a result of the differential thermal expansion, the element could bow in the direction of the hotter and more expanded material. Bowing deformations are primarily due to bending moments produced by thermal and mechanical mechanisms and, to a lesser extent, due to external loads. An example of external loads that may induce element bowing is the compressive axial loads transmitted through the end plates by neighbouring elements.

Reported criteria (operating limit)

Acceptance criteria for fuel element bowing and bundle deformation based on operating limits are that the fuel element bowing is less than 0.47 mm and the end plate deformation/distortion is less than 0.5 mm. These limits are equal to the acceptance criteria used for hot cell examination of the irradiated bundles. These criteria prevent fuel element-to-fuel element contact and fuel element-to-pressure tube contact and ensure that no plastic deformation of the end plate does occur.

Discussion

Tests were conducted at Stern Laboratories to characterize the fuel element lateral deflection and permanent strain caused by dryout during possible transient conditions. The test was intended to impose realistic temperature fields due to dryout patches on the fuel sheath with steep temperature gradients while under coolant pressure of 10 MPa [39, 148, 149]. Based on the results of these tests, for a temperature gradient of 100°C, the maximum measured bow is 0.15 mm and the plastic bow is ~0.1 mm. Under AOO conditions, the maximum sheath temperature is restricted to 450°C, and the maximum temperature gradient around the sheath will not exceed 100°C.

Under conditions with much higher temperature gradient where flow stratification occurs, the maximum fuel element bowing was calculated using the model presented in [150]. The fuel bundle geometry is such that the inter-element spacing is approximately 1.8 mm, and that the spacer pads occupy 1.55 mm of this space, Thus, there would only be 0.25 mm spacing for the element to bow before it becomes constrained. Consequently, the maximum deflection possible at the mid-plane is limited to 0.25 mm. After the element becomes constrained at the mid-plane, further deflection of the element between the mid-plane and the end plate is possible. A bow calculation performed for the segment of fuel element between one end plate and the mid-plane gives a maximum element bow of 0.17 mm.

It is noted that the duration of AOO transients is very short and that a plastic bow may not be materialized, i.e. there is no bow effect once the transient is terminated.

Based on the above, for the AOO transient scenarios where the maximum temperature is limited to 450°C and the circumferential temperature gradient to 100°C for a short period of time, the fuel element bow is expected to be negligible. In addition, the end plate is well cooled and not expected to experience any temperature gradient under AOO conditions.

II.8. SHEATH COLLAPSE INTO AXIAL GAPS

The pellet stack length is a derived quantity, compatible with the necessary axial gap and other element dimensions. A minimum value of axial gap between the pellet stack and the end cap is specified to accommodate the axial expansion of the pellet stack relative to the fuel sheath. In practice, the collapsible sheath grips the pellets and the relative movement is small. The maximum tolerance is specified to prevent the possibility of the sheath collapsing into excessive axial gaps.

Under high system pressure, the sheath may collapse into the axial gaps between the fuel pellets in the fuel element. The collapse may occur elastically or by creep. The maximum axial gap (clearance) during manufacturing is approximately 3.8 mm between the fuel stack and the end caps. A larger axial gap is likely to allow axial collapse of the sheath since the fuel pellets do not provide adequate support against the coolant pressure, especially when the internal gas pressure is significantly lower than the coolant pressure. Such collapse could occur when the system is either pressurized or when the sheath is more ductile at high temperatures.

Creep collapse of the sheath is not a concern during normal operation. However, at high temperatures, sheath collapse due to creep may occur earlier than predicted by elastic collapse formulae. The key parameters for assessing the sheath collapse are as follows:

- Sheath thickness and strength;
- Axial gap between the fuel pellets and the end cap;
- Coolant pressure;
- Gas pressure within the fuel element;
- Sheath temperature;
- Time at temperature.

The critical pressure for collapse of the fuel sheath into the axial gap is calculated using the elastic instability correlation [151]. At higher temperatures, where creep collapse may occur, the ratio of the creep rate for the analysis transient/time conditions to that from sheath collapse tests is used to determine the time before collapse.

To ensure that the worst conditions are evaluated, the axial gap is assumed to be the maximum axial gap allowed in manufacturing of the fuel element, and the full gap is assumed to reside at one end of the element. In reality, once the fuel has been at power, the gap would be somewhat distributed between the fuel pellets because the pellets would have expanded to take up some of the gap during normal operation.

The fuel elements that are most susceptible to collapse are those that reach the highest peak temperature and have the lowest internal gas pressure. The highest sheath temperature is 450°C for AOO events.

Reported criteria (operating limit)

Acceptance criterion for sheath collapse into the axial gap of the fuel element based on operating limits is that there is no significant elastic or creep collapse, i.e. no acute collapse, of the sheath that is not supported by pellets. An acute collapse is defined as sheath strain greater than 5% without sheath failure.

Discussion

The driving force for the sheath collapse is the difference between the fuel element internal gas pressure and the external coolant pressure. When the internal gas pressure is high during normal operation, the possibility of collapse is reduced. In this assessment, a maximum coolant pressure of 11 MPa was used, and the internal gas pressures were between 5.0 MPa and 9.0 MPa. The duration of the AOO transient simulation considered for the analyzed AOOs is short (~60 s); however, the reactor regulating system or the shutdown system #1 would terminate the transient, resulting in even shorter time at temperature.

Elastic collapse

The sheath will not collapse elastically under AOO conditions. The pressure for elastic collapse is about 15 MPa for a 4 mm gap and is higher for smaller gaps. In AOO transient scenarios, the largest axial gap calculated is about 1 mm and the highest coolant pressure is 11 MPa. Consequently, the elastic sheath collapse in axial gaps is precluded.

Creep collapse

Sheath collapse tests were performed at pressures of 10 MPa and higher, at temperatures ranging from 340°C to 730°C and for axial gaps up to 3 mm. The collapse 'threshold' for a 3 mm axial gap was observed at 425°C. At 475°C, 'acute' collapse occurred. Acute collapse was defined as sheath strain greater than 5% without sheath failure. The duration of the tests was about 2 minutes.

Under AOO conditions, the maximum axial gap is approximately 1 mm, the maximum sheath temperature is limited to 450°C and the maximum increase in sheath strain is limited to 0.5%. These conditions are below the creep collapse threshold and clearly preclude the occurrence of the acute collapse scenario.

II.9. LOBE COLLAPSE OR FUEL SHEATH LONGITUDINAL RIDGING

The maximum permissible range of as-fabricated diametral clearance between the sheath and the pellet is defined by the necessity to avoid the formation of longitudinal ridges in the sheath. The actual diametral clearance is chosen by the manufacturer to facilitate loading of pellets into the sheath.

When a new fuel bundle is inserted in the reactor and the sheath is not yet supported by the thermal expansion of the pellets, the external pressure of the coolant will collapse the sheath onto the pellets. If the radial gap is large, relatively stable longitudinal ridges, or lobes, may be formed on the sheath. If these ridges are formed, plastic strain cycling of the sheath at the ridge (due to expansion and contraction of the pellets) may lead to low cycle fatigue failure. A small as-manufactured radial gap ensures that longitudinal ridges are not formed. The key parameters to longitudinal ridging are the sheath thickness, the radial gap between the sheath and the fuel pellets, coolant pressure, sheath temperature, and sheath strength.

Reported criteria (operating limit)

Acceptance criterion for the formation of the longitudinal ridging of the sheath based on operating limits is that the stress in a ridge of the sheath is limited below 0.2% yield strength where creep effects will not increase the height of the ridge.

Discussion

The minimum pressure necessary to cause longitudinal ridging in elements with large radial gaps is estimated to be 9.2 MPa and the pressure necessary to cause ridging is 14 MPa for a radial gap of 90 μm (about the as-manufactured gap).

The results of the AOO transient stylized analysis showed that a maximum radial gap of less than 5 μm will develop through the transient for all scenarios considered. This radial gap is far less than the acceptance criterion of 0.2% strain at the ridge. The nature of the AOO transient in terms of short time-at-temperature duration would preclude sheath degradation by this mechanism. Also, the fuel manufacturers have performed autoclave tests at 400°C for 16 hours with no indication of longitudinal ridging on subsequent inspection. As a result, there is no risk of developing longitudinal ridges due to elastic or creep deformation and, consequently, the acceptance criterion is satisfied.

II.10. SHEATH OXIDATION

The rate of sheath oxidation is highly temperature dependent. At low temperatures, sheath oxidation is very limited. As the temperature of the sheath increases, so does the oxidation rate. Sheath oxidation is not allowed to impair normal heat transfer or the integrity of the sheath.

The sheath oxidation process creates an oxide layer of ZrO_2 on the outside surface of the sheath. The oxide layer slows further oxidation as oxygen needs to penetrate the oxide layer to reach the remaining unoxidized sheath surface. The ZrO_2 is brittle and could crack and flake off the sheath resulting in the reduction of the sheath strength and thickness and providing new sites for further accelerated oxidation. At higher temperature, the increased oxidation of the sheath could impede heat transfer to the coolant. If the sheath is sufficiently oxidized, it could fail following quenching and rewet.

The amount of allowable sheath oxidation was determined considering fuel integrity and heat transfer. Examinations of the fuel sheath oxide thickness provide a guide as to what is normal and acceptable. The sheath oxide thickness is typically up to 5 μm, with oxide patches up to 15 μm at the end of the life of the fuel.

Reported criteria (operating limit)

Acceptance criteria for sheath oxidation based on operating limits include:

(a) Less than 10% of the wall thickness is oxidized (based on manufacturing specifications for the fuel sheath);

(b) The additional oxide layer on the outside surface of the sheath due to the AOO causes no more than 5°C temperature increase on the inside surface of the sheath at 100% full power.

The first criterion addresses local fuel oxidation conditions (e.g. a number of bundles or individual elements in a reactor) and total oxidation to the end of service, while the second one addresses global fuel oxidation conditions (i.e. the entire reactor core bundles). In terms of ZrO_2 thickness, the above limits are translated into

- ~64 μm (local or limited oxidation accumulated to the end of service);
- ~6.8 μm for peak power bundle (global or reactor-wide oxidation).

The ZrO_2 thickness of 64 μm was calculated assuming a nominal sheath thickness of 0.41 mm and Pilling-Bedworth ratio[21] of 1.56.

The ZrO_2 thickness values for the second criterion were calculated assuming steady state heat transfer from the sheath inner surface to the coolant. The sheath-to-coolant heat transfer coefficient and the coolant temperature were assumed to be 50 kW/($m^2 \cdot K$) and 280°C, respectively. The thermal conductivity of zirconium dioxide of ~1.9 W/(m·K) was obtained from MATPRO [28] based on experimental data covering the temperature range of 300 to 600 K. The maximum bundle power of 969 kW was also assumed for the peak power bundle.

Discussion

Given the short duration of the AOO transients and the relatively low maximum sheath temperatures (up to 450°C), only insignificant oxidation is expected during these transients.

II.11. HYDRIDE FORMATION

The hydrogen content retained during manufacture is controlled by quality control procedures that exclude hydrogenous material from the fuel or sheath. A manufacturing limit of 1 mg hydrogen gas per fuel element (excluding that which was originally contained in the zirconium alloy material) is specified [152]. Deuterium originates from the heavy water reactor coolant, and deuteriding rates are a function of the in-reactor operating conditions. No limiting level is specified for deuterium pick-up, however PHTS chemistry conditions are specified and adhered to in order to keep levels similar to those for which there is previous satisfactory experience. Deuterium pick-up is enhanced if the fuel element is defected. Quality control procedures ensure that the number of defected fuel elements that are manufactured is very small.

