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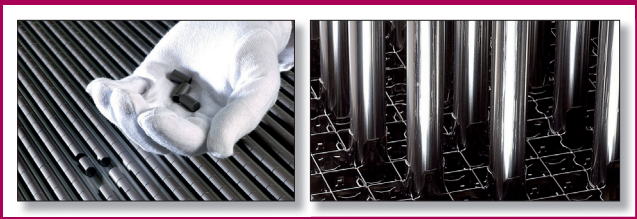
Nuclear Fuel Safety Criteria Technical Review

Second Edition



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NUCLEAR ENERGY AGENCY

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The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

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The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Cover photos: Nuclear fuel pellets (AREVA); nuclear fuel assembly spacer grid with guide tubes (AREVA).

Foreword

Most of the current nuclear fuel safety criteria were established during the 1960s and early 1970s. Although these criteria were validated against experiments with fuel designs available at that time, a number of tests were based on unirradiated fuels. Additional verification was performed as these designs evolved, but mostly with the aim of showing that the new designs adequately complied with existing criteria, and not to establish new limits.

In 1996, the OECD Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Task Force on Fuel Safety Criteria was given the mandate to review existing fuel safety criteria and to focus on new fuel and core designs, new cladding materials and industry manufacturing processes. The task force was also asked to identify those areas in which additional efforts might be necessary to ensure that the technical bases for fuel safety criteria remain adequate.

As a result of this work, a set of fuel-related safety criteria was presented – along with both the rationale for having such criteria and possible new design and operational issues which could have an effect on them – in the 2001 NEA report entitled *Nuclear Fuel Safety Criteria Technical Review*.

The NEA Working Group on Fuel Safety (WGFS), a successor to the task force, is tasked with advancing the understanding of fuel safety issues by assessing the technical basis for current safety criteria and their applicability to high burn-up and to new fuel designs and materials. The group aims to facilitate international convergence in this area, including the review of experimental approaches as well as the interpretation and use of experimental data relevant for safety.

Like its predecessors, the WGFS has re-examined fuel-related criteria and for each of these criteria, it has presented a brief description of each criterion along with its rationale. Design changes, such as different cladding materials, higher burn-up and the use of mixed-oxide (MOX) fuels, can affect fuel-related margins and, in some cases, the criteria themselves. Some of the more important effects are cited in an attempt to identify criteria that need re-evaluation. The discussion does not cover all possible effects, but should be sufficient to identify those criteria that should continue to be examined.

This second edition of *Nuclear Fuel Safety Criteria Technical Review* provides the results of the WGFS re-examination.

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1. Introduction

With the advent of new fuel and core designs, the adoption of higher performance reactor operation, and the implementation of advanced design and analysis methods, there is a concern that current fuel safety criteria may not remain adequate under these new conditions (e.g. higher burn-up).

Historically, fuel safety margins were defined as the conservatisms in the safety criteria, which in turn were also fixed in a conservative manner; here, the expression "conservatism" expresses the fact that bounding or limiting values were chosen for model parameters, plant and fuel design data, and fuel operating history values. Unfortunately, some of these conservatisms are not quantified (or quantifiable), and the amount of safety margins available or the reduction thereof is difficult to substantiate.

For the regulator it is important to know the margins and their verification basis, as the industry requests approval of new fuels or methods; likewise, for the utilities and vendors it is important to know what margins are available, to identify in which direction further progress may be made to optimise fuel design and fuel cycle cost. Naturally, each party involved will have to decide on how much margin should be in place, when criteria have been established.

Most of the current fuel safety criteria were established during the 1960s and early 1970s. Although these criteria were verified against experiments with fuel designs available at that time, a number of tests were based on unirradiated fuels. Additional verification was performed as these designs evolved, but mostly with the aim of showing the new designs adequately complied with existing criteria, and not to establish new limits.

Current criteria have so far fulfilled their function, in that during decades of operational experience no incidents have been reported caused by inadequacy of fuel safety criteria. New demands on fuel and plant performance, however, have reduced the available margins; also, optimising fuel utilisation and core performance show a trend towards conditions in which less operational and experimental experience exists.

In 1996, the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Task Force on Fuel Safety Criteria was given the mandate to review the existing fuel safety criteria, and to focus on new fuel and core designs, new cladding materials and industry manufacturing processes. The task force was also charged with identifying those areas where additional efforts might be necessary to ensure that the technical bases for fuel safety criteria remained adequate.

In 2001, the considerations of the task force were published in the NEA/CSNI report entitled *Nuclear Fuel Safety Criteria Technical Review* [1].

In 2003, a successor to the task force, the CSNI Expert Group on Fuel Safety Margins, compiled information on the present fuel safety criteria used in NEA member countries with the objective of determining national practices in the use of fuel safety criteria, and to identify the differences and commonalities between the different countries. The group issued its findings as "Fuel Safety Criteria in NEA Member Countries" [2].

These previous NEA reports and recent CSNI-sponsored seminars have contributed to an improved understanding of fuel safety criteria. While the criteria may have evolved, the construction of new reactors has placed similar plant designs in multiple NEA member countries and has placed greater emphasis on common, well-understood fuel safety criteria within those plant designs. However, no comprehensive update has been produced since the *Nuclear Fuel Safety Criteria Technical Review* and the "Fuel Safety Criteria in NEA Member Countries" survey.

The differences in earlier reported fuel safety criteria (and possibly fuel design criteria) indicate areas where further international co-operation would be worthwhile. It was presumed that further investigation will result in a better understanding of the reasons for national differences and may also contribute to a convergence among NEA member countries. It is with this goal that the current CSNI group, the Working Group on Fuel Safety (WGFS), has embarked on a re-examination of the fuel safety criteria.

Like its predecessors, the WGFS has re-examined fuel-related criteria[1] with only a modest attempt to categorise them according to event type or risk significance. For each of these criteria, the working group presents a brief description of the criterion as it is used in several applications along with the rationale for having such a criterion. Design changes, such as different cladding materials, higher burn-up, and the use of mixed-oxide (MOX) fuels, can affect fuel-related margins and, in some cases, the criteria themselves. Some of the more important effects are mentioned in an attempt to identify criteria that need re-evaluation. The discussion does not cover all possible effects, but should be sufficient to identify those criteria that should continue to be addressed.

As a result of this re-assessment, the working group continues to regard the current framework of fuel safety criteria as generally applicable. However, the numeric values in the individual safety criteria may change in accordance with the particular fuel and core design features. Some specific criteria and associated values continue to be modified and adjusted on the basis of new experimental data and analyses.

Complete or sufficient information is not available for a number of issues discussed in this report. These include CRUD deposition, cladding oxidation and

1. The working group has avoided the distinction between safety and design criteria, as the differences are not uniquely defined in each country. Sometimes the difference is related to vendor methodologies.

hydriding, rod internal gas pressure, pellet-cladding and thermal-mechanical loads, fuel melting, fuel fragmentation, cladding embrittlement, gap activity, radioactive source term, high burn-up, mixed-oxide fuel, slow or incomplete control rod insertion, axial offset anomaly, cladding elongation, and cladding stability. Under the auspices of the OECD Nuclear Energy Agency, active research is being conducted in many of these areas through programmes including the Halden Reactor Project in Norway, the Studsvik Cladding Integrity Project in Sweden, and the CABRI International Project in France. These issues have been, and will continue to be addressed by the Working Group on Fuel Safety, as directed by the NEA Committee on the Safety of Nuclear Installations.

For this re-assessment of fuel criteria, the following process is recommended:

- Continue to develop best-estimate analysis methods, together with a suitable uncertainty analysis, in all areas of safety analysis.
- Continue to perform experimental studies for benchmarking of best-estimate codes and extending the verification and validation basis for safety criteria and codes (the amount of testing may be reduced as code quality advances).
- Review, and adjust where necessary, safety criteria values based on the above codes, methods and test data. Define or quantify necessary margin to safety limits.

2. Re-examination by the working group

2.1. Fuel safety, operational and design criteria: historical perspective

The goal of reactor safety is to ensure that the operation of commercial nuclear power plants does not contribute significantly to individual or societal health risks. Reactor safety is concerned with the prevention of radiation-related damage to the public from the operation of commercial nuclear reactors; fuel operational or design limits are introduced to avoid fuel failures during normal operation, and to mitigate the consequences of accidents in which substantial damage is done to the reactor core.

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

Fuel safety criteria, with derivative fuel design and operational limits, are the focus of this report. The current safety criteria for light water reactors, which form the large majority of the existing commercial nuclear power plants in the world, were developed during the late 1960s and early 1970s. An underlying idea in this development process is that the consequences of these postulated events are inversely related to their probability. For the sake of simplicity the postulated events are divided into two categories: anticipated transients (or anticipated operational occurrences, AOOs) and postulated accidents. In general, those events whose probability of occurrence varied from ~1 to 10^{-2}/yr were characterised as anticipated transients, or simply transients, while all other events whose probability was less than 10^{-2}/yr were characterised as (postulated) accidents.

The frequency spectrum within both of these categories varies. Within the transient spectrum are the more frequent events (classified in most countries as inherent to normal operation, or Condition I events), and the less frequent ones (classified in most countries as faults of moderate frequency, or Condition II events). Within the accident spectrum are events that may lead to failure of some fuel rods (e.g. reactor coolant pump seizure, in most countries classified as Condition III events) as well as postulated accidents of low probability (referred to as a design basis accident or DBA, in most countries classified as Condition IV events). Condition IV events include the loss-of-coolant accident (LOCA), and the reactivity initiated accident (RIA), both of which can lead to more substantial fuel failures. The last two DBAs are assumed to have a likelihood or probability of occurrence in the range of 10^{-4} to 10^{-6}/yr.

These probabilities were taken into account in the development of fuel safety criteria. For the more probable transients, as an example, safety criteria allow for no fuel failures. This is usually ensured by no (or only a very small number of) fuel

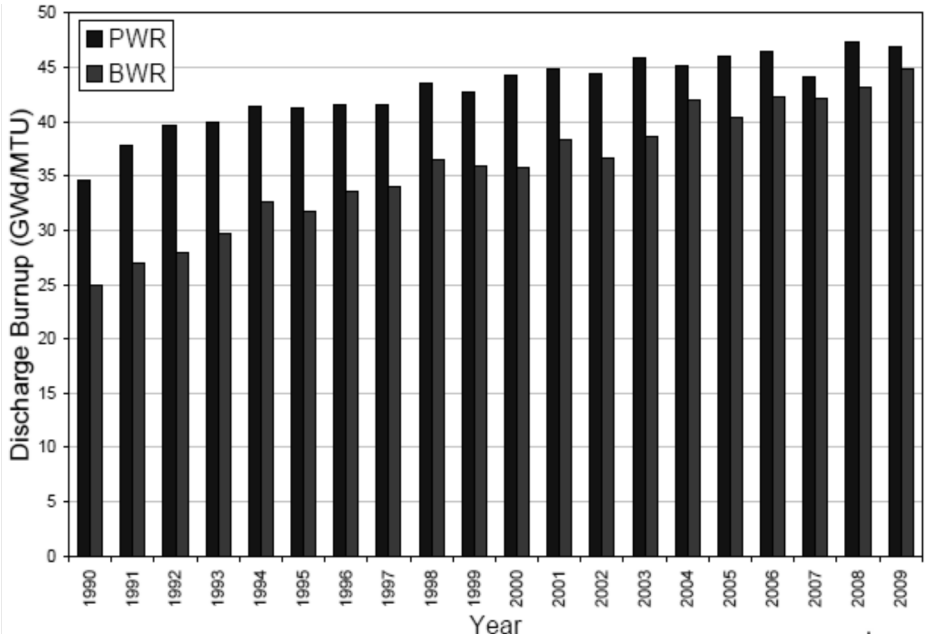
rods in the core to experience the boiling crisis, no fuel melting and no pellet-cladding interaction (PCI). With regard to boiling crisis, departure from nucleate boiling (DNB) for the pressurised water reactor (PWR) or critical heat flux (CHF) for the boiling water reactor (BWR) must be avoided during normal operation and anticipated operational occurrences. If fuel failure occurs during normal operation, the number of failed rods is *de facto* limited by the operational limits on coolant activity. If these limits are exceeded, the operator has to reduce power or shut down the plant. For less probable accidents, the criteria are usually established to ensure core coolability (e.g. limits to the energy deposition in the fuel during a RIA or limits on the temperature and total oxidation of the cladding following a LOCA). Criteria for normal operating conditions were also developed to ensure that the initial fuel conditions prior to a transient or accident do not compromise or lead to exceeding the fuel safety criteria themselves.