The H_2/D_2 levels are above the solid solubility level for zirconium at the operating temperature but have not been found to cause any operational problems. This is because the hydride is ductile and is dispersed uniformly enough that the volume expansion (of zirconium hydride versus Zr) does not cause excessive local stresses. At lower temperatures, however, zirconium hydride becomes brittle and the possibility of cracking is increased.

The amount of hydrogen absorbed by the sheath is directly related to the amount of sheath oxidation, since hydrogen is formed during the oxidation process, as follows:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 \qquad \text{(II.1)}$$

The fraction absorbed is typically less than 50% of the amount released in the oxidation process [153]. However, the conditions for sheath oxidation in steam during the AOO transient may lead to a higher hydrogen uptake. Therefore, the incremental amount of hydrogen absorbed is calculated conservatively assuming that 100% of the hydrogen released in the oxidation process is absorbed.

Reported criteria (operating limit)

Acceptance criterion for the hydride formation in the sheath based on operating limits is that there is no increase in hydride concentration beyond the maximum observed in CANDU fuel

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[21] The Pilling-Bedworth ratio is defined as the ratio of the molar volume of the metal oxide to the molar volume of the corresponding metal.

normally. A maximum hydrogen concentration of 200 μg/g (hydrogen equivalent) has been observed in fuel where operation or fuel handling has not been impaired [128, 153]. The normal hydrogen concentration is less than 100 μg/g.

Discussion

The rate of sheath oxidation and hydriding is dependent on the oxide layer thickness. An increase of 20 μg/g in hydrogen concentration for oxidized sheath is judged acceptable to avoid additional fuel handling or operational defects. For fresh fuel with little or no sheath oxidation, the extent of oxidation and hydriding during the AOO transient can be much higher without detrimental effects, since the initial hydrogen concentration in the sheath would be much lower.

Since the oxidation level at sheath temperatures below 450°C for a short duration, the increase in the hydrogen concentration in the sheath due to the AOO transient is low and acceptable.

II.12. HYDRIDE MIGRATION

Hydrogen dissolved in the Zircaloy components redistributes to cooler regions by diffusion along the thermal gradient, as well as the concentration gradient. When the hydrogen migrates to a colder area on the fuel sheath, end caps or end plates, it precipitates to form solid zirconium hydride, if the solubility of hydrogen is exceeded. This is a mechanism for failure of these components. The parameters that affect the rate and extent of hydrogen migration and precipitation are:

- Hydrogen concentration in the material;
- Temperature;
- Temperature gradient;
- Stress gradient.

The concentration of hydrogen/deuterium during normal operation is approximately100 μg/g, about 50 μg/g hydrogen from manufacture and about 50 μg/g equivalent from normal operation.

Hydrogen absorption can lead to degradation of zirconium alloys. Under thermal gradients it can form a solid hydride and fail the component, whereas under stress it can lead to delayed hydride cracking. The hydrogen solid solubility limit in the Zircaloy components of the fuel bundle (i.e. fuel sheath, end caps and end plates) is 80 to 100 ppm (corresponding to 80 to 100 μg/g) at 300°C. Hydrogen is formed as a result of the oxidation process. Thus, limiting the extent of sheath oxidation, which depends mainly on temperature, is very important.

Reported criteria (operating limit)

Acceptance criterion for the hydride formation in the sheath based on operating limits is that there is no local precipitation of hydride as a result of AOOs. During normal operation, any concentration of localized hydride is redistributed with no component failure.

Discussion

A large thermal gradient on the order of a few hundred degrees is necessary between elements within the fuel bundles in order for hydride migration/precipitation to occur. Such thermal gradient usually does occur in stagnation/stratification scenarios where the lower half of the bundle is cooled by saturated liquid and the upper half is cooled by superheated steam, as in the intermittent buoyancy induced flow (IBIF) scenario where the temperature difference between neighbouring elements is ~250°C and the average thermal gradient of 30.8°C/mm when the water level is in the mid-span. Thus, hydrides may precipitate locally to the cool surfaces at such temperature difference and temperature gradient.

Tests were conducted which had similar conditions to the IBIF scenario in terms of temperature difference and gradient. The tests were performed with Zr-2.5wt% Nb containing 300 μg/g of hydrogen. Although pressure tube material was used for the tests, the behaviour of hydrogen and hydride precipitation is nearly identical to that in zirconium alloys [154]. Hydride blisters started to form at about 28 hours in the tests. Under AOO transient conditions, the formation of hydride blisters would take longer times since the sheath temperature gradient is lower.

The stress gradient for diffusion is across the thickness of the end plate web, and not along its length where the large temperature difference and gradient develop. Therefore, the driving force for precipitation is not in the same direction as the thermal gradient. As a result, no significant effect from stress is expected on the end plate.

Hydrogen may also migrate locally to the crack tip and cause delayed hydride cracking. However, this is usually a concern during cooldown transients when the hydrogen precipitates at the crack tips. On heat-up transients, tests have shown that this problem is limited to temperatures below 210°C where crack velocities are too low to be a concern [155].

Since the fuel is well cooled under AOO transient conditions, the thermal gradient is expected to be insignificant compared to that in the IBIF scenario and the experiments, and therefore blister formation would not occur.

When comparing the timespan needed for the hydride blisters to develop (>28 hours) with the short duration of an AOO transient (minutes), any local precipitation of hydride as a result of an AOO transient can be precluded. Based on the above discussion, there will be no local precipitation of hydrides due to AOO transients experiencing sheath temperature excursions up to 450°C.

II.13. PRE-DEFECTED FUEL ELEMENT DEGRADATION

During normal operation, a fuel element may be defected and may be in the core for a period of weeks or months, depending on the size of the defect and the signal available to locate and remove the defected fuel bundle. The concern for this analysis is whether further degradation of the fuel could occur as a result of the transient, and whether this degradation is acceptable.

Considering that the duration of an AOO transient time is quite short, no significant deterioration of a pre-defected fuel element is expected, taking into account that the fuel is well cooled during the transient and the expected temperature increase is quite small. However, this fuel degradation mechanism was considered here for the sake of completeness.

Reported criteria (operating limit)

Acceptance criteria for pre-defected fuel element degradation based on operating limits include:

(a) Average O/U atomic ratio in pellets remains below 2.231;

(b) There is no significant increase in fuel centreline temperature due to thermal conductivity changes in the pellet;

(c) Increase in sheath oxidation thickness is less than 0.5 μm for the outer surface and 1.2 μm for the inner surface;

(d) Increase in sheath hydride concentration remains below that of the intact sheath. This corresponds to a maximum hydrogen pick-up by the sheath of no more than an additional 20 μg/g, i.e. 10% of the maximum concentration of hydrogen observed in fuel where operation or fuel handling has not been impaired.

A fuel centreline temperature increase of 50°C was selected as being sufficiently conservative as it bounds currently fabricated fuel. The increase in fuel centreline temperature during an AOO transient is expected to be small since the system pressure remains relatively high. The higher system pressure will lead to collapse of the sheath onto the fuel pellets, providing good contact and removing heat from the fuel to the coolant, resulting in lower fuel/fuel centreline temperatures.

Discussion

The degradation mechanisms of pre-defected fuel and the acceptance criteria were evaluated by a combination of modelling and sensitivity analyses for the aspects modelled in ELOCA. ELOCA is the CANDU Owners Group's industry standard toolset used for evaluating fuel behaviour under accident conditions.

Fuel O/U Atomic Ratio, Conductivity and Centreline Temperature Increase [Limits (a) & (b)]

The profile of the stoichiometric deviation in the pellet will be determined by reaction kinetics, equilibrium thermodynamics, and diffusion. At higher pellet temperature, towards the pellet centreline, the reaction rate will increase with temperature. However, any interaction of the coolant with the fuel sheath and pellet periphery would decrease the availability of oxygen for interaction with the fuel and decrease the equilibrium stoichiometric deviation. Thermal diffusion will drive oxygen towards the higher temperature regions of the fuel, while chemical diffusion will drive oxygen towards areas of low oxygen concentration [156].

Centreline temperature increases of pre-defected fuel caused by changes in fuel stoichiometry and thermal conductivity have been assessed in detail using a generalized MATPRO thermal conductivity model [28] and the fuel performance code ELOCA. The maximum increase in the pre-defected fuel centreline temperature was calculated to be 8°C at time zero of the AOO transient and decreased as the transient progressed due to the lower influence of the fuel stoichiometric deviation on fuel thermal conductivity at higher temperatures. This value is well below the service limit of 50°C specified above for the centreline temperature increase of pre-defected fuel.

Sheath Oxidation and Hydriding [Service limits (c) & (d)]

Pre-defected fuel sheath oxidation has been assessed and the increase in the oxide thickness for outer surface of the pre-defected fuel sheath is estimated below 0.04 μm due to an AOO transient. This value is within the specified service limit specified above for the outer surface of the pre-defected fuel sheath.

If the sheath temperature of the outer surface reaches 450°C, a bounding value of 500°C is selected to evaluate the increase in sheath oxidation for the inner surface of the sheath. This bounding value covers the current estimated increase in the sheath temperature induced by the stoichiometric deviation of the defected fuel.

The increase in the oxide thickness during a 60 seconds transient is estimated to be ~0.1 μm. In practice, the increase needs to be lower than 0.1 μm, because the shutdown system #1 will trip long before the 60 seconds simulation time. This value is within the service limit specified above for the inner surface of the pre-defected fuel sheath.

The amount of hydrogen pick-up due to the AOO transient can be estimated assuming all of the oxide thickness increase is converted into hydrogen that is absorbed by the sheath. This is very conservative since not all the produced hydrogen will be taken by the sheath. The amount of hydrogen estimated is negligibly small and much below the service limit specified above for hydrogen pick-up.

APPENDIX III. IDENTIFICATION OF PHYSICAL BARRIERS FOR HORIZONTAL CHANNEL-TYPE PHWRS DURING ACCIDENT CONDITIONS

Specific safety functions required for the fuel and fuel channels in horizontal channel-type PHWRs during accident conditions include:

(a) Control the release of the fission products such that the public dose limits are complied with;

(b) Ensure that the removal of the heat generated by fission and fission product decay is not impeded during the short term and long term phases of the accident sequence. This function is necessary to fulfil the previous safety function.

Safety function (a) above can be divided into the following logical paths:

- Fission products are generated by the fission process and are embedded in the fuel matrix;
- Fission products are released from the fuel matrix to the pellet-to-sheath gap;
- Failure of the sheaths allows the fission products to migrate into the PHTS;
- Failures of other reactor components may allow the fission products to be released into the containment.

The following physical barriers[22] relevant to fuel and fuel channels can be identified from this division:

- Fuel matrix retains and/or delays the release of fission products to the fuel-to-sheath gap;
- Fuel sheath retains the fission products inside of the fuel elements;
- PHTS confines the fission products inside the pressure boundary.

These barriers follow the defence in depth concept and the same set was identified for the LWRs.