During the late 1960s and early 1970s a number of experiments were carried out, which provided information about fuel and reactor core behaviour for the more serious design basis accident and beyond design basis accident conditions. This information was used to develop the fuel safety criteria for these accidents as well as the related analytical methods. During the development of these criteria and methods, high burn-up was thought to be around 40 gigawatt-days per metric ton of uranium (GWd/t);[1] data up to this burn-up had been included in databases for criteria establishment, and regulatory decisions, and it was believed that some extrapolation to higher burn-up could be made. By the mid-1980s, however, changes in pellet microstructure had been observed from a variety of data at higher burn-up along with increases in the rate of cladding corrosion and hydriding (leading to mechanical properties degradation). It thus became clear that something different was occurring at high burn-up and/or new operating environments, and that continued extrapolation of data from the existing low burn-up and traditional operating environment was not appropriate.

Meanwhile regulatory authorities in a number of countries had allowed reactors to operate at exposures higher than those used in the development of the fuel safety criteria discussed earlier; in the United States, for example, the Nuclear Regulatory Commission (NRC) has licensed fuel burn-up in commercial pressurised water reactors up to 62 GWd/t (average exposure of the peak rod). It is interesting to note that the burn-up extension trend is stabilising, as shown in the following figure.

1. For this report, fuel burn-up is expressed in terms of gigawatt-days per metric ton of uranium (GWd/tU) or gigawatt-days per metric ton of heavy metal (GWd/tHM). Both are abbreviated GWd/t and both are equivalent to one megawatt-day per kilogram (MWd/kgU), but not equivalent to MWd/kgUO_2. Another aspect that is not always stated clearly is whether the burn-up is rod average or a more locally expressed (and higher) value.

Figure 1: Average US discharged assembly burn-up trends



Source: EPRI.

In Europe, high burn-up test programmes continue to be employed, with fuel rods in lead test assemblies attaining exposures of up to 100 GWd/t mainly to gather data for modelling development and validation. In countries where the fuel cycle is closed (e.g. France), there is no incentive to extend the burn-up beyond the current limits; higher burn-up would degrade the isotopic composition of the reprocessed fuels (MOX and reprocessed uranium) and the fuel cycle economics.

As a result of the worldwide trend to increase fuel burn-up well beyond the level of 40 GWd/t during the last 30 years, and the observations regarding pellet microstructure changes and increased rates of cladding corrosion/hydriding at higher burn-up (note: mechanically, corrosion is not harmful to the cladding, only hydrides in the bulk layer are detrimental), a number of test programmes were initiated, both of an experimental and analytical nature, to evaluate the effects of the higher burn-up on fuel behaviour, especially under RIA and LOCA conditions. The need and rationale for the additional work are described in the 1996 CSNI report on “Transient Behaviour of High Burn-up Fuel” [3].

Interest peaked after tests – related to the high burn-up fuel behaviour during the reactivity initiated accident – were performed by the French in the CABRI facility and by the Japanese in the NSRR facility. During two specific tests [REP Na-1 [4] and HBO-1 [5], respectively], performed with highly irradiated fuel, rods failed and some amount of fuel dispersal was observed at significantly lower enthalpy values than the peak fuel enthalpy limits that had been established earlier or previously

approved by the various regulatory authorities. This led to expanded efforts in a number of countries to gain a more complete understanding of highly irradiated fuel behaviour under postulated accident conditions.

The Halden LOCA tests with high burn-up fuel indicated that under accident conditions severe fuel fragmentation can take place and the fine grains of fragmented pellets can be released from the damaged fuel rod into the coolant. This phenomenon needs further investigations, but it calls attention to the fact that further increase of fuel burn-up can be limited by LOCA considerations.

In a 1996 report on *Nuclear Safety Research in OECD Countries*, the Committee on the Safety of Nuclear Installations (CSNI) recommended that "fuel damage limits at high burn-up" be recognised as a safety research area to which priority should be assigned. Specifically, that report indicated "Fuel damage limits should be established for the entire range up to high burn-up. Limits should be based upon appropriate parameters to ensure fuel integrity (i.e. enthalpy or enthalpy rise, DNB, cladding oxidation), and should consider the full range of possible transients, including reactivity insertion and LOCAs". As a consequence, the CSNI and its NEA counterpart, the Committee on Nuclear Regulatory Activities (CNRA), decided to undertake an effort involving a much broader (than only high burn-up related issues) look at fuel behaviour and requirements needed to assure appropriate safety margins of modern fuels and core designs.

2.2. Types of criteria

Different types of fuel criteria need to be recognised. In its 2003 report on "Fuel Safety Criteria in NEA Member Countries" [2], the CSNI working group previously identified several categories of criteria, based on the sources of the criteria. With some more recent explanatory text, these categories are:

- *Safety criteria* – Criteria imposed by the regulator. If preserved, safety criteria ensure that the impact of a DBA on the environment is acceptable.
- *Operational criteria* – Criteria specific to the fuel design and provided by the fuel vendor as part of the licensing basis. Operational criteria ensure that safety criteria are not violated.
- *Design criteria* – Limits employed by vendors and/or utilities for fuel and core design. Design criteria are preserved during the normal operation and anticipated transients.

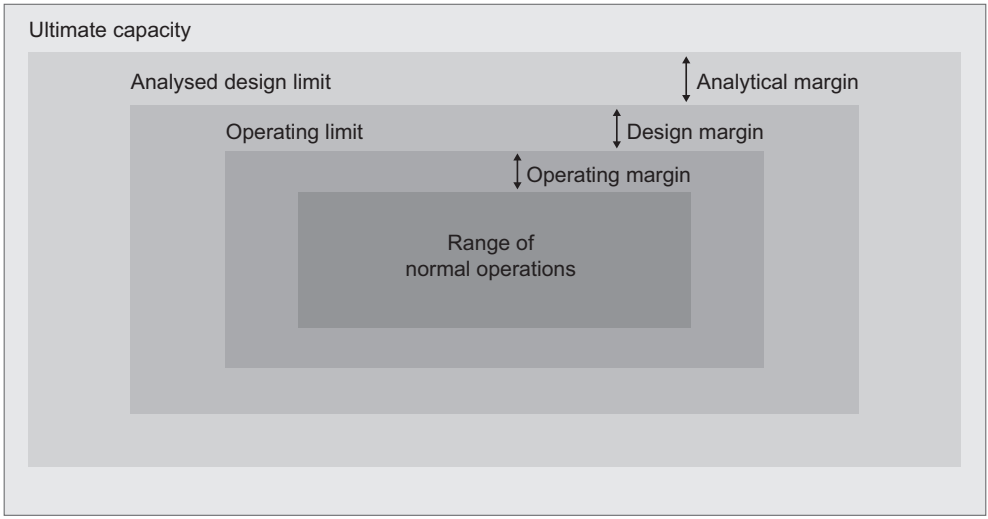
Unfortunately, not all criteria can be precisely identified by originating authorities, and the criteria vary between countries. Some criteria, such as fuel rod internal gas pressure limits, have originated with a fuel vendor, and have been subsequently adopted and uniformly applied by the regulatory authorities.

Other "safety" criteria, including the previously mentioned rod internal pressure and DNB limits, apply only during normal operation and anticipated transients, and not during accident conditions. During a number of design basis accidents, these criteria are assumed to be violated.

The relative conservatism (or "margin") between these categories is not always clear. Further, the relative order of the categories is subject to debate. For example, based on the definitions above, the design criteria are most restrictive and the safety criteria the least restrictive.

The Institute for Nuclear Power Operations (INPO) has adopted somewhat different criteria [6], which are identified as *operating* limit, *design* limit and *analytical* limit (or ultimate capacity). These criteria, which are identified in a different manner than the NEA 2003 report, are schematically shown in the following figure.

Figure 2: Concept of limits and margins



Source: INPO.

Presumably, the "analytical" margin in this figure is analogous to the "safety" margin used in the 2003 NEA report. Unlike the NEA example, the INPO "operating" limit is the most restrictive and the "analytical" limit is the least restrictive. The INPO "design" limit is intermediate to the other two limits.

In a third example, a pressure vessel might employ the following limits:

Design limit 12 MPa.

Safety limit 10 MPa.

Operational limit 8 MPa.

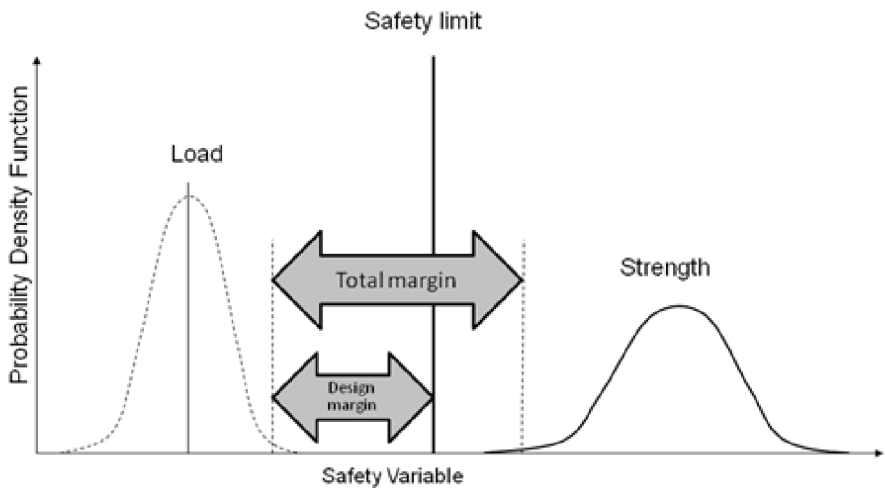
In this third case, the design limit is the least restrictive (least conservative), and the operational limit the most restrictive (most conservative).

The OECD Nuclear Energy Agency has conducted significant work on the issue of margins and limits. In the Safety Margin Action Plan [7], margin is defined as the

"difference" (conservatism) between the actual state (load) and damage state (strength). Since both load and strength involve uncertainties or probability distributions, that NEA task group examined safety margin in the context of risk assessment.

The following figure shows the probability densities for both load and strength, and the relationship between the two.

Figure 3: The relationship between load and strength in terms of margin



In this case, the allowed load is smaller than the assumed strength, and the difference between the two is the margin. Quantifying the total margin (to say nothing of the design margin) is difficult. Even under experimental conditions, it is not so easy to precisely define load and strength (e.g. whether to use tensile strength or compressive strength, and how the load should be applied). It is even more difficult to establish single lines (limits) to represent these distributions. Hence, the safety limit may be simple in concept but more difficult to apply in practice (especially if the limits are defined with a single explicit measurable parameter).

Of special interest in the Safety Margin Action Plan is the idea that there are two different types of margin: margin available to the licensee and margin available to the regulator. This is schematically shown in Figure 4.

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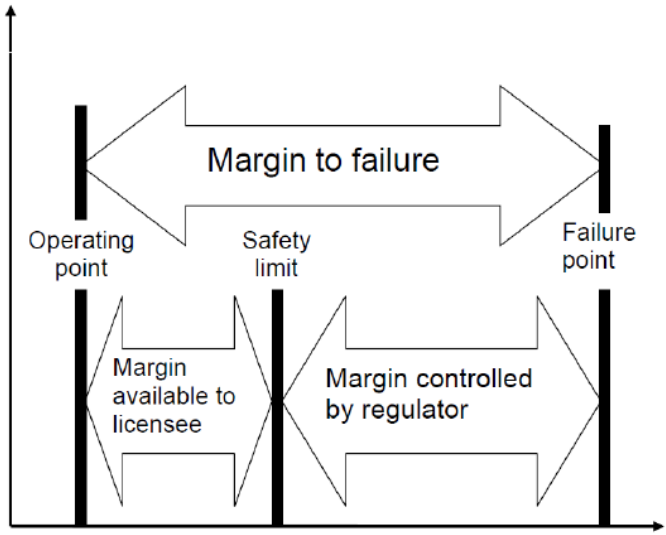
Figure 4: Types of margin



The concept of margin in the 2003 report on "Fuel Safety Criteria in NEA member countries" [2] is not strictly consistent with that used in the 2007 report on Safety Margin Action Plan [7]. The former report (on which this work is focused) presented the criteria as either *safety*, *operational* or *design* criteria. The criteria considered in the 2003 report are presented in Table 1.

Table 1: Types of criteria and technical issues identified in the 2003 NEA report on "Fuel Safety Criteria in NEA Member Countries"

Safety criteria	Operational criteria	Design criteria
DNB/CPR safety limit	DNB/CPR operating limit	Crud deposition
Reactivity coefficients	LHGR limit	Stress/strain/fatigue
Shutdown margin	PCI	Oxidation/hydriding
Enrichment	Coolant activity	Hydride concentration
Internal gas pressure	Gap activity	Transport loads
PCMI	Source term	FA fretting wear
RIA fragmentation	Control rod drop time	Clad diameter increase
Non-LOCA runaway oxidation	RIA fuel failure limit	Cladding elongation
LOCA-PCT		Radial peaking factor
LOCA-oxidation		3D peaking factor
LOCA-H release		
LOCA-long-term cooling		
Seismic loads		
Holddown force		
Criticality		
Burn-up		

In this table, it should be noted that some criteria (DNB/CPR and RIA) appear twice – once in the category of *safety* criteria, and once in the category of *operational* criteria. Hydrogen and hydriding appear in both *safety* and *design* criteria categories. Other technical issues, such as fuel melting, do not appear at all. This makes it difficult to directly apply either the categories or the criteria identified in the 2003 report on fuel safety criteria.

The Working Group on Fuel Safety generally agrees with concept of safety, operational and design criteria. The group further agrees on most of the technical issues identified in the previous report. However, those technical issues have now been placed in a single general category of *fuel (safety) criteria* along with some *other considerations* of interest.

Other changes have been made since the 2003 report on fuel safety criteria. For example, the related concepts of departure from nucleate boiling (DNB) and critical power ratio (CPR) have been combined into a single element called critical heat flux (CHF).

In summary, the following 23 fuel criteria are now considered in the current report, along with a number of "other considerations" (see Table 2):

Table 2: Fuel (safety) criteria and other considerations in this report

Fuel (safety) criteria	Other considerations
Critical heat flux	Core management
Reactivity coefficients	Mixed-oxide fuel
Criticality and shutdown margin	Mixed assembly cores
Fuel enrichment	Slow or incomplete control rod insertion
CRUD deposition	Axial offset anomaly
Stress/strain/fatigue	Cladding diameter increase
Oxidation and hydriding	Cladding elongation
Rod internal gas pressure	Radial peaking factors
Thermal mechanical loads and PCMI	3D peaking factors
Pellet cladding interaction (PCI)	Cladding stability
Fuel melting	
LHGR limits	
RIA cladding failure	
Fuel fragmentation	
Non-LOCA cladding embrittlement/temperature	
LOCA cladding embrittlement	
Blowdown/seismic/transportation loads	
Assembly holddown force	
Fretting wear	
Coolant activity	
Fuel gap activity	
Source term	
Burn-up	

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It has been mentioned during the discussion that the last safety criteria of the list (burn-up) has a different status than the others: burn-up is, indeed, legally limited in most countries but it is not a safety criteria *per se*. As a matter of fact, burn-up effect is (more or less) embedded in almost all the other safety criteria and thus accounted for, through the appropriate models included in the calculation codes used to design the fuel and through the burn-up adapted safety limits (e.g. new RIA and LOCA limits for high burn-up fuels).

The Working Group on Fuel Safety presumes that this list may be amended and supplemented. Whether an individual criterion applies during normal operation, anticipated transients, or accident conditions is noted in the text. Regarding relative conservatism of each category, margins may be set differently in different countries, and will thus depend on the technical and regulatory interpretation of the criteria.

2.3. Changes in fuel design and operation

The current fuel safety criteria were developed in the late 1960s to early 1970s and were based on tests and related analyses with the then utilised fuel and core designs, fuel and cladding materials (originally Zircaloy-2 for BWRs and Zircaloy-4 for PWRs), and burn-up levels not exceeding 40 GWd/t. In order to optimise fuel cycle cost, the nuclear industry began work in the mid-1980s on new fuel and core designs with the aim of increasing the fuel burn-up, by extending the number of cycles or the cycle length or upgrading the power level. This again leads to a number of basic design changes, for example new cladding alloys.

Fuel design should be in concordance with the general design criteria (e.g. Reference 8) that governs the design and operation of nuclear power stations including fuels. Thus, the existing fuel criteria should be examined against the general design criteria, as applicable.

High fuel burn-up is of great interest to some utilities due to their need for reducing fuel cycle cost, and may be enhanced by electric power deregulation. Thus, high burn-up capability is very important in the design of fuel, and this has triggered activities worldwide.

To achieve high discharge exposures and gain thermal margins, more advanced fuel designs were introduced. The fuel pin geometry changed from coarse pins with large fuel cladding diameters to slimmer pins with smaller fuel and cladding diameters thus reducing the heat flux per cladding surface area. The number of rods per assembly has increased in both BWR and PWR applications. The cladding wall thickness also has been reduced and lies today in the range of 700 to 600 µm (or even lower). Additional mixing devices were introduced to gain more margin.

In parallel with these dimensional changes, the cladding materials for light water reactor (LWR) fuel have also undergone significant evolutions. To reduce the corrosion rate and hydrogen uptake in the base zirconium metal, new alloys for

PWR cladding were introduced.[2] In addition to these changes in alloy composition, several inner and outer liner concepts have been introduced to cope with performance problems [e.g. BWR inner liners for stress corrosion cracking/pellet-cladding mechanical interaction (SCC-PCMI) resistance, PWR outer liners for corrosion reduction at high power].

2.4. Assessing the potential effects of changes

Numerous criteria related to fuel damage are used in fuel design and safety analyses: these fuel criteria may differ from country to country. Some are used to minimise cladding degradation during normal operation. Some are used to maintain cladding integrity during anticipated transients, thus avoiding fission product release. Some are used to limit fuel damage and ensure core coolability during design basis accidents, and some are used to limit the public risk from low probability severe accidents.

It can be difficult to categorise these fuel criteria according to event type. For example, limits are sometimes placed on cladding oxidation during normal operation to ensure good operational performance, while in other instances, such oxidation limits may be linked to cladding mechanical strength/ductility for LOCA performance.

The fuel criteria are therefore listed with only modest attempt to categorise them according to event type or risk significance. The matter of relative importance of these fuel criteria is left to the regulatory agencies and others who utilise this information. For each of the fuel criteria, a brief description of the criterion as it is used in several applications is provided along with the rationale for having such a criterion.

Design changes such as different cladding materials, higher burn-up, and the use of MOX fuels, can affect fuel-related margins and, in some cases, the fuel criteria themselves. In the following paragraphs, some of the more important effects are mentioned in order to indicate whether the criteria need to be re-evaluated. The following discussion may not cover all possible effects, but should be sufficient to identify those criteria that need to be addressed.

2. There is little evidence that niobium affects corrosion and hydrogen pickup. However, it is known to introduce dimensional stability. Niobium has been used in Russian alloys for years and is not new.

3. Review of fuel (safety) criteria

In this chapter, the possible implications from new design changes on all currently approved fuel (safety) criteria are discussed. An assessment of the need for re-evaluation will be given along with each individual criterion. Throughout this review, the basis for the safety criteria is assumed to be unchanged from the original basis.

Research is being conducted by various organisations around the world on the effects of new design changes such as different cladding materials, higher burn-up, and the use of fissile plutonium in MOX fuels. The working group has made an effort, through its members and through its contacts with the industry, to identify such research related to the individual fuel (safety) criteria and the need, if any, for additional efforts in this area.

3.1. Critical heat flux

Critical heat flux (CHF) or boiling crisis describes a thermal limit where a phase change in the reactor coolant occurs during heating. In a PWR, the CHF occurs when the bubble density from nucleate boiling in the boundary layer of a fuel rod is so great that adjacent bubbles coalesce and form a vapour film on the surface of the rod. Heat transfer across the vapour film is relatively low relative to the liquid, and the occurrence of CHF is accompanied by a marked increase in the cladding surface temperature. Under such conditions, rapid oxidation (or even melting) of the cladding can take place. This may result in cladding failures. Similarly, in a BWR the (critical) heat flux at the onset of transition boiling must not be exceeded.

In the BWR, the CHF is reflected by the critical power ratio (CPR), the ratio of the critical heat flux to the actual heat flux of a fuel rod. In the PWR, the critical heat flux is reflected in the departure from nucleate boiling ratio (DNBR), the ratio of the CHF (the heat flux needed to cause departure from nucleate boiling) to the local heat flux of a fuel rod. These ratios incorporate margin to the phenomena.

CHF correlations are derived from the analysis of experimental data from electrically-heated (unirradiated), large (usually full-scale) fuel bundles or arrays tested under laboratory conditions. The correlations make it possible to determine the critical heat flux over a wide range of test conditions, such as pressure and flow rate. The limiting DNBR is a safety limit defined such that fuel rods will not experience CHF during normal or expected operation (Condition I and II events). This limit is also used to indicate fuel failures for some postulated accidents (Condition III and IV events) in evaluating off-site dose rates. In other postulated accidents, such as the large break LOCA, most rods in the core are expected to exceed CHF and off-site doses are determined by other methods.

Since the correlation links the CHF to the test parameters under which it was derived, this correlation is a mathematical fit to test (not operational) data. The fuel supplier performs such tests for every assembly design specifically. Thus, the CHF correlation is fuel assembly type specific. The correlation parameters include pressure, mass velocity and flow quality; and tests are performed while varying each of these parameters separately.

The statistical method to establish the safety limit is sometimes based on a Monte Carlo technique, which calculates the critical heat flux for each assembly at multiple locations and under varying test conditions, while introducing random variations in the input variables based on their known uncertainties.

As a consequence, critical heat flux correlations may be considered to properly reflect the modern fuel and core designs; it is one of the few areas where statistical methods are applied consistently, with a rigorous uncertainty treatment. Fuel suppliers have developed critical heat flux correlations (e.g. W-3, GEXL, FC) that are successfully applied worldwide; to date, no fuel has failed due to inadequacies in establishing these safety limits.[1]

A remaining issue is that uncertainties in the experimental case do not necessarily represent uncertainties in the reactor core – particularly in local power levels – although that assumption is often made. Typically, DNBR safety limits are around 1.15. From this limit, it is presumed that, with 95% confidence and 95% probability, DNB shall not occur for maximum powered fuel rods under steady-state and AOO conditions. The issue is whether the "maximum powered fuel rods" in the core have been correctly captured.