Safety function (b) above can be divided into the following logical paths:

- Heat is generated by the absorption of the fission fragments energy in the fuel matrix and by decay of fission products;
- The generated heat is transferred from the fuel matrix to the fuel sheath[23];
- Heat is removed from the fuel elements/bundles by the coolant of the PHTS (and emergency core cooling) and transferred to the secondary side, and directly into containment via postulated breaks;

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[22] A physical barrier is a physical boundary which performs the function of preventing, delaying or attenuating the unwanted movement of radioactive material as well as the physical boundary that provides shielding in operational states and/or accident conditions.

[23] Note that a small amount of heat can be transferred directly to other structures by gamma radiation.

- Depending on the efficacy of these mechanisms, some heat may be transferred from the fuel elements/bundles to the pressure tubes; in which case
  - (i) Heat is transferred from fuel to the pressure tubes and to the calandria tubes,
  - (ii) Heat is transferred from the calandria tubes to the moderator (severe core damage accidents are not considered).

The following physical barriers relevant to fuel and fuel channels can be identified from this division:

- Fuel sheath participates in heat transfer and preserves the bundle geometry enabling heat removal;
- PHTS removes the heat generated by the fuel bundles
  - o Pressure tubes, as part of the PHTS, contribute to axial heat removal from bundles through the channel flow;
- Calandria tubes, as part of the fuel channel assemblies, contribute to radial heat removal from the pressure tube to the moderator.

Therefore, physical barriers considered as part of or affecting the fuel design include:

- Fuel matrix – fission product retention/delay;
- Fuel sheath – fission product retention and fuel coolability;
- PHTS – fission product retention and fuel coolability
  - o Pressure tubes – fuel coolability;
- Calandria tubes – fuel coolability.

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ANNEX I. FUEL ACCEPTANCE CRITERIA DURING OPERATIONAL STATES – A DESIGNER'S PERSPECTIVE

In the CBV, as in the traditional approach, verification experiments, analyses, and assessments are performed to demonstrate that the fuel bundle can meet each of the design requirements imposed by plant design and operation without unacceptable damage to the fuel bundle or to its surroundings. This is done for the full range of credible combinations of design specifications for manufacturing and for operation. Each verification experiment, analysis, and assessment consider a set of related design requirements and determines the numeric values of the fuel design performance parameters associated with pertinent damage mechanisms under the relevant operational conditions.

In the traditional approach, numeric values of these performance parameters constitute "limits" within which the fuel bundle design has been verified to satisfy its design requirements, and within which it must be used; in other words, limits beyond which the fuel may still not be damaged but for which the necessary analyses and validation tests have not yet been performed.

In the CBV approach, numeric values of the performance parameters are compared to pre-determined numerical limits ("acceptance criteria") associated with the damage mechanisms in order to demonstrate that the design requirements are met and that neither the fuel nor its surroundings are damaged. To the extent practical, the acceptance criteria are developed from 'first principles', i.e. from known material damage mechanisms in operational states and from fundamental applications of engineering principles. Their numeric values are obtained by subtracting prudent margins from material damage limits.

To a large extent, the CBV approach establishes acceptance criteria that are independent of specific design features; however, numeric values of some material properties may differ from one design to another. It also allows a designer to quantify the margin(s) by which the fuel avoids damage from each damage mechanism.

Compared to the traditional approach, fuel verification using mature CBV would (a) be quicker; (b) facilitate enhanced comprehensiveness; and (c) enable easier licensing in some international jurisdictions.

I-1. INTRODUCTION

The main text of this publication provides the perspectives of reactor operators about fuel acceptance criteria for current CANDU fuels, at current nuclear power plants, during current operating conditions or ones that are reasonably similar.

A fuel designer sometimes encounters needs beyond that – for example to design fuel for a new, significantly different reactor and/or for new operating conditions that differ significantly from current ones, as explained in more detail in Section I-2. To satisfy such needs, fuel designers at Atomic Energy of Canada Limited (AECL) have, in the early 21st century, further evolved the traditional approach for design and verification of CANDU fuel and called it "criteria-based verification" (CBV). This Annex I describes that approach.

The detailed descriptions in this Annex are focussed towards normal operating conditions; nevertheless, the approach described here, and the resulting criteria also apply to AOOs.

The current definition of AOO revolves around the frequency of occurrence of identified events. For fuel damage mechanisms, however, the local temperature is an important driver. The specific damage mechanisms covered in this publication are those that are active at temperatures near those experienced during normal operations. If an AOO were to drive the fuel to significantly higher temperatures, then additional damage mechanisms may also need to be considered where appropriate, for example beryllium braze penetration.

Although the CBV approach was developed for a significantly advanced fuel design, it can also be applied to any evolution of current designs and/or operating conditions – small, medium or large.

Some aspects of the CBV approach, and illustrative examples of its implementation, have already been described in several publications [I-1 – I-7]. The main features of this approach are summarized in Section I-3 of this Annex.

Different jurisdictions and organizations use a variety of words for the activity that confirms whether or not a proposed design is adequate for its intended use, such as 'verification', 'validation', or 'qualification'. For simplicity, this Annex uses the same term that is used in existing literature on CBV [I-1 – I-7]: 'verification'. For the same reason, this Annex labels the related criteria as 'acceptance criteria'. Thus, in this Annex, 'acceptance criteria' have the same meaning as 'safety criteria' do in the main text of this publication. Various other terms are also used in the literature to describe essentially the same thing, such as 'design criteria'. This Annex considers them interchangeable.

This Annex starts with an illustration of a relatively broader need of a fuel designer and some of the associated challenges. It then provides a checklist of credible damage mechanisms and an illustrative list of related acceptance criteria at a high level. Explanations and key features of above are also included.

I-2. ILLUSTRATIVE NEED

One role of a fuel designer is to modify the fuel design or its operating conditions, sometimes significantly, to address new challenges. This can include designing fuel for a new reactor.

The CBV approach was initially developed during the design of the Advanced CANDU Reactor (ACR). The reactor's designers called for fuel to achieve an average discharge burnup of about 20 GW·d/tU. This is about 2-3 times the current average discharge burnup of CANDU fuels. Also, ACR's coolant was hotter than the current CANDU reactors.

The current fuel designs were judged incapable of meeting this challenge satisfactorily; therefore, a new fuel design was crafted for this application. In many performance areas, however, insufficient operational experience was available to reliably confirm whether or not this new design would be consistently satisfactory in all the intended operating conditions.

One such aspect was the strength of the end plate and of the assembly weld during discharge from the reactor. One way to confirm the strength of the ACR bundle would have been via an out-of-reactor 'type test' using a string of prototype unirradiated fuel bundles. Unirradiated Zircaloy has considerable ductility; therefore, it can survive large plastic strains that were expected during discharge of ACR fuel. In the reactor, however, the fuel is discharged after irradiation, and ductility of Zircaloy decreases considerably with fast fluence [I-8]. Therefore, in principle, the strength test for the ACR fuel string would ideally be performed on a string of irradiated high burnup ACR prototype bundles. That would have needed considerable time and expense – see Section I-3.3 for a more detailed discussion.

Several other similar situations were also encountered during the design of ACR fuel – mostly due to combinations of longer in-reactor residence and hotter coolant. These included aspects such as additional oxidation, additional deuteriding, more demanding power ramps, and additional creep droop. Traditional methodologies for the verification of these aspects were judged to be too time consuming, sometimes quite expensive, and/or not capable of providing the degree of comprehensiveness that high burnup ACR fuel designers preferred (e.g. for creep droop) – see Section I-3.3 for a more detailed discussion. Therefore, the traditional approach was evolved where needed.

I-3. CRITERIA-BASED VERIFICATION

This section, provides an overview of the process that is employed in criteria-based verification for normal operations and examines its similarity with approaches that are used in verifications of some other aspects of fuel design.

I-3.1. Overview

Criteria-based verification (CBV) starts with a systematic, comprehensive checklist of all credible damage mechanisms. The purpose of the checklist is to minimize the possibility of inadvertently overlooking a damage mechanism that may be important in the new circumstances but is outside the focus of the designer's attention because, for example, the current fuel has not failed recently from that mechanism in an operating nuclear power plant. Section I-4.2 provides an illustrative checklist. To this 'standard' list of damage mechanisms, it would be necessary to add further damage mechanisms, if appropriate, to reflect new circumstances pertinent to the new design and/or operating conditions.

During an initial stage of the design process, all damage mechanisms in the above checklist need to be at least considered, and a conscious decision needs to be made for each. For example, one possible decision may be to quickly dismiss that damage mechanism outright because its severity in that specific application is judged to be nowhere near damaging. Another possible decision may be to flag that damage mechanism for more detailed evaluation.

For all relevant damage scenarios, each pertinent damage mechanism is evaluated against a corresponding acceptance criterion. The evaluations can be carried out using engineering judgements, tests, analyses, or a combination thereof.

To the extent permitted by available technology, an acceptance criterion is built from material damage limit minus a minimum acceptable margin, i.e. from first principles. Section 1.5 of this Annex provides more complete descriptions of these and other key features of the acceptance criteria.

Sometimes, total damage from a damage mechanism comprises a combination of a series of damages that are accumulated during multiple events in the life of a fuel bundle. For example, fatigue damage can accumulate during lateral flow induced vibrations in the reactor, and during repeated changes in power, and also during passage of the fuel bundle through the crossflow region. Such cumulative damages need to be summed from individual components as appropriate.

For this purpose, evaluations are made for the entire life of the fuel during which fuel integrity is required. This means that ‘upstream damages’ – from fuel’s transportation to the reactor – as well as ‘handover’ requirements for downstream operations after discharge from the reactor would also be considered. However, the scope of this Annex is limited to fuel’s life in the reactor during operational states.

CBV quantifies the margins to failure in terms of damage parameters. This is useful for exploring the possibility of extending the fuel design and operating envelope, or for estimating the remaining margin to accommodate significant deviations from the fuel’s traditional operating envelope.

I-3.2. Similarities with other approaches

For some damage mechanisms, CBV is very similar to the traditional approach to CANDU fuel. For example, in the traditional design verifications, avoidance of pellet melting is confirmed by checking pellet temperature against the melting point of the pellet. The latter is a material damage limit. CBV is built on the same principle. In addition, CBV extends this traditional approach to all damage mechanisms, and provides a systematic framework for doing so, for example by making an explicit provision of minimum acceptable margins. Section I-5 describes these aspects in greater detail.

CBV also shares many features with the approach that is currently used in accident analyses of CANDU fuel.

CBV is also very similar to the approach that has already been used for operational states in LWR and advanced gas cooled reactor fuel designs for a number of years. In a broad sense, the stronger elements of the LWR approach have now been adapted to CANDU fuel, albeit with appropriate adjustments to reflect CANDU conditions. As well, in some aspects, the CBV approach also reflects increased knowledge that is now available.

I-3.3. Advantages of mature criteria-based verification

Full application of the CBV approach would need numeric values of material damage limits in a number of areas. The CANDU fuel industry already has a few, for example melting point of UO_2. In a few other subject areas, the LWR community has already accumulated significant generic research data on UO_2 and on Zircaloy that can also be applied to CANDU fuel. Nevertheless, in some other areas, additional CANDU-specific research would be helpful, for example to refine some material damage limits to conditions specific to CANDU reactors. Also, it would be considered adding, enhancing and validating a few additional features in computer codes. After initial one-off investments of this nature, mature CBV can be expected to yield the advantages listed below.

I-3.3.1. Speed and Cost

Some verification experiments require very long elapsed times. As an illustrative example, to perform a strength test on a string of fully irradiated high burnup bundles, at least twelve such bundles would need to be irradiated in a first step. Even more bundles would need to be irradiated if the full spectrum of credible combinations of fabrication and operational variabilities were also to be investigated.

First, when the new fuel design and/or coolant conditions differ significantly from the current ones, it becomes very hard to build the necessary consensus to test the new fuel at appropriate conditions in an operating nuclear power plant. The alternative would be to consider irradiating the test bundles in a research reactor that can accommodate CANDU fuel bundles. But even when domestic research reactors are functional, they usually offer very few locations where the neutron flux is high enough to mimic fuel in the core of a nuclear power plant at high power. Therefore, the twelve (or more) fuel bundles that would be required for the test would need to be irradiated in series; that would require a long time. Further, even if appropriate capability and capacity can be found in overseas research reactors, irradiations to ACR burnups would be very expensive.

Second, after irradiating the prototype fuel bundles, the type test would need to be done in a shielded facility such as a hot cell. It may be a challenge to find an existing shielded facility that can accommodate a full string of CANDU fuel bundles in a horizontal orientation. Building appropriate new facilities would be time consuming as well as expensive.

In contrast, numerical analyses can usually be done relatively quickly and at significantly lower cost. Overall, selection of experiments versus analyses would need to be done on a case-by-case basis after considering their relative merits in each specific situation.

I-3.3.2. Comprehensiveness

For some aspects of fuel verification, the traditional approach poses considerable challenges. As an illustrative example, fuel is irradiated horizontally in CANDU reactors; therefore, gravity activates some unique processes such as droop. Operational feedback to date reveals that this aspect is satisfactory addressed in current fuel designs and reactors. However, the droop can be expected to be higher in ACR fuel because (a) its outer elements have smaller diameter, and (b) its high burnup and hooter coolant would cause larger creep. The creep droop would be absent in a vertical irradiation in a research reactor, but it can be covered through numeric analyses during CBV. Similar other illustrative examples can also be constructed. Thus, comprehensiveness of fuel verification can be enhanced through CBV.

I-3.3.3. Licensing

CBV is better aligned with current practices in the LWR and advanced gas cooled reactor fuel industries; therefore, it facilitates easier licensing of CANDU fuel in some international jurisdictions.

I-4. DAMAGE MECHANISMS

This section describes the salient features of damage mechanisms, lists credible damage mechanisms in CANDU fuel and explains them.

I-4.1. Features

A ‘damage mechanism’ is a process that, if excessive, would render the fuel unsuited to fulfil its design requirements. Acceptance criteria are pertinent numeric limits that are used to determine whether or not the fuel or its surroundings are damaged.

In this context, ‘not damaged’ means that the fuel elements do not fail, that fuel bundle dimensions remain within operational tolerances, that functional capabilities are not reduced below those assumed in safety analyses and/or specified in design requirements, and that the fuel bundle maintains its structural integrity. ‘Fuel element failure’ means that the fuel element leaks and that the final fission product barrier in the fuel – the sheath – has therefore been breached.

In addition, it is well known that in some adverse situations, fuel has a potential to damage the pressure tube, for example through fretting and wear. AECL's designers of pressure tubes have historically assigned an allowance for such purposes, traditionally called the 'wear and corrosion allowance'. At AECL, it has historically been the responsibility of the fuel designer to confirm that all credible interactions of the fuel with the pressure tube are within such allowances. For this reason, damage mechanisms related to interactions between the fuel and its surroundings are also included in the checklist in Section I-4.2.

A variety of techniques are available to a designer to avoid damage. They can be implemented during any of the three stages of a fuel's life – (a) during design, for example through choices for materials, geometries or dimensions; (b) during fabrication, for example by manufacturing fuel within the range of specifications set by the designer; and (c) during operation, for example by operating the fuel within parameters set by the designer. To avoid significant damage, a designer will choose whichever combination of above is the most effective, practical, safe, and economic. Three illustrative examples are provided:

- Power ramp damage is currently avoided through three key steps: (i) by specifying a CANLUB layer of a certain thickness and composition; (ii) by specifying the allowable combinations of power ramps, powers, and burnups; and (iii) by devising practical fuelling schemes that achieve (ii) above. In addition, control of pellet density has also been suggested. Specifications for all above are crafted during design. Step (i) above then implemented during manufacturing and the next two are implemented during operation;

- Deuteride/hydride damage is currently avoided mainly by limiting the amount of initial hydrogen within a fuel element, and by controlling the pH of the coolant. Both steps are identified and specified during the design phase. The first is implemented during manufacturing, the second during operation;

- Contact between a deformed fuel sheath and the pressure tube is avoided by strategically locating and sizing the bearing pads, and by controlling several other dimensions and clearances. These aspects are implemented during the design phase by controlling the geometry and dimensions of the fuel, and during manufacturing by building within those specifications.

In the above examples, the designer has controlled the above damage mechanisms through a combination of steps during all three stages of a fuel's life, i.e. design (e.g. geometry), manufacturing (e.g. hydrogen content), and operation (e.g. fuelling scheme).

For the above reason, ACR's fuel designers considered it their responsibility to systematically identify and limit all credible damage mechanisms during all stages of a fuel's life.

Although the damage mechanisms listed in Section I-4.2 were identified during design studies of ACR fuel, they are considered generic to fuel designs that are relevant to current CANDU fuel designs.

Sometimes a designer is tempted to consider features that may introduce new potential damage mechanisms. As an illustrative example, in the past AECL's researchers have investigated inserting thin discs between neighbouring pellets [I-9]. This provides an additional path to conduct the heat away to the coolant and reduces pellet temperature. In such situations, appropriate new damage mechanisms need to be added to the checklist.

In several cases, a number of individual processes and/or operational parameters contribute to a given damage mechanism. For example, melting of the pellet – a damage mechanism – is affected by element power, thermal conductivity of the pellet, heat conduction, etc.

Conversely, sometimes a given operational parameter and/or a process affect a number of damage mechanisms. For example, element power (and its change), an operational parameter, can contribute to a variety of damage mechanisms including central melting, gas overpressure, and stress corrosion cracking. Likewise, bundle geometry change, a process, can potentially contribute to the following damage mechanisms: degradation of heat transfer between the fuel element and the coolant; excessive interaction loads along the fuel string; excessive interference with interfacing equipment; and fuel bundle jamming. Nevertheless, neither element power, nor bundle deformation, are damage mechanisms (power is an operational parameter, deformation is a process).

The primary purpose of the checklist in Section I-4.2 is to help probe that no known damage mechanism is inadvertently ignored during the design process. Inclusion of a damage mechanism in this checklist does not mean that that specific damage mechanism will necessarily be at a dangerous level in a given fuel design for given operating conditions. Rather, its inclusion merely means that the potential impact of that damage mechanism needs to be at least considered at some stage of fuel design.

I-4.2. Checklist

Based on first principles – and confirmed later through a comprehensive search of CANDU fuel defect experiences [I-3] – ACR's fuel designers have identified the damage mechanisms listed below [I-1].

The checklist below is at a level similar to the one in U.S. NRC's Standard Review Plan for Fuel System Design [I-10], however, the contents of this Annex have been tailored towards CANDU fuels. The checklist in [I-10] lists 21 damage mechanisms for all LWR plant states – normal operations, AOOs and DBAs; a majority of them apply to operational states. In comparison, this Annex lists 14 damage mechanisms for CANDU fuel that focus exclusively on operational states.

The checklist categorizes the damage mechanisms into three major groups: thermal integrity, structural integrity, and considerations of compatibility between the fuel and the interfacing systems.

- (a) Thermal integrity
  - (i) Overheating of the pellet,
  - (ii) Overheating of the sheath and other structural materials;
- (b) Structural integrity
  - (i) Element internal gas pressure,
  - (ii) Stress corrosion cracking,
  - (iii) Static mechanical overstress and/or overstrain,
  - (iv) Mechanical rupture due to impact loads,

(v) Fatigue,

(vi) Loss of control of geometry,

(vii) Primary deuteriding/hydriding,

(viii) Oxides, crud and deposits;

(c) Compatibility with surroundings

(i) Excessive interaction loads along the fuel string,

(ii) Excessive interference with interfacing equipment,

(iii) Wear,

(iv) Crevice corrosion.

The following Sections I-4.3 to I-4.5 describe the damage mechanisms listed above.

I-4.3. Thermal integrity

I-4.3.1. Overheating of the pellet

If the pellet melts, the resulting volumetric expansion of the pellet may potentially push the sheath past breaking.

CANDU experience so far is that molten UO_2 stays confined to its original location. Nevertheless, if it does flow away, it can potentially reach the sheath or the end cap and melt them too.

Although many factors determine pellet temperatures, a primary driver is the power produced in the fuel element.

Some local variations can impact the above, for example, end flux peaking can potentially create a temporary but significant local peak in pellet temperature in some CANDU reactors. A CANDU fuel string consists of a number of short bundles in a channel. One consequence is that local peaks of flux are created at the ends of fuel bundles, which in turn lead to local peaks of pellet temperature near the ends of pellet stacks. A variety of factors determine the magnitudes of the temperature peaks. Their detailed discussion is beyond the scope of this Annex, however, Refs [I-11, I-12] provide additional information. Under some situations during operational states in some reactors, the end flux peak could be considerable [I-11], and for a short duration in some reactors, it can potentially even occur in a high power location.

I-4.3.2. Overheating of the sheath and other structural materials

Overheating can quickly degrade the strength of the sheath material. It can also lead to rapid oxidation, crevice corrosion, and creep. Excessive bowing can occur if the overheating is localized, large and circumferentially non-uniform. Excessive local bowing can potentially damage a neighbouring fuel element and even the pressure tube. In the extreme case, the sheath could melt.

At joineries, specifications frequently permit some degree of incomplete bonding. The unbonded areas can impede heat transfer and increase local temperatures. Local eutectics may form in the brazed region; this may alter some local material properties. In the extreme, the appendage may no longer stay attached to the sheath.

The current practice is to avoid dryout at power during normal operation. This is sufficient to prevent overheating of the sheath.

Although this is a sufficient condition to avoid overheating the sheath, it is not necessarily necessary. A more nuanced mechanistic approach is also possible.

During AOOs, limited dryout for a short duration is currently permitted.

Dryout is affected primarily by thermal and hydraulic conditions in and surrounding a fuel element. They, in turn, are affected by a number of other performance parameters as well, including fuel deformations. Therefore, the latter also needs to be considered where appropriate.

I-4.4. Structural integrity

I-4.4.1. Element internal gas pressure

Fission gas is generated within the grains of UO_2, and some of it migrates to the ‘open’ space between the pellets and the sheath. If the fission gas release is large, internal gas pressure can exceed the coolant pressure. In conjunction with a corrosive internal environment, local hydrides and local oxides, an excessive internal overpressure in a fuel element can potentially cause cracks at two locations of stress concentrations: (a) at the sheath/end cap junction, and (b) at the junction of the sheath and the bearing/spacer pad.

Internal gas overpressure can potentially also cause excessive outward creep of the sheath, which in turn can thin the sheath. If the thinning is excessive, the sheath can crack. Excessive creep in fuel elements can potentially also affect the thermohydraulic conditions in the surrounding coolant, thus potentially also affecting critical heat flux.

For the above reasons, it is good engineering practice to limit the maximum gas pressure that is allowed in the fuel element. In the current practice, this is achieved by providing sufficient ‘empty’ space within the fuel element, and by controlling the power and the burnup of the fuel element.