CPR and DNBR limits ensure that only a very small amount of fuel cladding (0.1% of all fuel rods, in most countries: in Germany, DNB shall not occur for the highest rated rods) is statistically (95/95 level) expected to fail during anticipated operational occurrences (AOO), and indicate when fuel failure occurs during postulated accidents so that off-site doses can be estimated.

To also maintain adequate fuel performance margins during normal steady-state operation, additional margin is usually applied to the safety limit CPR/DNB, which corresponds to the heat flux increase during the worst AOO; this constitutes the operating limit that is continuously verified during plant operation.

It is unlikely that critical heat flux methodology, the related safety limits, or the methods used to establish these limits, would be subject to significant change. Some testing seems to be needed, including full scale testing to establish the proper thermal-hydraulic modelling of new assembly designs.

However, CPR and DNBR correlations are generally developed from data on unoxidised, or lightly oxidised, fresh cladding tubes and may not be accurate for high burn-up cladding. For a given linear heat generation rate (LHGR), the heat transfer coefficient for rough oxidised rods will be higher compared to smooth

1. The 1988 dry-out fuel failures in Oskarshamn 2 (Sweden) were caused by excessive channel bow and incorrect core monitoring model input data. A total of four rods operated around 20-30% in excess of the safety limit CPR for several months prior to failure.

unoxidised rods. That is, CPR/DNBR based on unoxidised rods is bounding in the absence of other heat transfer effects of the oxide layers.

Fuel rod heat transfer characteristics are likely to be affected by heavy oxide coatings (which sometimes exhibit spallation) that may appear on cladding at high burn-up, or by heavy CRUD[2] layers. Material and fabrication variations may make small changes in heat transfer characteristics, but the effect of oxidation or CRUD on surface conditions could be an important effect. Thus, the effect of oxidation or CRUD on surface conditions ought to be addressed, but not within CPR or DNB.

It has been noted that exceeding critical heat flux, particularly for a short period of time (a few seconds) may not adversely affect the cladding. Historically, the United States has maintained that cladding-to-coolant heat transfer must not exceed critical heat flux. However, Japan has been conducting research on post-boiling transition fuel integrity and has proposed a standard in their regulations for judging fuel integrity in cases of boiling transition and the reuse of assemblies that have previously undergone transition boiling [9]. The Japanese assert the need for three specific, accurate correlations in order to predict the fuel cladding temperature after boiling transition: (1) the onset time of boiling transition, (2) the heat transfer coefficient between cladding surface and coolant after boiling transition, and (3) a correlation to predict rewet time.

Finally, it is noted that significantly higher cladding temperatures due to operation in a film boiling regime may not affect alternative cladding materials (e.g. ceramics). Also, the phenomenon of transition to film boiling may not occur in some environments (e.g. sodium coolant or very high pressure light water reactors where the coolant may be operated beyond the vapour-liquid critical point for water at 22.1 MPa and 374°C).

Current CPR and DNBR safety limits from NEA member countries are shown in Table 3.

2. CRUD – chalk river unidentified deposits – corrosion products deposited on the fuel rod cladding, first identified at Chalk River Laboratories in Canada.

Table 3: DNB/CPR safety limit

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of recent changes	Type of methods and rational
Belgium	CHF/DNB	Depending on the correlation used	95/95, correlations: W3 (for low pressure), WRB-, HTP, ERB-, FC, ABBX-, ...depending on the fuel supplier	DNB operating limit	No change, verification required (new design, mixed cores)	Statistical
Canada	CHF		Correlations (e.g. Balint-Cheng)			
Czech Rep.	DNB	Depending on the correlation and TH models used	95/95 VVER440: variants of Czech correlation of PG type 95/95 VVER 1000: special Russian correlation CRT-1 for bundles with mixing vanes	3-D peaking at VVER440, DNB oper. limit at VVER1000	Verification required	Statistical
Finland	DNB/CPR	1.33/1.06	95/95/<0.1% of rods may experience DNB, correlations		No change (burn-up limit 40 GWd/t)	Statistical
France	CHF	1.17, 1.30	95/95, correlations (WRB-, W-3 for low pressure)	Operat. limits (e.g. axial offset)	No change, verif. required (new design, mixed core)	Statistical
Germany	DNB/CPR	1.15/1.09	95/95-correlations (PWR), <1 rod experience dryout -- THAM method (BWR), all correlations are FA specific	Addit. oper. crit.	Values change dep. on design	Statistical
Hungary	DNB	1.33	95/95, correlations (Bezrukov)	Pin power limit	Values change dep. on design	Statistical
Japan	DNB/CPR	1.17*/1.06*	95/95/<0.1% of rods may experience DNB, correlations (e.g. MIRC-1, NFI-1)	DNB/CPR oper. limit	No effect	Statistical
Korea (Rep. of)	DNB	Depending on the correlation used	95/95, correlations (KCE-1, NGF, WRB-1 etc.)	DNB oper. limit	No change, verif. required	Statistical
Netherlands	DNB	1.30	95/95, correlations (W-3)	DNB oper. limit		Statistical
Spain	DNB/CPR	Various	95/95, correlations provided by fuel vendors	DNB/CPR oper. limit	Values change dep. on design	Statistical
Sweden	DNB/CPR	1.17/1.06	95/95, correlations (VRB-1)<0.1% of rods may experience dryout	DNB/CPR oper. limit	Values change dep. on design	Statistical
Switzerland	DNB/CPR	1.15-1.45/1.09	95/95, correlations	DNB/CPR oper. limit	Values change dep. on design	Statistical
UK	DNB		95/95, correlations	DNB oper. limit		Statistical
USA	DNB	various	95/95, correlations	DNB oper. limit		Statistical

*Not criteria but typical value.

3.2. Reactivity coefficients

The concept of reactivity coefficients has been introduced in order to simplify the analytical treatment, e.g. quantifying the feedback reactivities in the point kinetic equation and increase our understanding of reactivity changes due to various physical parameters. Reactivity coefficients are thus an analytical matter;

in terms of LWR safety criteria, there is a general requirement that either the moderator temperature coefficient (see below) or the total of all reactivity coefficients be negative when the reactor is critical, for providing negative reactivity feedback (or that the effects of any positive reactivity coefficient be inconsequential).

The reactivity coefficients depend on the following five reactor core state variables which are to some extent independent of each other:

- Fuel temperature T_f.
- Moderator (coolant) temperature T_m.
- Steam volume (void) fraction in the coolant (μ).
- System pressure P_s.
- Boron concentration.

The fuel temperature or Doppler coefficient $d\rho/dT_f$, where ρ is reactivity, responds promptly to the energy deposited in the fuel, whereas the other coefficients are delayed. The fuel time constant (on the order of a few seconds), which depends mainly on the fuel specific heat, conductivity and diameter, affects the time delay of changes in moderator temperature and void fraction. The fuel temperature coefficient therefore depends slightly on the enrichment and the fuel burn-up – the higher the burn-up, the harder the spectrum, so in general the change of the fuel temperature coefficient with burn-up is small in light water reactors.

The strong negative void coefficient in BWRs gives these reactors inherent stabilising characteristics without operator intervention. In modern fuel designs, water is added in the central part of the bundle by special water channels of various geometries inside the fuel assembly, which is not heated up as much as the coolant water in the rest of the assembly and has a much lower void fraction thus producing a less negative void coefficient.

In PWR under normal operating sequences there is no void in the core. However, in the case of abnormal events like loss of primary coolant or loss of pressure, the coolant may start to boil and void appears and reduces the neutron absorption in boron which results in a positive contribution to the void coefficient (however there is also less moderator available). At operating temperature when the boron concentration is low, this effect will be small and the void coefficient remains negative. At low temperature when the boron concentration is high, the effect is large and the void coefficient may turn positive.

An increase of the moderator/coolant temperature T_m causes mainly two effects:

- the density of the water decreases and the effect is similar to that of void increase;
- the thermal neutron spectrum becomes harder and so the effective neutron cross-sections change.

In a PWR with a strongly borated coolant $d\rho/dT_m$ is negative at normal operating conditions but is slightly positive at lower temperatures. Due to the boron concentration decrease at the end of the cycle, the moderator temperature coefficient is becoming also more negative at the end of the cycle. This has some impact on cooling down accidents such as the steam line break accident because more positive reactivity is introduced from the cooling and the reactor returns to a higher power level than before.

The system pressure in a BWR is related to the saturation temperature of the moderator. Depressurisation of the system will cause flashing, i.e. production of steam bubbles in the water. Such an event introduces a negative reactivity change in a BWR and does not lead to safety problems as far as reactivity is concerned.

The effect of a positive pressure pulse is only of interest in a BWR, where significant voiding exists. A sudden increase of the system pressure, e.g. one caused by a turbine trip, will result in a partial void collapse leading to a positive reactivity change.

In order to have the same cycle length, high fuel burn-up usually implies the loading of more reactive fresh fuel bundles. This additional reactivity is compensated for by fuel (addition of burnable poison) and core design, keeping in mind that the basic safety criterion (either the moderator temperature coefficient or the negative total reactivity coefficient) must be fulfilled.

In summary, although the reactivity coefficients may be affected, the effects of new fuel design changes are not considered to affect the corresponding safety criteria themselves.

3.3. Criticality and shutdown margin

Attaining reactor subcriticality must be assured either by sufficient reactivity worth of control rods and/or sufficient boron concentration in the primary coolant.

For control rods, this subcriticality requirement becomes the so-called shutdown margin (SDM). SDM is defined as the margin to criticality ($k_{eff} = 1$) in the situation with all control rods inserted and the strongest control rod withdrawn. The SDM should be sufficient for achieving hot zero power; for the BWR, SDM is analysed at cold zero power with a xenon-free core, for conservatism. The technical specification limit for SDM, usually of the order 0.3 – 0.5% $\Delta K/K$, is mostly established from the assumed envelope of uncertainties in the determination of k_{eff} and the control rod manufacturing tolerances. This limit is usually verified at least during (reload) cycle startup; design limits for SDM are usually 1% $\Delta K/K$ or higher, to protect against unforeseen systematic biases in the prediction of the k_{eff} value.

The required subcriticality in shutdown state is given in the operating technical specifications (OTS). The required subcriticality is met by adjusting the required boron concentration for a given RCCAS position (e.g. controls rods inserted, shutdown banks out). The SDM corresponds to the amount of subcriticality after a trip with the assumption of a stuck rod. The SDM value is taken as the initial subcriticality for the steam line break accident analysis.

For the PWR, an increase of boron concentration is required to achieve cold shutdown; this is provided by the available boron/volume control systems. Generally, for PWR and BWR, the boron SDM is the margin to criticality (k_{eff} =1) for the situation in which the emergency boron injection system is activated.[3] The (high) boron concentration should be sufficient to assure that the reactor achieves shutdown without control rod movement; for conservatism, no credit is taken for xenon present in the core. Emergency boron SDM limits are established similar to the above control rod SDM limits, i.e. those based on calculational and system uncertainties. Values for the emergency boron SDM range from 1 to 4% $\Delta K/K$, depending on whether the analysis is performed using generic and/or cold reactor conditions or more realistically reflects specific plants/cycles. Normally, the emergency boron SDM is not explicitly included in plant technical specifications, but is rather verified analytically as part of the safety analysis and reload licensing process.

Highly optimised core designs have often shown a decrease in margin to the SDM criteria (usage of higher enrichment levels, often in conjunction with more burnable poison). However, modern fuel designs are also optimised to improve the SDM performance, and may counteract these effects. These fuel enrichment and core design strategies are provoked or enhanced by operating strategies to save fuel cycle cost, which includes high fuel discharge exposures, long fuel cycles and/or thermal power uprates. In the case of MOX fuel, smaller control rod worth and boron worth have also reduced the SDM performance.

These reduced margins have, in some cases, induced plant changes such as:

- use of new control rods with higher worth (more/different absorbing material);[4]
- higher number of installed control rods (if plant design permits);
- increase of boron system capacity (if possible);
- use of enriched boron.

in order to compensate for the lost margin. Ultimately, fuel and core must be designed such that safety criteria are met; these criteria have not been challenged so far.

It is judged that the existing SDM criteria themselves are unaffected by the new design changes. However, if realistic or best-estimate modelling is used to establish or analyse these criteria, such models should be well verified; in particular, the associated modelling uncertainty should be quantified in order to assess the margin to safety.

3. The SDM is not an OTS value in France.
4. Use of part-length or limited worth ("gray") control rods will reduce the RCCA worth and will give no benefit for the SDM.

3.4. Fuel enrichment

Enrichment limits around 5 wt% U^{235} are used in connection with criticality considerations for fabrication, handling, and transportation. For some high burn-up applications, higher enrichments may be needed. To date, the validation of criticality safety codes and associated cross-section libraries for LWR fuel has focused on enrichments of 5 wt% and below. Neither benchmarks of code performance nor the bases for extrapolating code performance in the enrichment range of 5-10 wt% have been well established. Moving into this range will require care because the physics of criticality begins to change as enrichments reach 6 wt% and beyond, where single moderated assemblies can go critical and criticality of weakly moderated or unmoderated systems becomes possible. Enrichments above 5 wt% will require redesign of some fuel fabrication and handling equipment and fuel transportation packages. The possibility of recriticality during accidents, in particular in severe accident core melt sequences should also be addressed as this could alter the progression of such accidents.

The International Atomic Energy Agency (IAEA) has reviewed data on the current status and future trends of global high enrichment inventory [10] and conducted the water reactor fuel extended burn-up study [11] to assess the economic effects of burn-up extension. That study evaluated uranium utilisation for PWRs and BWRs (e.g. the need for higher enrichments and possible fuel design changes) and considered environmental, safety and licensing implications. Other burn-up extension and optimisation studies were sponsored by the Electric Power Research Institute (EPRI) and the US Department of Energy. These studies [12-13] evaluated the economics and obstacles to the burn-up optimisation of both PWRs and BWRs for enrichments of 5% and greater.

Commercial fuel with enrichment higher than 5 wt% also has been discussed by the OECD/NEA Nuclear Science Committee Working Party on Nuclear Criticality Safety (NSC/WPNCS). Work is currently ongoing in Japan, but has not yet risen to the level of consideration by CSNI/WGFS.

3.5. CRUD deposition

The maximum amount of CRUD deposited on the cladding can be estimated. This is sometimes done as a function of burn-up (and/or power) but at least at the end of the fuel lifetime. The maximum value has to be considered, and the assumed value is verified against data from measurement [5] (e.g. hot cell examinations). Various CRUD levels are being used by vendors, according to the design models and/or the fuel designs themselves. Firm (safety) limits on CRUD deposition are not defined, although the amount of CRUD deposited and its composition can be significant to the corrosion performance of the cladding (example: CRUD induced localised corrosion or CILC). In addition, investigators should consider the various forms of CRUD (tenacious, fluffy), the thermal

5. The issue of post-irradiation measurement of CRUD thickness is controversial as some CRUD dissolves on shutdown.

conductivity of CRUD, the effects on cladding-to-coolant heat transfer, and when it occurs or when it is deposited (during operation/during shutdown).

New design changes, such as cladding materials and their manufacturing processes (e.g. surface finish), may well influence the build-up of CRUD and thereby the corrosion performance of the fuel clad. The CRUD composition could affect the corrosion locally, either by acting as a thermal insulator or by chemically favouring the corrosion process. Also the water chemistry characteristics influence the type and character of the CRUD build-up: as an example, the ratio of 2-valence to 3-valence components in the reactor water (e.g. Zn/Ni or Fe, respectively) could determine the type of CRUD (spinel vs. hematite), thus influencing the corrosion rate in a different manner.

Experience has shown that the most important factor to consider when implementing the chemistry strategies is to address the correlation between CRUD deposition and corrosion kinetics at the same time, because some practices that can be good from one perspective but poor for the other. New fuel and core designs, high burn-up and long fuel cycles are issues that could influence CRUD build-up through associated changes in cladding materials, surface area and power history. No specific limits are directly imposed regarding maximum acceptable CRUD levels, but its influence has to be considered both on the thermal models as well as on the corrosion kinetics models. An acceptable amount of CRUD might be also driven by operational limits on AOA (axial offset anomalies), especially if the CRUD deposition is not homogeneously distributed within the core.

With the severe thermal duty that occurs with fuel management strategies supporting extended cycle length and core power uprates, and in order to reduce radiation levels in plant components, strategies with modified water chemistry with e.g. higher lithium concentration, resulting in higher pH values, or with the injection of Zn or Fe into the primary coolant for reducing dose rates or increased corrosion protection, have been introduced in the last years. This chemistry, for example, has proven to be adequate to control CRUD deposition. Notwithstanding that, as the plants in transition to longer operating cycles require extra loading of soluble boron at beginning-of-life, to maintain the pH at the required level (around 7.2) with this boron concentration the fuel has to be operated with high lithium concentration – above 2.2 parts-per-million (ppm) for some time, which could increase the corrosion rate. There is also a concern with large porous CRUD depositions in PWRs leading to boron pick-up, thereby causing distortion of the core axial power profile and reduced SDM. Usually, the transition to a modified water chemistry is guided by monitoring the corrosion behaviour of the fuel rod cladding.

It must be noted that the industry is undertaking efforts to improve the knowledge of possible effects of water chemistry, based on accumulated experience and research work, and to incorporate this improved knowledge in a number of reactor water operating guidelines [14-15].

In the past, criteria on CRUD deposition were considered “derived” criteria, and only indirectly safety related. However, unexpected large amounts of CRUD were observed at the River Bend facility (United States) during the Cycle 8 and Cycle 11 outages. CRUD-induced localised cladding corrosion failure of fuel was observed.

The root cause appeared to be thermally-induced accelerated corrosion, due to elevated iron and copper deposits associated with a chemistry excursion during the operation. Much of the copper originated from the use of brass condensers, which are being phased out, and with careful monitoring and control of the feed-water iron and copper the issue is now better managed.

Although no firm limits were previously established for CRUD, the NRC received a petition for rulemaking on this issue [16]. Specifically, the petitioner asked the NRC to require that nuclear power reactors be operated in a manner to limit the thickness of CRUD layers and/or the thickness of oxide layers on fuel rod cladding surfaces to ensure that the facilities operate in compliance with the emergency core cooling system (ECCS) acceptance criteria. The petitioner also requested that the requirements pertaining to ECCS evaluation models be amended to explicitly require that the steady-state temperature distribution and stored energy in reactor fuel at the onset of a postulated LOCA be calculated by factoring in the role that the thermal resistance of CRUD and/or oxide layers on fuel cladding plays in increasing the stored energy of the fuel.

Although this petition has not yet resulted in any explicit regulatory change, the NRC has determined that the rulemaking requests in this petition, known as PRM-50-84, will be considered in the ongoing 10 CFR 50.46(b) "Performance-Based ECCS Cladding Acceptance Criteria" rulemaking, which is currently ongoing in the United States.

3.6. Stress/strain/fatigue

Generally conservative design limits are taken for cladding stress (e.g. around 0.2% yield or tensile strength at operating temperature).

For strain, there are several different but closely-related limits. To add to the confusion, there are actually several forms of the so-called "1% strain" criterion in use.

The first "1% strain" criterion applies on the long-term strain that occurs after gap closure induced by outer overpressure (creep down). The process includes thermal expansion, but it is dominated by the swelling process. There are two different values used, the 1% strain criterion relates to the tangential (circumferential) strain only, and the 2.5% strain criterion relates to the combined tangential and axial strain, the so-called equivalent strain, which is the vector addition from tangential and axial direction. The 1% tangential strain and 2.5% equivalent strain are approximately equivalent in terms of cladding load.

This strain limit applies for Condition I and II events. This pellet-cladding mechanical interaction (or PCMI) phenomenon is caused by a combination of cladding creep, fuel swelling (which reaches a maximum at end of fuel life), fuel rod internal pressure (Section 3.8), and fuel pellet thermal expansion. The margins from these limits to actual failure stresses and strains are defined from the fuel vendor's database for a particular fuel, cladding, and burn-up range.

In some countries, the 2.5% criterion has been replaced by the value of 3.5%. The reason for the increase from 2.5% to 3.5% was the need for higher burn-up

(>60 MWd/kg). It has been shown that, for some cladding materials, that 3.5% strain will have enough margin to failure.

The second 1% strain criterion is used to define the maximum load that can be sustained by a fuel rod for a short time. This strain criterion (tangential, transient strain) was postulated independent of the DNB criteria, which should limit the power before the 1% strain limit is effective.

Up to now, this second criterion has been used as a pellet-cladding mechanical interaction (PCMI) criterion for operational over-power transients (Condition II events). However, it is clear that this criterion alone will not prevent the cladding from PCMI or PCI failures. The DNB-criterion may be more restrictive.[6]

For PCI prevention operational rules are established, e.g. limiting the absolute value of power increase and the power increase velocity. These rules apply above a certain power level and depend on the so-called cladding condition, which is the time at a certain power level.

The cumulative number of strain fatigue cycles on each fuel assembly structural member is assumed to be significantly less than the design fatigue lifetime, which in turn is based on appropriate data and usually includes a safety factor (2) on stress amplitude and a safety factor (20) on the number of cycles.

These strain and fatigue limits, together with others such as SCC-PCMI and fuel rod internal pressure, are used by some fuel vendors to define the fuel specific thermal-mechanical limit; this limit is expressed as a burn-up dependent linear heat rate curve (in W/cm). In some applications, the curve conservatively bounds all thermal-mechanical phenomena, including stress, strain and fatigue limits; it is set to cover the transient thermal/mechanical power limit, which ranges from about 10 to 50% above normal. Some vendors invented statistical methodologies (Monte Carlo analyses) to introduce integral concepts (explicit cycle verifications for long-term behaviour in combination with operating limits for transients) for assessment of fuel behaviour during reactor operation (Conditions I and II).

These thermal mechanical analyses are performed by the fuel vendor, with models that are updated against new experimental test data as well as against data from operational feedback as they become available. Analysis of new designs and design changes can also be obtained from sophisticated fuel performance codes. The use of such codes allows the expected fuel duty to be modelled and therefore assists in investigating the effects of new core designs or revised operating practices as well as the effects from a modified fuel design.

Mechanical and physical properties used in these fuel performance codes depend on parameters like material composition, fabrication (including heat treatment), fluence and hydrogen content. The properties may be affected by new design changes and, in particular, by high burn-up. Hence, continuous verification and validation of fuel design models is essential to ensure that the proper basis for design and operation exists.

6. Due to brittle characteristics of cladding caused by oxidation and hydriding in high burn-up fuel, the DNB-criterion may not be more restrictive, depending on the circumstances.

3.7. Oxidation and hydriding

Oxidation and hydriding under normal reactor operating conditions directly impact fuel performance, not only during normal operation, but during transients and accidents as well.