I-4.4.2. Stress corrosion cracking

During irradiation, Zircaloy is embrittled by fast neutrons, hydrides, and oxides. During subsequent power ramps, the embrittled Zircaloy can experience high stresses and strains in the presence of a corrosive internal environment. This combination can potentially crack the fuel element via stress corrosion cracking (SCC). The most vulnerable locations are circumferential ridges, sheath/end cap junctions, sheath/pad junctions, and pellet chips if any.

This can be avoided by keeping the local combinations of stresses, strains, corrosives, neutron embrittlement, hydrides, and oxides below the level that causes SCC in the sheath material.

I-4.4.3. Static mechanical overstress/overstrain

During several situations such as refuelling, structural components of the fuel bundle can potentially be exposed to relatively high loads and/or to relatively sparse supports, leading to a potential for static mechanical overstress/overstrain.

A variety of loads and processes affect the magnitudes of resulting stresses and strains. These include, for example, coolant drag load, coolant pressure, internal gas pressure, thermal expansion and contraction, pellet densification, fission product swelling, elastic stresses and strains, plasticity, creep, pellet cracking, and local stress concentrations.

Different combinations of loads and supports result in different magnitudes of peak local stresses and strains at various locations. Some illustrative locations of high local stress/strain concentrations are circumferential ridges, sheath/end cap junctions, sheath/pad junctions, end cap/end plate junctions; rib/ring junctions; axial gaps between the pellet stack and the sheath; and longitudinal ridges.

At the various joineries in CANDU fuel, the unbonded areas (allowed by specifications) also affect the local levels and concentrations of stresses.

The resulting stresses and strains are often quite complex and frequently well into the plastic range. For the latter reason, some fuel designers favour using strains rather than stresses to ascertain damage from static loads.

To avoid potential failures due to overstress/overstrain, a balance is maintained between the bundle's strength and applied loads.

I-4.4.4. Mechanical rupture due to impact loads

Situations such as refuelling and/or start/restart can sometimes require a fuel bundle to travel from one location to another where it hits a stationary fuel string. This can potentially impose significant impact loads on the fuel bundles. This usually occurs after some amount of embrittlement has already occurred in the Zircaloy, for example due to fast neutrons, hydrides, and oxides. If excessive, this combination can potentially damage the stationary bundle and/or the travelling bundle.

This is currently controlled by limiting the impact velocity to an acceptable level.

I-4.4.5. Fatigue

Fuel can experience cycles of stresses and strains due to a variety of repetitive loads such as flow induced vibrations, axial resonance due to acoustics, crossflow, pressure and temperature cycles in the coolant, and repeated manoeuvres in power. These can expose the fuel to potential failure via fatigue.

In parallel, depending on which part of the fuel is so affected, the corresponding strength could have been degraded from effects such as corrosive environment, neutron embrittlement, hydrides, and oxides.

Prior to the irradiation, fuel may have also accumulated some fatigue cycles during transportation from the manufacturer to the reactor. If so, there would be a corresponding reduction in the number of cycles that are available to resist alternating loads during fuel's residence in the reactor.

This damage mechanism is avoided by ensuring that the cumulative lifetime fatigue loads are below the pertinent fatigue strength.

I-4.4.6. Loss of control of geometry

Fuel designers are sometimes tempted to reduce the diameter of a fuel element or to increase its length. Some illustrative examples are: the CANFLEX fuel bundle [I-13] in which some fuel elements have relatively smaller diameter, the HAC fuel bundle [I-14] in which some fuel elements have even smaller diameter, and the CARA fuel bundle [I-15, I-16] which is twice the usual length of a CANDU bundle. Excessive changes of such nature can eventually risk introducing mechanical buckling into the fuel design, which can potentially deform the bundle into an uncontrolled, unknown shape. Therefore, the designer needs to check for this possibility and if necessary, mitigate it.

I-4.4.7. Primary deuteriding/hydriding

The fuel sheath can pick up hydrogen from the CANLUB layer and the fuel matrix. In addition, it can pick up deuterium from the coolant. Over time, the hydrogen and the deuterium can migrate within the fuel bundle to locations that are relatively cooler or at higher tensile stresses.

The migration can concentrate the overall deuterium and the hydrogen into a few locations. If excessive, this in turn can result in precipitation of deuterides and hydrides. Excessive local deuterides and hydrides can reduce the local ductility of Zircaloy, rendering it less capable of carrying its loads.

In the above context, relatively cooler locations are outer surface of the sheath; end cap region; assembly weld; and end plate. If a fuel string is supported by a latch, the end plate and the assembly weld can become locations of relatively higher long term tensile stress.

Primary failures due to deuterides/hydrides can be prevented by limiting the local concentrations of deuterides/hydrides to acceptable levels.

I-4.4.8. Oxides, crud and deposits

Burnup in natural uranium fuel is usually low. Also, CANDU reactors generally maintain excellent control of coolant chemistry. These features mean that sheath oxidation and crud are both usually very low.

An atypical incident did occur in the Pickering reactor during 2008–2012: Coolant chemistry was affected for a brief period which led to significant deposits of crud on the fuel [I-17]. Corrective actions restored the required balance.

During the design of ACR fuel, (a) fuel burnup was significantly higher and (b) coolant temperature was also higher. Sufficiently significant operational experience was not available for this combination. Therefore, oxidation and crud were included in this checklist.

References [I-18 – I-20] discuss consequences of oxides, cruds, and deposits. While some consequences differ between oxides, crud and deposits, others are largely similar. Due to the significant overall similarities in their impacts, they are listed together in this section.

- Heat transfer – Excessive levels of oxides, crud and deposits can impair heat transfer between the sheath and the coolant. This can aggravate other damage mechanisms that are driven by temperature, as listed in other parts of this section. Some illustrative examples are: accelerated fission gas release and higher internal pressure; and circumferentially non-uniform temperature resulting in bowing;

- Flaking/spalling – If the oxide is too thick, it can flake/spall and create radioactive debris in the primary heat transport. Crud and deposits are generated not in the fuel but elsewhere in the reactor; therefore fuel is not the original source for that particular debris;
- Deuteride/hydride lens – If the oxide, crud and deposits flake in a non-uniform manner, local gradients of temperature are created in the underlying sheath. That in turn causes migration of deuterium/hydrogen to cooler areas. This concentrates deuterium/hydrogen at local cold spots and can form local deuterides/hydrides. They have been observed in LWR fuels; LWR industry calls them 'hydride lenses'. The sheath can fail at a hydride lens due to reduced ductility;
- Through-wall hole – In the limit, through-wall oxidation will create a hole in the sheath through which fission products can escape;
- Loss of load carrying material and ductility – Zircaloy is lost during oxidation. Also, material adjacent to oxide has high oxygen content which reduces its ductility. These aspects reduce the capacity of the component to carry load;
- Flux distortion – LWR operators have reported that heavy local deposits of crud can change the local absorption of neutrons, hence distort the flux shape and power distribution. These can have power related consequences.

In summary, in current PHWRs with natural uranium fuel, except for an atypical incident in Pickering during 2008–2012, oxidation and crud are usually insignificant. Nevertheless, they are included in this checklist mainly to cover more demanding duty cycles such as higher burnups.

I-4.5. Compatibility with surroundings

I-4.5.1. Excessive interaction loads along the fuel string

In the reactor, the fuel string can potentially expand thermally at the operating temperature. If the corresponding cavity in the fuel channel does not provide sufficient length to accommodate this expansion, failure can result.

This is usually addressed by providing sufficient space in the channel's cavity.

I-4.5.2. Excessive interference with interfacing equipment

The as-built bundle usually undergoes a variety of deformations during irradiation, such as diametral expansion/contraction, bowing, droop, sag, doming, creep or parallelogramming. This can affect the clearances with equipment that interfaces with the fuel bundle (e.g. pressure tube, fuelling machine). The altered clearances are not allowed to be below those that have been assumed in safety analyses, for example in calculations of heat transfer coefficients or critical heat flux.

Second, during loading into and unloading from the reactor core, the fuel bundle needs to navigate narrow spaces and sometimes even bends. Thus, there is a need to ensure that the as-built fuel bundle as well as the deformed fuel bundle would pass through the fuel channel and through the fuel handling equipment without jamming.

Third, to avoid local overheating, the hot sheath of a deformed bundle is not allowed to contact the pressure tube or even a neighbouring sheath.

These aspects are usually addressed by tight control of bundle's dimensions, flexibility and deformations.

I-4.5.3. Wear

Wear in the fuel bundle and in the pressure tube can be caused by a variety of sources (e.g. sliding, erosion, fretting and vibrations). The resulting marks, scrapes and grooves in the pressure tube are collectively called 'flaws'. Flaws due to wear have been found in examinations of fuel bundles and of pressure tubes [I-21].

Fuel bundles slide into the fuel channel. This can cause sliding wear in the bearing pads and in the pressure tubes.

Erosion can occur due to the high velocity of the coolant, sometimes exacerbated by debris in the coolant.

Section I-4.4.5 has already listed drivers that can vibrate the fuel element and the bundle. Fretting due to vibrations can potentially cause wear in spacer pads, bearing pads and the pressure tube.

Vibrations can also 'rock' the fuel bundle, which results in sliding marks.

Excessive wear can lead to three potential consequences, as described below.

- Excessive fretting of spacer pads can potentially rub the corner of a spacer pad into the adjacent sheath and damage it. Such damage has indeed been observed;
- Flaws can promote delayed deuteride/hydride cracking in pressure tubes. Fretting flaws are indeed currently evaluated for initiation of delayed deuteride/hydride cracking in pressure tubes by using a procedure based on fitness-for-service guidelines [I-21];
- Wear can reduce the ability of the pressure tube to carry loads. Also, a sharp flaw can increase the local concentration of stress in the pressure tube. A flaw that is initially small can later grow; therefore growth of flaws is monitored.

Amounts of wear (including fretting) are usually controlled by limiting their driving forces to acceptable levels.

I-4.5.4. Crevice corrosion

Crevice corrosion can potentially occur in crevices such as between bearing pads and the pressure tube or between bearing pads and the sheath.

Crevices restrict the flow of coolant. The reduced amount coolant can partially boil off locally, which in turn can increase the local concentration of corrosives such as Li. This, in turn, can potentially accelerate the local corrosion of Zircaloy.

To date, crevice corrosion has occurred at two locations:

- Crevice between the bearing pad and the pressure tube: In this region, the local temperature can reach of the order of 350°C. This can elevate local concentrations of lithium hydroxide, causing accelerated local oxidation of the zirconium alloys. In operating reactors, local corrosion has been observed in the crevice between bearing pads and pressure tubes [I-22]. Subsequent test programs have confirmed this mechanism [I-22]. In some CANDU reactors, pressure tubes are periodically monitored for crevice corrosion marks. Unusual growth, if any, is investigated further;

- Crevice between the bearing pad and the sheath: This region can experience temperatures of the order of 400°C. In fuel simulation tests, through-wall penetration of the sheath was observed in only a few days. Through-wall hole in the sheath has also been observed in the reactor.

Crevice corrosion is limited by controlling local temperature, geometry, and lithium concentration.

I-5. FORMULATIONS OF ACCEPTANCE CRITERIA

Acceptance criteria are used to confirm whether or not damage parameters are within acceptable levels.

The fuel acceptance criteria described in this Annex are not an outcome of the fuel design verification process but are upfront inputs into this process. Defining fuel acceptance criteria is now one of element of the process by which requirements of CSA N286:1012 [I-24] are met.