Cladding corrosion or oxidation degrades material properties, most importantly the effective cladding-to-coolant heat transfer (with a subsequent increase in fuel temperature and thus the stored energy of the fuel). Hydrogen absorption by the cladding and subsequent formation of hydrides may lead to cladding embrittlement; these phenomena are increasingly important at higher exposures for some of the existing alloys; the significance of the issue is much reduced for modern alloys with improved corrosion and hydrogen pickup performance. For these reasons, the composition and fabrication of zirconium-based cladding materials have become highly optimised. For BWRs, where Zircaloy-2 is used, the thermal-mechanical processing is optimised to result in an intermediate second phase particle (SPP) size. Too large a SPP size could result in nodular corrosion, but too small a SPP size could result in higher uniform corrosion. Although Zircaloy-2 continues to be used in BWRs, Zircaloy-4 has been largely replaced in PWRs by low-tin outer liner cladding concepts (DUPLEX) or by Zr-Nb and Zr-Nb-Sn alloys (e.g. ZIRLO, M5, MDA and E110).

Uniform oxidation or corrosion rates differ between PWRs and BWRs.[7] With the lower operating coolant temperature, uniform corrosion is much less critical for BWRs; in contrast, PWRs are less susceptible to nodular or local phenomena (e.g. CRUD-induced localised corrosion, enhanced shadow corrosion) due to much less oppressive heat transfer and flow conditions as well as a reducing chemical potential at low oxygen level[8] (see below). Regarding uniform corrosion and hydrogen pick-up, Japanese research results show the following tendency:

- For BWR Zircaloy-2 cladding, corrosion is low, however, hydrogen pick-up rate increases significantly as oxide thickness increases. Hydrogen pick-up may also depend on in-core residence time [17-18].
- For PWR cladding, hydrogen pick-up rate remains almost constant up to high burn-up. However, due to the thermal feedback effect on corrosion temperature uniform corrosion increases at higher oxide thicknesses/high burn-ups[9] which is of special relevance for non optimised cladding

7. The heat flux overlaps between PWRs and BWRs. The lower operating temperature of BWRs means lower oxide and thus lower hydrogen pickup and thus the cladding typically does not form a hydride rim. The formation of a hydride rim in PWRs causes corrosion acceleration which lead to further thermal feedback.
8. The reduced oxygen potential is more important and could be the primarily reason for the difference.
9. The understanding of the accelerated corrosion at high burn-up may be unclear. As the oxide thickens (at higher burn-up), the temperature may not necessarily increase due to the lower power level at elevated burn-up. The accelerated corrosion may be due to the formation of hydride rim, which only applies to alloys with hydrogen pickup of more than ~350 ppm.

materials (e.g. standard Zircaloy-4). This effect depends on power and the power history [19].

High duty operation (high coolant temperatures, high enrichments, high power levels, high burn-ups) should be monitored by post-irradiation examination.

Hydrogen may be added to the coolant to decrease the amount of oxygen present, which is formed by radiolysis, which consequently decreases the rate of zirconium oxidation (this is a remedy for PWRs only). There are no conclusive evidence that increased hydrogen pickup is dependent on the coolant hydrogen concentration. However, if the hydrogen concentration is increased too much, cladding hydriding and subsequent embrittlement could be increased. In some cases hydrogen is added to reduce the recirculation piping radiation dose rates; noble metals may be injected simultaneously, to limit the amount of added hydrogen.

For design purposes, oxide thickness and hydride concentration limits are normally assumed at end of fuel life. For traditional alloys (e.g. Zircaloy) values are usually in the range of 100 micron and 500-600 ppm (wall averaged value), respectively; these values are taken from post-irradiation examination (PIE) data, and represent reasonable bounds on data measured from fuel exposed in commercial PWRs. The same limits are sometimes also used for modern alloys that exhibit a much lower corrosion propensity. The relevance of these criteria is questionable in this case because such high levels of oxidation and hydriding have never been observed under reactor operation conditions.

Post-irradiation examination data from high burn-up fuel show that local oxidation thickness and hydrogen content may exceed current design limits. From a cladding integrity point of view, local oxide thickness and hydrogen contents could be more limiting and important than average ones. Especially, in the case of hydrogen content, circumferential non-uniform distribution or hydride rim might be important factors for cladding's mechanical integrity during normal operation and postulated design basis accidents.

In several countries the design limits of an average cladding oxide thickness at end of fuel life of 100 micron, and of an average hydride concentration of 500-600 ppm, have effectively become approved safety criteria via the approval of fuel vendor design methodologies (note: unfortunately the interpretation of these criteria is not unambiguous, because the cladding region over which the average is taken is often ill defined). Also criteria limiting the number of cladding defects due to oxidation are found in some cases. In other countries no explicit limits are defined; in all cases, however, oxidation and hydriding are considered when analysing cladding properties for performing stress and strain-related fuel rod design evaluations.

An oxidation effect which has not been much considered in the past and which could be important is fuel bonding induced internal oxidation of the cladding (it may be called "chemical bonding" instead of "oxidation"). At increasing burn-up the pellet-cladding gap in the fuel rods tends to close due to swelling and cladding creep-down and a bonding layer is formed between the pellet and the cladding. This bonding layer may have a deteriorating effect on fuel rod behaviour

under irradiation, but may have a beneficial effect for pellet-cladding interaction/stress corrosion cracking (PCI/SCC). This has been proven from high burn-up ramp tests at high burn-up and post-tests examinations: beyond a certain burn-up threshold (around 45 GWd/t) fuel rods do not fail any longer by SCC-PCMI; it is assumed that the bonding layer acts as a barrier to the corrosive fission products (iodine) generated in the central part of the pellets. The bonding layer acts also as a media favouring the stress redistribution, thus limiting the stress concentration and potential for SCC. However, if the cladding is heavily hydrided, another type of failure mechanism may be activated: the outside-in PCMI failure mechanism (the crack is initiated in the outer hydride rim layer) which is a very different mechanism from the inside-out SCC-PCMI mechanism (the crack is initiated on the inner side of the cladding).

In general, the bonding layer prevents the axial transport of fission gases in a fuel rod (gas transport is also working via internal gaps and cracks) and induces a severe pellet cladding mechanical interaction (interaction is driven by the expanding pellet – swelling, thermal expansion). Those phenomena have to be accounted for in all transients implying significant thermal changes in the fuel-pellet composite.

Depending on the maximal temperature reached during a transient, internal oxidation of the fuel cladding is becoming increasingly important as a function of bonding and fuel burn-up. At high fuel burn-up the bonding effect is important; full bonding begins to occur at fuel burn-up of about 40 GWd/t (depending on the fuel rod design, e.g. in BWRs, fuel rod first closure is found at this level). Bonding causes diffusion of fission products such as iodine, cesium and cadmium into the cladding. At high temperatures, these effects cause internal oxidation and embrittlement of the fuel cladding and should be considered when assessing the effects of oxidation. NRC's recent rulemaking efforts [20] on LOCA criteria in the United States suggests the NRC is planning to impose a two-sided oxidation requirement. Similar inner and outer diameter oxygen stabilised alpha phase thickness observed in the Halden test was cited in support of the requirement, in which the Halden report suggested bonded fuel supplied the oxygen for the inner diameter oxidation. However, Argonne National Laboratory test data indicated the inner diameter oxidation does not occur to the same extend as outer diameter oxidation, suggesting the oxygen source on the inside is the limited. Further testing sponsored by EPRI suggests the oxygen stabilised alpha phase observed in the Halden test can be generated from the oxygen contained in the inner cladding surface oxide (not from fuel). There is clearly some inner cladding surface oxidation, however, the actual extent and impact on the cladding performance is still not quantified.

In some countries, there are no formal criteria related to oxide thickness and hydride concentration. This was considered justified by the fact that oxide thickness and hydride concentration are not directly responsible for fuel failure. However, oxide and hydride influence stress and strain performance, and ultimately the fracture toughness, of the cladding material. There is an obvious direct influence on the initial condition of the fuel rod assumed in the transient and accident analyses, as well as on the level of safety relevant parameters such as fuel temperature and internal pressure. Moreover, Japanese research results [17]

show that cladding ductility decreases with cladding hydrogen content (ductility limits are derived directly from experience feedback which indirectly includes such effects). Therefore, consideration should still be given to impose limits on oxide thickness and hydride concentration (at least to consider possible effects on general fuel rod behaviour).

Extremely high hydrogen content may affect current 1% strain limit.[10] Another concern is hydride re-orientation. It is known that hydride re-orientation will occur under certain temperature histories and tensile stress conditions, especially for recrystallised cladding. Radial hydrides may result in more degradation of cladding mechanical properties. However, recent test results indicate this degradation is mostly recovered when the temperature of the cladding is raised to above 100°C [21]. Such hydride re-orientation will occur for fuel rods in which internal pressure exceeds coolant pressure (only in the case of *high internal overpressure*, which can induce *relevant high stress* levels). High burn-up BWR fuels examined suggest such hydride re-orientation [22]. When a non lift-off internal pressure criterion is adopted, attention should be given to hydride re-orientation.

Additional issues, such as the oxide cladding spalling and very high local concentration of hydrides in the cladding wall, are not covered by the present limitations but are addressed when RIA limits are determined. Also, from a LOCA performance point of view, a high hydrogen content may lead to degradation of residual cladding ductility [23]. In addition, non-uniform hydride distribution should be discussed because, at the current stage, circumferentially averaged oxide thickness and hydrogen content are the only criteria considered. Indeed, ductility is much more governed by the maximum hydride concentration (weakest point) than by the average value. More tests are needed to confirm these effects.

Extensive research, including tests on corrosion rates, and fuel inspection programmes in commercial nuclear power plants (NPPs) has led to a basis for burn-ups well beyond 40 GWd/t, and some data are available at even higher burn-ups. Cladding containing advanced zirconium alloys and multi-layer type fuel claddings have been developed and tested out under irradiation in commercial NPPs (lead test rods or lead test assemblies, some of these programmes with subsequent destructive examination). In addition, some tests to cover transient and accident fuel performance have been made. This information has led to full core loadings, and in some countries the nuclear industry has the possibility to continue to push burn-up levels substantially beyond 60 GWd/t level, if it is economically justified.

In summary, as corrosion of some of the traditional zirconium-based alloys is probably one of the leading parameters that limit the lifetime of nuclear fuel, there is a rationale for reviewing the adequacy of the current applicable limits on maximum local oxidation and hydriding levels in the cladding, especially in view of the performance of highly burnt fuel.

10. Cladding materials have been tested with significantly high hydrogen content and survived at the 1% strain limit. Only in the cases of stress corrosion cracking does the cladding fail below 1% strain.

3.8. Rod internal gas pressure

Fission gas release and resulting fuel rod internal pressure is an important aspect of fuel behaviour. Traditionally it has been a limiting factor in setting the thermal-mechanical limit.

The fission gas release is dependent on a) the fuel microstructure and chemistry, b) its development with time, and c) the fuel temperature, which is strongly influenced by the power rating and the burn-up. At high burn-up (higher than 40-60 GWd/t) fission gas release tends to increase rapidly (continuously). Also available experiments involving fission gas release under transient conditions, indicate very high fission gas releases in the high burn-up region of the fuel; furthermore, fission gas release is strongly influenced by the formation of the peripheral fuel rim at high burn-up which is especially important for transient/accident conditions. These phenomena are not yet well understood, nor can existing analytical tools predict them satisfactorily but significant progresses have been made in the last few years, thanks to separate effects in-pile tests and appropriate advanced models.

Increases in fission gas release can lead to high fuel rod internal pressures and could also lead to a deterioration of the thermal conductivity of the gas in the plenum/fuel rod free volumes and, more importantly, of the heat transfer between the pellets and the cladding due to the resulting gap size modification. The fission gases Xe and Kr decrease the thermal conductivity of the helium gas in the gap, which increases the fuel temperature; when the gap is closed, this effect becomes less significant. This induces a feedback mechanism since an increased fuel temperature enhances the fission gas release. Due to the above mentioned thermal feedback mechanism and the sensitivity to changes in rod power level, the fission gas release in various rods can be highly irregular.

The high internal rod pressures can have an important effect on fuel cladding under transients and postulated accidents behaviour (ballooning, burst, etc.). For example, during a LOCA, the pressure differential across the cladding wall may be inverted within seconds due to early complete system pressure drop.

Two alternative criteria for acceptable internal gas pressure are currently used in various countries by their regulatory authorities. In the first option, the rod internal pressure is held below the nominal pressure in the reactor coolant system (RCS) during normal operation in order to prevent outward creep of the cladding. In the second option, the rod internal pressure may exceed the RCS pressure, but is limited so that the cladding creep-out rate due to an internal rod pressure greater than the reactor coolant system pressure is not expected to exceed the fuel swelling rate, i.e. the fuel to cladding gap does not open (this is the so-called “no lift-off” criterion). The two alternative criteria – absolute (reactor coolant system pressure) or relative (cladding lift-off) – for acceptable internal gas pressure are currently used in various countries by their regulatory authorities. At high fuel burn-up either criterion could, in transient and accident conditions, lead to a high internal pressure of the fuel rod with a possible effect on stored energy, cladding ballooning and bursting, which could challenge core coolability, and thus the level/limits resulting from this safety criterion.

The consequence of rod internal pressure build up at elevated burn-up has been extensively studied for the last ten years. One such study, a lift-off test series (IFA 610) with UO_2 and MOX fuel rods of 50-60 GWd/t was performed at the Halden research reactor [24]. For the UO_2 rod, lift-off occurred at an overpressure (above the system pressure) of around 130 bar; results for the MOX fuel rod indicate an even higher lift-off pressure. On the basis of the Halden findings, as well on the basis of the Studsvik ROPE (Rod Over Pressure Experiments) data, the "no lift-off" criterion was proposed by some vendors, thus considering outward creep rate/strain and tensile stress due to overpressure. Nevertheless, the high over-pressure thresholds found in the Halden experimental programme show there are significant margins regarding cladding lift-off, but they cannot be used directly as safety criteria; rod internal pressure has to be limited to comply with other requirements, in particular for the spent fuel (to avoid hydride reorientation during transportation in dry casks for instance).

For MOX fuel, the fission gas and helium releases are higher as compared to UO_2 fuel, and the adequacy of the relative (lift-off) criterion for acceptable internal fuel rod gas pressure may also require further study. This is not a matter of the MOX fuel itself but more a matter of the related reactivity history expressed in the power histories (see next sentence). An acceleration of fission gas release with exposure is observed, also because of the higher linear heat generation rate in high burn-up MOX due to the higher reactivity level. The reactivity level itself may not be higher, but the power histories are more demanding. The development of rod internal pressure as function of burn-up on MOX fuel needs to be well characterised, also in consideration of the production method and plutonium content in the MOX fuel.

These criteria should not be fundamentally affected by design changes, although methods to demonstrate compliance will be affected.

3.9. Thermal mechanical loads and PCMI

Pellet-cladding mechanical interaction (PCMI) refers to the stress and/or strain on the cladding from an expanding pellet, especially during a transient. Pellet expansion results mainly from thermal expansion and gaseous swelling, and if the stress is large enough it can result in cladding failure. PCMI differs from the related SCC-PCI phenomenon inasmuch as the latter refers to power ramps (with sufficiently high power levels, sufficiently high ramp rates) where the stress is held for a relatively long period of time and corrosion is necessary for cracking to take place.

The avoidance of mechanical fracture of the cladding during transients due to pellet-cladding mechanical interaction, which is the basic safety criterion, is partially covered by the limit on uniform cladding (plastic and elastic) transient strain of 1%. However, PCMI-induced failures can occur at local strain levels well below 1% – particularly for brittle cladding at high burn-up (if the cladding exhibits a highly concentrated outer hydride rim) [25].

The cladding mechanical property degradation is one of the most important issues for high burn-up fuel utilisation, and the in-reactor performance of the

cladding strongly depends on the cladding material properties. The cladding material properties for high burn-up fuel should be carefully examined in order to reflect these data properly.

A range of power-increasing transients where PCMI may be important are addressed in plant and reload licensing safety analyses (e.g. loss of feedwater heating in a BWR and steamline break in a PWR – a safety level that may not require fuel rod integrity). If the PCMI stress is low enough or if the cladding ductility is high enough, PCMI will not be the mechanism for cladding failure. In those cases, the cladding temperature would rise because of the increasing power, and eventually critical heat flux might be exceeded and lead to cladding damage. For the latter transients CPR/DNBR fuel integrity criteria are generally limiting, and these transients are usually analysed from this perspective (i.e. without looking at PCMI).

Several things might occur at high burn-up that could result in early cladding failure by PCMI. First, the pellet-to-cladding gap reduction will eliminate some free expansion of the pellet prior to contact with the cladding. Second, the large accumulation of fission gas on fuel grain boundaries will also expand during a hold period after a power increase, which would contribute to the cladding strain, which is called the gaseous swelling phenomenon (the most important part of gaseous swelling is a delayed gaseous swelling contribution, the instantaneous gaseous swelling contribution is much lower). Third, cladding ductility is progressively reduced by radiation embrittlement already at intermediate exposure such that a mechanical failure might be considered (but actually such a failure mechanism is never activated because the pellet expansion is a strain driven loading and in such condition the cladding can withstand several percent strain without failing[11]). Fourth, if cladding hydriding further reduces the ductility of the cladding waterside at high exposure, mainly at lower cladding temperatures, PCMI failures could occur for those transients that were CPR/DNBR-limited before, and thus the critical heat flux type of analysis would then be inappropriate for safety evaluation.

Another failure mechanism is so-called outside-in cladding failure due to delayed hydride cracking, which has been observed on ramp tested high burn-up BWR fuel with Zr-liner in Japan. According to the test results obtained so far, fuel failure is dependent on local power level and hold time at the terminal power level [26]. This type of fuel failure mechanism must also be taken into account.

Experimental data on PCMI for light water reactor fuel cover a range of burn-up up to 60 GWd/t; so far, none of these results point towards PCMI effects being prohibitive at high burn-up. However, as these experiments usually aimed at investigating other high burn-up effects such as fission gas release, pellet cladding

11. At intermediate burn-ups, the only failure mechanism is SCC-PCI, which has very little to do with the cladding mechanical properties (except the fact that the pellet-cladding gap closure kinetics may play a role; if the low stress creep rate of the cladding is low, the gap closure intervenes at a higher burn-up, when the pellet rim and the bonding layer exhibit beneficial effects, impairing stress corrosion incipient cracks on the inner side of the cladding).

interaction resulting from stress corrosion cracking (SCC) and delayed hydride cracking (DHC), it appears warranted to perform more tests focusing on PCMI directly. For such tests, it is necessary to develop a special test plan in order to exclude the effects of SCC and DHC.

One way to exclude SCC failures from power ramp tests is to use non-irradiated fuel pellets in irradiated cladding, but such an approach would not be representative of a real fuel rod (pellet-cladding gap state would not be reproduced). On the other hand, to exclude DHC failures (for high burn-up BWR Zr-lined cladding), a power ramp test with short hold time may be effective. This is because the DHC process occurs at a relatively long hold times. However, at present there are no data for DHC under very high power conditions (over 50 kW/m), where the cladding stress and strain are primarily driven by PCMI (namely pellet thermal expansion). The DHC failures deal with specific types of cladding materials. Such an approach should not be extended to all cladding materials.

In summary, some concerns regarding the effect of high burn-up exist which should be addressed by performing more tests focusing on PCMI directly. Fuel design and performance codes may be used, provided they are well benchmarked, validated and verified against experimental data. Also, some more testing of PCMI for benchmarking these codes and verifying their results appears to be justified. [Note: with advanced claddings commercially available for PWR (like M5 or Opt ZIRLO), hydriding is no longer an issue. As a consequence, hydride assisted PCMI failure has become a non-issue. In addition, to try to fail such claddings under a strain driven loading is unlikely. The only incentive to perform additional prototypical PCMI tests is to provide relevant data to be able to relax the 1% strain limit.]

The thermal-mechanical limit (a burn-up dependent curve) is established while including the PCMI phenomenon, as well as various other phenomena (fuel rod internal pressure, stress/strain, fatigue, fuel melting, cladding corrosion and ballooning). In some countries, the limit is set to bound all these effects; also, the limit includes the effect of thermal and mechanical overpower during normal transients (AOO). Traditionally, this implies that conservatism is assumed to address uncertainties in various areas: models and model parameters (e.g. fission gas release), manufacturing tolerances and fuel /core management (e.g. as power histories during operation.) Thus, an overlay of conservatism exists with the margin to the real (nominal) limit not well quantified.

In modern fuel design methodologies the approach is different, namely – similar to the approach taken for establishing the CPR/DNBR safety limit – on a statistical basis [27] or [28]. The parameters in the areas mentioned above are treated as distributions, with quantified ranges of uncertainties, and a Monte Carlo model varies all these parameters for the multiple calculations of important design features (e.g. internal rod pressure). With a known design or safety limit, the necessary margin may then be identified.

To adequately cover modern fuel and core designs, the already mentioned best-estimate methods, along with associated uncertainty analysis, could be applied in order to reduce unnecessary conservatism. This implies, however, that

the fuel codes need to be well validated and verified; thus, experimental tests are to continue to provide the basis for such verification and validation. Additionally, code benchmarking is very valuable in order to evaluate the margin that should be considered in the safety analysis.

The basic safety criterion – the maintenance of fuel rod integrity – is not affected by new design changes, however the supporting strain limit (1% strain) may change.

For high burn-up cladding, the tendency to pick up hydrogen and the hydride morphology may be the most important issues in determining strain limit, and the presence and orientation of hydrides will depend on parameters such as cladding final heat treatment during fabrication (stress relieved versus recrystallised), burn-up level, and thermal-mechanical loads (inducing stresses), which usually are considered in fuel design (such as rod internal pressure criteria). However, attention must be given to the relevance of tube burst tests for the assessment of a strain limit [29]. These tests generally lead to conservative failure strains due to the plastic instability which occurs after the uniform elongation or ultimate tensile strength is exceeded.

3.10. Pellet cladding interaction (PCI)/stress corrosion cracking (SCC)

Some pellet cladding interaction (PCI) failures may be associated with stress corrosion cracking (SCC) in the cladding material, and are dependent upon local power ramps during reactor startup or manoeuvring (e.g. rod adjustments/swaps, load follow) and during normal transients (AOO). Both the stress from the power increase and the corrosion level (chemical component, e.g. iodine) at the pellet-cladding gap are necessary conditions for SCC-PCI. A crack, initiated at a microscopic defect in the inner side of the cladding, propagates until the stress in the remaining load-bearing part of the cladding exceeds the ultimate tensile strength, resulting in failure. Fresh fuel rods usually do not fail by SCC-PCI (because the gap is too widely open and consumes most of the power change until it closes and also because there are not enough corrosive component available). Neither do fuel rods operated at constant power fail from this phenomenon.