The objective is to establish, to the extent practical, fuel acceptance criteria which are independent of specific fuel design features. They need to be established for all credible fuel element and fuel bundle damage mechanisms that may be activated during operational states.

This section first provides an overview of the process that is used to establish acceptance criteria; then an illustrative numeric example is given for one specific acceptance criterion; and then the key features of acceptance criteria are described.

I-5.1. Hierarchy

A variety of parameters and processes are usually involved in a given damage mechanism, and they have a hierarchy which plays an important role in establishing the 'ideal' acceptance criterion for any damage mechanism.

Figure I-1 illustrates this concept for pellet melting.

At 'Level 1' are parameters that are controlled directly by humans, namely:

- Design aspects – These include choices of materials, geometries, dimensions, etc.;
- Fabrication aspects – These include manufacturing techniques used to implement the designer's specifications, for example for pellet density, filling gas, diametral clearance, and integrity and completeness of bonding between components;
- Operational aspects – These include control of pertinent operating conditions to within the designer's specifications, for example element power and coolant conditions.

The above parameters then interact with each other in the reactor through processes such as radial distributions of neutron flux, end flux peaking, heat generation, heat conduction, or heat transfer coefficients. These processes are labelled as 'Level 2' in Fig. I-1.

The processes of Level 2 result in the two key mechanistic parameters that determine melting: temperature and melting point. Figure I-1 labels them as ‘Level 3’.

Their interaction determines whether or not the damage will occur, i.e., in this specific example, whether or not the local material will melt. This is labelled as ‘Level 4’ in Fig. I-1 – the damage mechanism.

In principle, an acceptance criterion can be specified through several parameters, individually or in combination. To the extent practical, a fuel designer would ideally prefer to formulate it at Level 3, so that (a) the criterion is independent of specific design features to the extent practical, and (b) the fuel designer can retain flexibility to adjust all Level 1 parameters for an optimal balance among various design objectives.

For the above reason, to the extent practical, this Annex has reported acceptance criteria at Level 3 of Fig. I-1, i.e. in terms of key mechanistic parameters that measure the degree of damage in the ‘penultimate’ stage.

After the key mechanistic parameters are so identified, the acceptance criteria are formulated by first establishing a numeric value for the material damage limit associated with those parameters, and then subtracting a minimum acceptable margin from it. Thus, four steps are required in total, as explained in Sections I-5.2 and I-5.3.

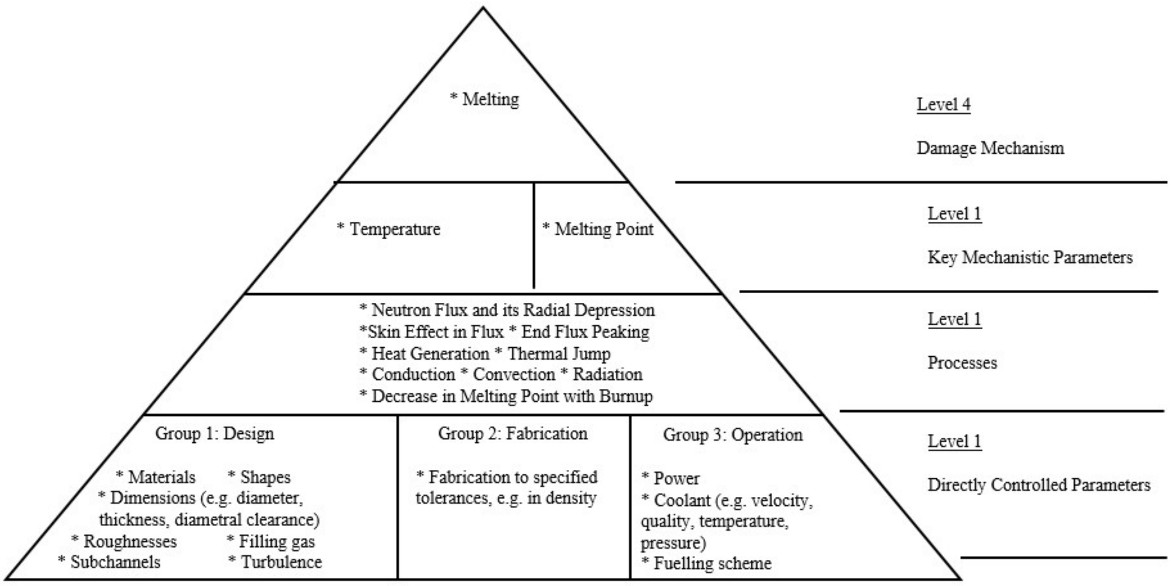


FIG. I-1. Simplified hierarchy of parameters in pellet melting (reproduced courtesy of SNC-LAVALIN, Candu Energy Inc.).

I-5.2. Material damage limits

In the first step, a hierarchical diagram is drawn for each damage mechanism similar to Fig. I-1, and based on that, the most appropriate damage parameter, ideally at ‘Level 3’ (to the extent practical), is identified.

In the second step, the numeric value of the ‘material damage limit’ for the identified damage parameter is established. At this numeric value, the material ‘fails’, for example by cracking or melting. Thus, material damage limits are the limits beyond which the fuel would be damaged to the extent that it would no longer meet some of its design requirements.

The above numeric values are established through experiments and related analyses of the condition of the material when damage occurs. Where these material damage limits are not known in sufficient detail, close precursors can be used as surrogates.

There is almost always scatter in the data for material damage. For establishing an acceptance criterion, an appropriately conservative end of the range would be used.

I-5.3. Prudent margins

In the third step, a 'minimum acceptable margin' is established for that damage mechanism. A minimum acceptable margin largely reflects the state of knowledge in that damage mechanism. Reference [I-1] provides some guidelines for establishing numeric values of minimum acceptable margin.

In the fourth and final step, the acceptance criterion is obtained by subtracting the minimum acceptable margin from the material damage limit.

This process establishes an acceptance criterion in the sense that, if it is met, no further justification is needed to assert that the associated damage mechanism is avoided. Even when the fuel is operating at the limit of the acceptance criterion, it is still operating with the minimum acceptable margin. Thus, the acceptance criterion represents a sufficient, but not absolutely necessary, condition to avoid damage.

Therefore, when the design verification process subsequently confirms that all acceptance criteria are met, this also means that damage from each mechanism is avoided by at least its minimum acceptable margin.

Figure I-2 illustrates the above concepts. It shows that the actual range of operation may be below the acceptance criterion. Therefore, the actual margin may be higher than the minimum acceptable margin.

An illustrative numeric example is presented in Section I-5.4.

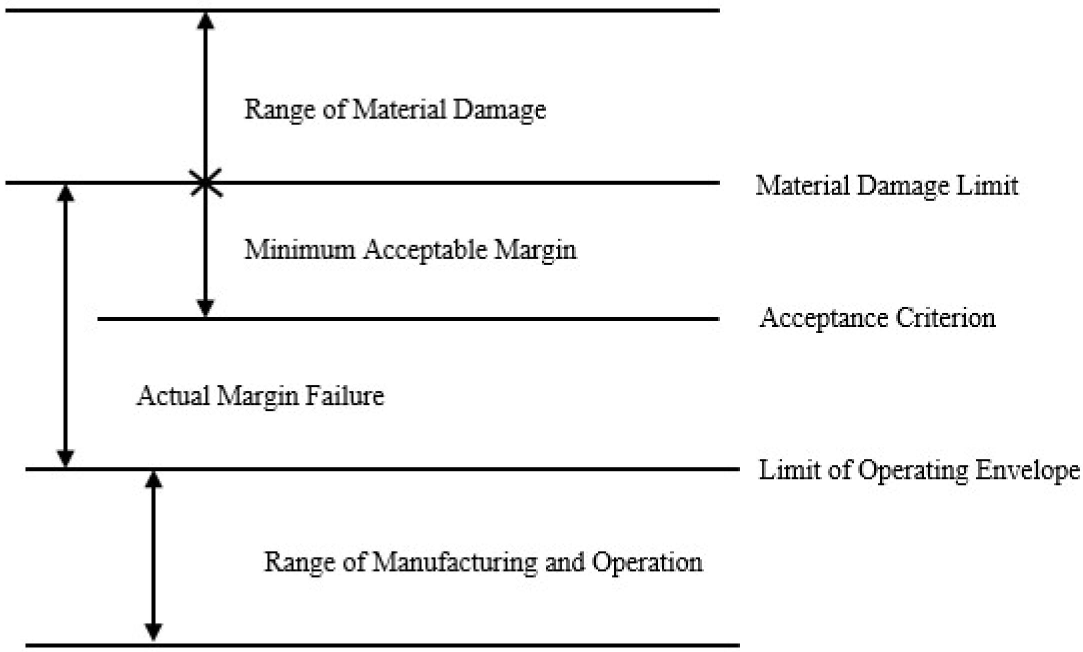
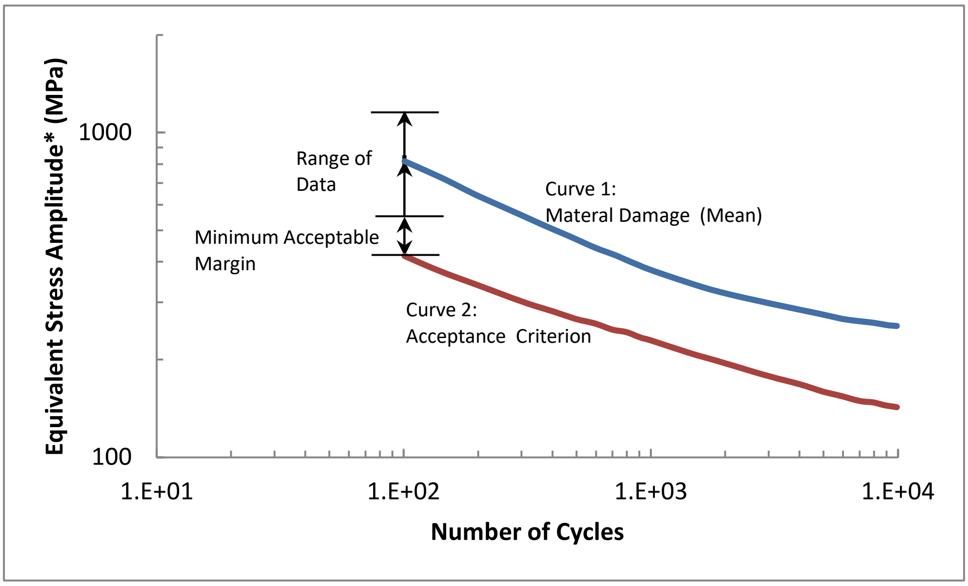


FIG. I-2. Damage limits and margins (reproduced courtesy of SNC-LAVALIN, Candu Energy Inc.).

I-5.4. Illustrative numeric example

Figure I-3 is a numeric example that illustrates the acceptance criterion proposed by O'Donnell and Langer for fatigue of Zircaloy [I-23]. Curve 1 was determined experimentally and represents the magnitude of alternating stress, peak-to-mean, that results in failure for a given number of cycles. Curve 1 was drawn through the middle of the scatter of data rather than through the lower end of the scatter.

To curve 1, either a factor of 2 to stress or a factor of 20 to the number of cycles was applied [I-23], whichever was more conservative at any point. These factors account for the scatter in data as well as an additional margin to cover (a) minor 'known unknowns' such as surface finish, and (b) 'unknown unknowns'. This results in curve 2 which is the acceptance criterion.



* Equivalent Stress Amplitude = Strain Amplitude (mean-to-peak) × Young's modulus

FIG. I-3. Fatigue design limit (reproduced courtesy of SNC-LAVALIN, Candu Energy Inc.).

Figure I-3 also illustrates the minimum acceptable margin at 100 cycles that is reflected in this acceptance criterion.

I-5.5. Key features

In a new fuel design, a fuel designer may sometimes consider using slightly different materials than current ones. As an illustrative example, a neutronic poison was added into the pellets of the central fuel element of the ACR fuel bundle. Features such as these can alter numeric values of some material properties. For this reason, a fuel designer usually prefers to use generic language (e.g. 'pellet material' rather than UO_2) to formulate acceptance criteria. Also, a fuel designer would normally not include the specific number for a material property in the formulation of the criterion and would, for example, refer to the 'pellet's melting point' rather than 2840°C.

Regardless of whether or not numeric values of material properties are embedded in a criterion, the analysis would need to establish and use numeric values pertinent to the specific material, for that particular fuel design and its operating conditions. Thus, the numeric values of material properties may differ from one design and operating conditions to another.

Sometimes a given damage mechanism may have several major aspects which are better controlled through more than one acceptance criterion. For example, in the traditional approach, excessive local stress/strain are avoided through separate criteria for, among others, longitudinal ridging of the sheath and structural integrity of end plates.

In using the criteria listed in the next section, the norm would be to deal with peak local values – unless a good reason is explicitly established to do otherwise. This means accounting for local influences such as the highest degree of local end flux peaking, local concentrations of stresses and strains, or migration and precipitation of deuterium and hydrogen.

One would consider the impacts of all pertinent conditions, processes and mechanisms that significantly affect any given material property or performance parameter. For example, when calculating critical heat flux, not only thermal and hydraulic conditions, but also the impacts of fuel deformations, if significant, would be considered. Similarly, when calculating ductility, the impact of fluence, temperature, deuterides, hydrides, oxides, crud and deposits would be accounted for. Many other illustrative examples can be constructed in a similar manner.

Interactions among the mechanisms would also be considered. As an illustrative example, impacts of oxides, crud and deposits would be considered in calculating temperatures.

Fuel acceptance criteria differ from operating values. As an illustrative example, consider the pellet centreline temperature. A fuel acceptance criterion might be formulated according to which the centreline temperature of the pellet needs to stay below the melting point of the pellet – the melting point being a material damage limit. However, fuel design and its operating conditions may be such that the actual operating temperature during normal operation is significantly below the melting point, i.e. the fuel may operate with considerable margin, and the margins may differ in different designs and reactors.

I-6. SPECIFIC ACCEPTANCE CRITERIA FOR CANDU FUEL

To respect commercial proprietary interests, numeric contents of specific criteria below are limited to open-literature information.

I-6.1. Thermal integrity

I-6.1.1. Overheating of the pellet

The local temperature in all parts of the pellet needs to stay below the minimum acceptable margin to the local melting point of the pellet material.

I-6.1.2. Overheating of the sheath and other structural materials

The local temperature in all parts of the sheath needs to stay below the minimum acceptable margin to the local melting point of the sheath material.

In practice, a precursor is used to implement this objective – e.g. by avoiding dryout during normal operations, and by limiting the severity and duration of dryout during AOOs.

Dryout is not in itself a damage mechanism, and it is a sufficient – though not necessary – condition to avoid overheating the sheath.

I-6.2. Structural integrity

I-6.2.1. Element internal gas pressure

Excess of internal pressure over coolant pressure needs to be less than the minimum acceptable margin to the differential pressure that causes cracking in the fuel sheath or end cap.

I-6.2.2. Stress corrosion cracking

Stresses/strains (or related powers and ramps) during power increases in fuel elements at circumferential ridges and at sheath/end cap junctions need to be below the minimum acceptable margin to the appropriate defect threshold. This includes the effects of pellet chips, if any, from refuelling impacts.

I-6.2.3. Static mechanical overstress and/or overstrain

Local principal strain/stress (elastic plus plastic) need to be less than the minimum acceptable margin to the available ductility/ultimate tensile strength; and local creep strain shall be less than the minimum acceptable margin to the creep rupture strain.

I-6.2.4. Mechanical rupture due to impact loads

Strain energy density during impact needs to be less than the minimum acceptable margin to that required to crack or break any metallic component of the fuel bundles.

I-6.2.5. Fatigue

Cumulative fatigue damage from repeated cycles of alternating stresses/strains needs to be below the allowable design fatigue life, with a minimum acceptable margin on magnitude of cyclic stress/strain or on number of cycles. When the fatigue cycles have variable amplitudes of strains/stresses, the sum of fatigue life fractions has to be less than 1.0.

An illustrative numeric example of this criterion has already been presented in Section I-5.4.

I-6.2.6. Loss of control of geometry

Axial and related loads on the fuel bundle are limited to less than the minimum acceptable margin to the bundle's buckling strength.

I-6.2.7. Primary deuteriding/hydriding

Equivalent concentration of internal hydrogen gas of an as-fabricated fuel element, excluding the sheath, is not allowed to exceed the minimum acceptable limit.

In addition, volume-average concentration of hydrogen/deuterium (in the form of soluble atomic hydrogen/deuterium and equivalent hydrides and deuterides including their orientations) over the cross-section of load bearing components needs to be below the minimum acceptable margin to the amount necessary to retain sufficient ductility.

I-6.2.8. Oxides, crud and deposits

The combined thickness of oxide, crud and deposits on the fuel sheath outer surface has to be below the minimum acceptable margins to the amount necessary for spalling from the surface and also for a through-wall hole.

I-6.3. Compatibility with surroundings

I-6.3.1. Excessive interaction loads along the fuel string

After considering all pertinent in-reactor deformations, the maximum length of the fuel string (e.g. in the fuel channel) is limited to less than the minimum available space (e.g. between shield plugs or latches), with an acceptable minimum margin.

I-6.3.2. Excessive interference with interfacing equipment

Fuel bundle dimensional changes (e.g. due to irradiation, loads, creep, bowing, droop) shall not result in clearances that are less than the minimum acceptable margin to contact between neighbouring sheaths or end caps, nor between pressure tube and sheath/end cap. Nor shall the on-power clearances – with or without fuel deformations – be below those assumed in verification and safety assessments, for example in calculations of heat transfer coefficients.

Net dimensions, including dimensional changes throughout fuel bundle's residence in the reactor, need to be within specified limits for interfacing equipment.

To allow passage of fuel through the reactor in all fuel handling operations, the axial force necessary to move the bundle has to be within design allowance, including all pertinent considerations such as on-power deformations, in-service contacts with neighbouring components if any are permitted, and changes in material properties.

I-6.3.3. Wear

At spacer pads, total wear from all sources such as lateral vibrations, axial vibrations, fretting, sliding and erosion need to be less than that which brings any part of a spacer in contact with a neighbouring sheath, with a minimum acceptable margin.

At bearing pads, total wear and corrosion from all sources need to be less than the thickness of the pad, with a minimum acceptable margin.

Depth of all wear and corrosion in the pressure tube from fuel bearing pads need to be within specified allowances.

I-6.3.4. Crevice corrosion

Below bearing pad or spacer pad, temperature at the fuel sheath outer surface needs to be below the minimum acceptable margin to that required to cause crevice corrosion of the sheath.

Depth of crevice corrosion in the pressure tube from fuel bearing pads needs to be within specified allowances.

I-7. SUMMARY

In a recently developed process – called 'criteria-based verification' – fourteen credible damage mechanisms have been identified for CANDU fuel during operational states during which fuel experiences temperatures near those experienced during normal operations. Their associated "acceptance criteria" help verify that the fuel is not damaged while delivering its functional requirements during normal operations, nor does it damage its surroundings. The criteria reflect basic limits of materials (e.g. melting point, ductility) minus prudent margins.

The damage mechanisms are organized into three groups:

- Mechanisms that can potentially threaten the thermal integrity of fuel through potential overheating. This comprises two damage mechanisms: pellet melting and sheath overheating;
- Mechanisms that can potentially threaten the structural integrity of fuel through potential cracks, breaks, or loss of structural stability in appropriate parts. This

comprises eight damage mechanisms – gas pressure, power ramps, mechanical overstress/overstrain, impact loads, fatigue, unstable geometry, primary hydrides/deuterides, and oxide, crud, deposits;

- Mechanisms that can adversely impact the geometric or other compatibility of fuel with interfacing systems. Geometric compatibility would ensure that critical parts mate/fit with their interfaces. Other compatibility would include chemical compatibility, such as limiting crevice corrosion to within design allowances. This comprises four damage mechanisms – string length, interference with interfacing equipment, wear, and crevice corrosion.

The fuel acceptance criteria, introduced relatively recently in the context of design and qualification of the ACR fuel, are design basis inputs which have the following characteristics:

- They are required inputs to the fuel design and fuel design qualification process. Defining fuel acceptance criteria is now one element of the process by which CSA N286:2012 requirements [I-24] are met;
- They are established for each known fuel damage mechanism in operational states;
- To the extent practical, they are independent of specific design features. Therefore, numeric values of material properties would usually not be built into the criterion. Nevertheless, when the criteria are applied, different numeric values may be used in different designs if they use different materials or operate at different conditions. Likewise, if a new design feature is added, additional criteria may be also required;
- The resulting limits are sufficient, but not necessarily necessary. Even at the acceptance criterion, the design would still contain residual margin to failure;
- If these criteria are met, no further justification is required to assert that the fuel is not damaged while delivering its functional requirements, nor does it damage its surroundings.

Compared to the traditional approach, fuel verification using mature CBV would (a) be quicker; (b) facilitate enhanced comprehensiveness; and (c) enable easier licensing in some international jurisdictions.

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ANNEX II. CATEGORIZATION OF PLANT STATES FOR CANDU REACTORS (CANADIAN PRACTICE)

II-1. INTRODUCTION

Requirement 20 of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1) [II-1] states:

> "A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures."

In Canada, significant efforts were made to account for the lessons learned from the Fukushima Daiichi accident. The Canadian nuclear industry updated its CANDU severe accident management guidelines [II-2] to reflect operating experience and lessons learned from the Fukushima Daiichi accident, and the Canadian Standards Association (CSA) issued its Standard N290.16-16 on Requirements for BDBAs [II-3]. The CNSC issued a number of regulatory documents [II-4 – II-6] providing requirements and guidance regarding BDBAs. For instance, the regulatory document REGDOC-2.5.2 on Design of Reactor Facilities: Nuclear Power Plants, published in 2014 [II-5], represents to a large degree the CNSC's adoption of the principles set forth in SSR-2/1 (Rev. 1) [II-1].

Although REGDOC-2.5.2 sets out requirements and guidance for the application of DECs to new licence applications for water-cooled nuclear power plants in Canada, the concept of DECs, as described therein, and guidance provided regarding its application have been used [II-7 – II-9] to explore possible applications of DECs to existing facilities in Canada. In fact, following the reviews for refurbishment or extended operation of existing Canadian nuclear power plants, many upgrades to those facilities have already been made to address DECs; these upgrades are briefly outlined in [II-8].

II-2. PLANT STATES, PLANT DESIGN ENVELOPE AND DESIGN EXTENSION CONDITIONS

For the purposes of the Canadian regulatory framework, a plant state is defined as "[a] configuration of nuclear power plant components, including the physical and thermodynamic states of the materials and the process fluids in them" [II-5]. As illustrated in Fig. II-1, in Canada[24], plant states are grouped into a number of categories primarily on the basis of their frequency of occurrence at the plant and include operational states (normal operation and AOOs), DBAs and BDBAs. "AOOs include all events with frequencies of occurrence equal to or greater than 10^{-2} per reactor year. DBAs include events with frequencies of occurrence equal to or greater than 10^{-5} per reactor year, but less than 10^{-2} per reactor year. BDBAs include events with frequencies of occurrence less than 10^{-5} per reactor year." [II-3 – II-6] Not included in Fig.

[24] Historically in Canada, postulated initiating events have been grouped as a function of the probability of occurrence in either five event classes (Darlington) [II-11], or four event classes (Pickering, Bruce and Point Lepreau sites) [II-12]. Those classical event classes provided the basis for safety analysis until the CNSC introduced, in 2008, new event classes that aligned with international practices including AOOs, DBAs and BDBAs. Action plans, to transition to this new analysis framework, have been proposed by licensees and accepted by the CNSC.

II-1, but also considered part of the plant states, is the 'post-accident' state defined in [II-6] as a long term safe stable state that is achieved in the reactor facilities after an accident.

<table>
<tr><td colspan="2">Operational states</td><td colspan="4">Accident conditions</td></tr>
<tr><td rowspan="3">Normal operation</td><td rowspan="3">Anticipated operational occurrences</td><td rowspan="3">Design-basis accidents</td><td colspan="3">Beyond-design-basis accidents</td></tr>
<tr><td colspan="2">Design-extension conditions</td><td>Practically eliminated conditions</td></tr>
<tr><td>No severe fuel degradation</td><td colspan="2">Severe accidents</td></tr>
<tr><td colspan="3">Design basis</td><td colspan="2">Design extension</td><td>Not considered as design extension</td></tr>
<tr><td colspan="6">Reducing frequency of occurrence ➔</td></tr>
</table>

FIG. II-1. Plant states – Canadian practice (reproduced courtesy of CNSC [II-4, II-5]).

The plant design envelope concept is introduced in REGDOC-2.5.2 [II-5] as "The range of conditions and events (including DECs) that are explicitly taken into account in the design of the nuclear power plant such that significant radioactive releases would be practically eliminated by the planned operation of process and control systems, safety systems, safety support systems and complementary design features". The plant design envelope comprises all plant states considered in the design: normal operation, AOOs, DBAs and DECs.

The concept of DECs has been introduced with the objective of defining those conditions which need to be considered in plant design, in addition to the design basis conditions, with the purpose of further strengthening the plant safety. DECs are defined as "a subset of beyond-design-basis accidents that are considered in the design process of the facility in accordance with best-estimate methodology to keep releases of radioactive material within acceptable limits" [II-3 – II-10]. As indicated in Fig. II-1, DECs comprises two categories of plant states: one without significant fuel degradation which are states with no core melting and no more than one channel failure (i.e. limited core damage events), and another which involves severe accident conditions that could include core (fuel) melt and/or two or more fuel channel failures (i.e. progressing into severe accidents where the safety goal is to maintain core geometry, not protect the fuel). According to the definition in [II-13], a severe accident in a CANDU reactor is an accident in which the fuel heat is not removed by the coolant in the PHTS.

The CNSC's definition of DECs is based on the one formulated by the IAEA [II-1], but has been slightly modified to clarify that DEC is a subset of BDBA; it does not include BDBAs that can be considered to be 'practically eliminated' where the latter term is defined [II-5, II-10] as the possibility of certain conditions occurring being physically impossible or with a high level of confidence to be extremely unlikely to arise. CANDU specific examples of BDBAs that are considered 'practically eliminated' are given in [II-8].

II-3. ROLES OF DESIGN EXTENSION CONDITIONS IN DEFENCE IN DEPTH AND PHYSICAL BARRIER INTEGRITY

The primary means of preventing accidents and mitigating the radiological consequences of accidents if they do occur is the application of the concept of defence in depth [II-1, II-14]. The essential means of each level of defence in depth prevent the need for activation of the essential means of the following level and, at the same time, they mitigate the consequences of the failure of the previous ones. The first level of defence has a predominant preventive function. Its aim is "to prevent deviations from normal operation and the failure of items important to safety" [II-1]. The fifth and final level of defence has only a mitigatory function; it addresses emergency response provisions, which envelopes inability of all previous defence in depth levels to successfully prevent or mitigate the event [II-14].

Canada [II-3 – II-5, II-7, II-8] has adopted the approach which consists in having both categories of DECs assigned to level 4. The grouping of DECs without core melt, and with core melt, in level 4 facilitates the differentiation between the set of rules for design and safety assessment to be applied for DECs, which are beyond DBAs, from those for DBA.

The CSA standard N290.16 [II-3] requires that the containment integrity be maintained for both defence in depth levels 4a and 4b to minimize radioactive releases. The provisions introduced at level 4a are aimed at not only ensuring that the plant's 'contain' fundamental safety function will be performed to the degree of effectiveness required to ensure containment integrity, but also that the plant's 'cool' and 'control' fundamental safety functions will be performed to the degree of effectiveness required to ensure (1) coolable core geometry and coolable bundle geometry, and (2) no contribution to core damage frequency and large release frequency. Maintenance of coolable core geometry implies no more than one channel failure, and coolable bundle geometry means that the channel decay heat can be removed from the bundles in a channel, in the long term, by the plant's cool safety function. The provisions at level 4b are to terminate accident progression at the earliest available opportunity, return the plant to a stable state and protect containment integrity.

II-4. FUEL ACCEPTANCE CRITERIA AS DEFENCE IN DEPTH PROVISIONS AND MEASURES

For normal operating conditions, fuel acceptance criteria are part of the provisions at defence in depth level 1, ensuring that there are adequate design margins for stable operation. Fuel acceptance criteria (that need to be met in order to avoid damaging the fuel and its interfaces) for AOOs are part of the provisions at defence in depth level 2, ensuring the effectiveness of the reactor regulating system. Fuel acceptance criteria for accident conditions are part of the provisions at defence in depth levels 3 and 4. The potential usefulness of fuel acceptance criteria for design extension conditions (defence in depth level 4) remains an area to be investigated.

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ANNEX III. CATEGORIZATION OF PLANT STATES (INDIAN PRACTICE)

Indian practice to categorize plant states is in accordance with the AERB Safety Codes on Design of Light Water Reactors (AERB/NPP-LWR/SC/D) and Design of Pressurized Heavy Water Reactors (AERB/NPP-PHWR/SC/D, Rev. 1, under revision) and the AERB Safety Guide on Design Basis Events for Pressurized Heavy Water Reactor, AERB/SG/D-5 (under revision).

The plant states are identified and grouped into a limited number of categories according to their likelihood of occurrence and defence in depth level which is required to mitigate such an event. Further, it needs to be ensured that there are no, or only minor, potential radiological consequences for all the plant states with a significant likelihood of occurrence.

The plant states considered in design for the deterministic safety analysis cover:

- Normal Operation (NO);
- Anticipated Operational Occurrences (AOO);
- Design Basis Accidents (DBA);
- Design Extension Conditions (DEC).

The plant states are illustrated in Fig. III-1, which is taken from the AERB Safety Guide on Accident Management Programme for Water Cooled Nuclear Power Plants, AERB/SG/D-26 (currently in preparation).

<table>
<tr><th colspan="2">Operational States</th><th colspan="4">Accident Conditions</th></tr>
<tr><td rowspan="2">Normal operations</td><td rowspan="2">Anticipated operational occurrences</td><td rowspan="2">Design basis accidents</td><td colspan="2">Design extension conditions</td><td>Practically eliminated conditions</td></tr>
<tr><td>Accidents without significant core/fuel*degradation</td><td>Accidents with core melt/significant core degradation[@]</td><td>Early or large release of radioactivity from containment</td></tr>
<tr><td colspan="5">Considered in design</td><td></td></tr>
<tr><td colspan="4"></td><td colspan="2">Severe accidents</td></tr>
</table>

*'Fuel' is used here to address the spent fuel pool events.
[@] 'Fuel' is not used here as accidents with fuel melt in the spent fuel pool are practically eliminated. 'Core melt' terminology is applicable to LWRs whereas 'significant core degradation' is applicable to PHWRs.

FIG. III-1. Plant states defined in an Indian Safety Guide.

DEFINITION OF TERMS

The following list provides the definitions of relevant terms that are either not included in the IAEA Safety Glossary (2018 Edition) or used with a specific or different meaning in this publication, taking into account uses in Member States contributing to it. The definitions appearing in this list apply for the purposes of this publication only.

design extension condition. In Canada [14], design extension conditions are a subset of beyond-design-basis accidents, deviating from the definition provided in [2]. See [157] for supporting reference.

failure mechanism. A means by which a barrier may fail due to the operation of some process or transition. See [158] for supporting reference.

fitness for service. The ability of equipment to continue operation for the desired period of time while maintaining conformance to applicable requirements. See [22] for supporting reference.

fuel degradation mechanisms. Mechanisms that could be active during normal operations, AOOs and accident conditions, and could impair the ability of the fuel (element/bundle) to function within its fuel acceptance criteria. For operational states they are referred to as 'damage mechanisms', and for accident conditions they are referred to as 'failure mechanisms'.

fuel design qualification. The set of tests and analysis that is performed to demonstrate that a fuel bundle design meets all its design and safety requirements.

margin. The conservatism (safety factor) included in operational limits and the design of every system, structure, and component in a nuclear plant. See [159] for supporting reference.

material damage limit. Indicates the threshold where damage to the fuel bundle components materials (including fuel cladding material) is expected to start.

plant design envelope. The range of conditions and events (including design extension conditions) that are explicitly taken into account in the design of the nuclear power plant such that significant radioactive releases would be practically eliminated by the planned operation of process and control systems, safety systems, safety support systems and complementary design features. See [9] for supporting reference.

safety margin. The difference or ratio in physical units between the limiting value of an assigned parameter the surpassing of which leads to the failure/damage of a system or component, and the actual value of that parameter in the plant. See [160] for supporting reference.

safe operating envelope. The set of limits and conditions within which the nuclear generating station needs to be operated to ensure compliance with the safety analysis upon which reactor operation is licensed and which can be monitored by or on behalf of the operator and can be controlled by the operator. See [15] for supporting reference.

LIST OF ABBREVIATIONS

ACR	Advanced CANDU Reactor
AECL	Atomic Energy of Canada Limited (former name of Canadian Nuclear Laboratories)
AOO	Anticipated operational occurrence
BDBA	Beyond design basis accident
CANDU	Canadian Deuterium Uranium
CANLUB	CANDU Lubricant
CBV	Criteria-based verification
CHF	Critical heat flux
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
DBA	Design basis accident
DEC	Design extension condition
GSS	Guaranteed shutdown state
IBIF	Intermittent buoyancy induced flow
LOCA	Loss of coolant accident
LWR	Light water reactor
NEA	Nuclear Energy Agency
OECD	Organization for Economic Cooperation and Development
PCMI	Pellet-cladding mechanical interaction
PHTS	Primary heat transport system
PHWR	Pressurized heavy water reactor
PWR	Pressurized water reactor
RIA	Reactivity initiated accident
SCC	Stress corrosion cracking

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