The SCC-PCI phenomenon has been extensively investigated after SCC-PCI failures were noted in operating reactors. To control the PCI phenomenon, operating rules (also called pre-conditioning interim operating management recommendations, or PCIOMRs) to limit local power increases and “condition” fuel to power ramping were implemented. These rules are usually a function of exposure (at intermediate burn-ups, when the gap is closed and the contact pressure is high, the fuel is less able to withstand ramping). In fact, the feedback from the ramp test database shows that SCC-PCI behaviour may differ between various fuel types depending on the pellet-cladding gap closure kinetics. It has been also observed that premature SCC-PCI failures can occur due to high local stress induced by pellets with “chips” or missing pellet surfaces (MPS) generated during the pellet manufacturing process. To establish and validate the operating rules, extensive power ramp tests were performed – by basically each fuel vendor – historically in the Studsvik and Petten test reactors, and now in the Osiris test reactor in France or the Halden test reactor in Norway; thus, the failure threshold

of the cladding is known very well up to 40-50 GWd/t and for power ramps well beyond normal operation. A certain amount of ramp testing has also been performed at higher burn-ups (up to 60-70 GWd/t), in particular in the frame of the OECD/NEA SCIP Project.

The PCI limits/rules typically contain a maximum ramp rate for reactor power increase (W/cm/hr), a maximum "single step" power increase (W/cm), and a threshold (in W/cm) above which such power increase limitations apply and a minimum time-period after which the fuel may be considered (pre)conditioned to larger power ramps.

In response to this issue, PCI resistant fuel types were developed. Notably, a small layer of zirconium ("barrier" or "liner", with or without small additives like Sn or Fe) was added at the inner part of the cladding as a remedy in BWR fuel designs. In addition, the modern fuel assemblies contain more fuel rods and therefore have a lower linear heat rating for each rod: this way, the fuel may permanently operate below the PCI threshold and thus not be in danger of failure from this mechanism. Another way of reducing the SCC/PCI risk of failure consists in filling the cladding with very short pellets, which reduce the "hour glassing effect" during power transients and thus the stresses levels applied to the cladding.

The PCI/SCC mechanism is sensitive to the gas composition in the gap but depends mostly on the availability of corrosive components generated by the fuel pellet during the transient itself. At high burn-up (typically above 45 GWd/t) the interface layer seems to become more and more protective [acting as an additional barrier to the corrosive species (iodine, cadmium and cesium) and as a media allowing local stresses redistribution. The electro-chemical behaviour of the layer may also play a role and needs further investigation]. There are test results indicating that high burn-up PWR fuels (around 60 GWd/t, local burn-up) are not likely to fail under the condition even exceeding failure threshold [19]. Since fission gas release from MOX fuel pellets will differ from that of UO_2 pellets, a MOX effect might also be possible but ramp testing feedback is showing that commercial MOX fuel is not susceptible to fail by SCC-PCI whatever the burn-up level. As a consequence no SCC-PCI limitation is applied in France for MOX fuel.

As for PCMI related safety criteria (Section 3.9), it might be interesting to continue to perform ramp tests in the high burn-up region in order to better understand the physical mechanisms that take place at the pellet-cladding interface. In view of the introduction of new cladding materials and/or new fuel types, a complete ramp testing programme, covering a wide range of burn-ups, has to be performed but the SCC-PCI limit will likely be derived from the ramp tests that will be carried on within the sensitive intermediate burn-up range (i.e. when pellet-cladding gap is closed and contact pressure is high).

During Condition II transient conditions PCI fuel failure mechanisms must also be taken into account. This failure mechanism may be significant in transients like the control rod withdrawal error, and in subcooled (cold water) transients.

For transients, there are several criteria for PCMI and SCC-PCI failure mechanisms, such as the DHC failure limit and the SCC failure limit. Other criteria, such as fuel melting (Section 3.11) and the 1% strain limits (Section 3.6), are related

to the PCMI mechanism. They do not cover each other and therefore should be evaluated considering power level and hold time at transient, fuel type and burn-up level for each transient.

In a number of countries, PCMI limits and SCC-PCI limits related to manufacturing-induced defects, such as MPS, are not licensed. However, operating limits (particularly the I-131 concentration level in the primary coolant) will bound plant operation.[12] In other countries, specific criteria are established to prevent PCI failures. This is particularly important where complex operation conditions, such as load follow or extended periods at low power level, are in effect.

Pellet-cladding interaction rules do pertain to safe fuel performance, and regulators will maintain that for non-PCI-resistant fuel these limits be adequate and that reactors obey these rules for core operation. The SCC-PCI limits should be kept updated, to be in concord with the respective fuel and core design envisaged; this is primarily done by performing ramp tests.

At present there is a good basis for SCC-PCI limits up to and beyond 50 GWd/t. A continuation of ramp testing is recommended to improve the basis at the higher burn-ups and as appropriate to the fuel design adopted. This is currently done in particular in the OECD/NEA SCIP-II Project. At the same time, fuel performance codes should be further developed and benchmarked against these ramp tests; finally, with sufficient modelling, the amount of testing could be reduced.

3.11. Fuel melting

Traditional practice in the design of light water reactor fuel has assumed that failure will occur if centreline melting takes place. This analysis is performed for the maximum linear heat generation rate throughout the core, including all hot spots and hot channel factors, and it normally accounts for the effects of burn-up and fuel composition (e.g. Pu or Gd content) on the melting point.

According to Section 4.2 of the NRC Standard Review Plan [30]:

> *Overheating of Fuel Pellets: It has also been traditional practice to assume that failure will occur if centreline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burn-up and composition on the melting point. For normal operation and anticipated operational occurrences, centreline melting is not permitted. For postulated accidents, the total number of rods that experience centreline melting should be assumed to fail for radiological dose calculation purposes. The centreline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the*

12. SCC-PCI failures following any class 2 transients are not considered here because such transients may affect the entire core and may lead to a large number of failed fuel rods, incompatible with the "operating limits".

> *cladding nor produce local hot spots. The assumption that centreline melting results in fuel failure is conservative.*[13]

For both normal operation and anticipated transients, centreline melting is not permitted. A reason for the criterion is that the transition from the solid to the liquid phase of UO_2 is accompanied by an increase (~13%) in volume.[14]

The centreline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. As pointed out in the NRC standard review plan, the assumption that centreline melting results in fuel failure is conservative.

Consistent with this discussion, regulatory guidance generally contains an explicit limit with regard to fuel melting during both normal operation and anticipated operational occurrences. Further, fuel melting may be used as a cladding failure criterion for some design basis accidents, and for these postulated accidents, the total number of rods that experience centreline melting is assumed to fail for radiological dose calculation.

For more aggressive conditions, present during some transients and accidents, this image is not representative. For example, during one hypothetical accident, the rod ejection event, NRC Regulatory Guide 1.77 [31] states:

> *For UO_2 fuel, a large fraction of this generated nuclear energy is stored momentarily in the fuel and then released to the rest of the system. If the fuel energy densities were high enough, there would exist the potential for prompt rupture of fuel pins and the consequent rapid heat transfer to the water from finely dispersed molten UO_2. Prompt fuel element rupture is defined herein as a rapid increase in internal fuel rod pressure due to extensive fuel melting, followed by rapid fragmentation and dispersal of fuel cladding into the coolant.*
>
> (Also see Section 3.14.)

It should be understood that fuel melting does not necessarily result in failure of the fuel or the fuel cladding. Evidence from Zircaloy-2 clad, vibratory compacted (83-85% T.D.) UO_2 fuel operated in the General Electric test reactor (GETR) to 20 000 MWd/tU at heat fluxes above 3.5 x 10^6 W/m^2 shows [32] that a UO_2 fuel rod can be successfully operated at a linear power sufficient to cause extensive melting.

13. After even low exposure, the radial power distribution in a fuel pellet peaks at the pellet surface rather than the pellet centreline. As a result, the maximum temperatures during a "rapid" transient may occur at locations other than the centreline. This is the opposite of normal operation or the "slow" power transient, where the maximum temperature is expected at the fuel centreline. To accommodate the rapid transient effects, the applicability of this paragraph in the NRC Standard Review Plan could be improved by substituting the word "fuel" for the word "centreline" in a number of places.
14. Page 2-49 of MATPRO (NUREG/CR-6150-Rev 2, Vol. 4) indicates that "during melting, an expansion equal to a linear strain of 0.043 occurs". This is equivalent to a volumetric expansion of 3 x 0.043 = 0.129.

Under these high power or even "slow" power transient conditions leading to massive melting of the central part of the pellet, the geometry of the fuel remains intact and, therefore, coolable. Assuming steady-state conditions, the centreline temperature in commercial light water reactor fuel reaches the melting point at power levels in the range of 82 kW/m (25 kW/ft). Linear power levels in operating reactors are generally much lower because they are limited by other, more restrictive conditions (e.g. thermal-hydraulic limits), but fuel melting continues to be avoided by design.

Another example related to "fast" power transient leads to the same conclusion: in the CABRI RIA test REP-Na2, the energy injection was high enough to melt part of the fuel pellet periphery but did not damage the cladding and the rod geometry remained intact.

The melting point of UO_2 is usually assumed to decrease with increasing burn-up. However, it must be noted that melting points were measured in Japan on high burn-up fuel (around 60 GWd/t) and the results suggest no clear reduction of melting points. Due to the difficulties in measuring melting point in high burn-up fuel samples, the results are not conclusive [22]. Further studies are necessary in order to accurately measure melting point in high burn-up samples.

A number of additives to UO_2 are being used or are being considered. These include burnable poisons (Gd_2O_3, ZrB_2) and materials to enhance either the mechanical compliance (SiO_2, Al_2O_3, Cr_2O_3) or the thermal conductivity (SiC, BeO) of the material. Only the effect of burnable poisons has been examined in any detail, and most of these efforts have focused on the changes in thermal conductivity rather than melting point.

3.12. LHGR limits

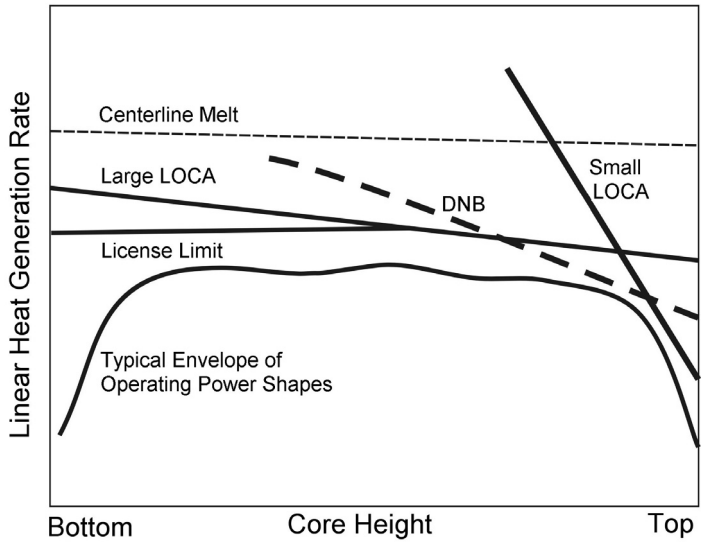
Linear heat generation rate (LHGR) limits are the most limiting of (1) thermal-hydraulic, (2) loss-of-coolant, and (3) thermal mechanical limits discussed in Sections 3.9 and 3.10. Figure 5 (taken from a plant final safety analysis report) schematically shows how these limits are combined.

This figure shows the limiting local linear heat generation rate for several different criteria as a function of core height. Some of these lines (centreline melt and departure from nucleate boiling) show a decrease with core elevation because the coolant temperature increases as a function of core height. Small break LOCA shows an even more pronounced decrease with core elevation because core uncovery for this event is usually limited to the top of the core. A typical envelope of operating power shapes is (must be) bounded by all of the criteria-based lines.

The next figure (Figure 6) is from ZIRAT15 Seminar (2011)[15] where it was adapted by Charles Patterson. It is a similar figure, but the maximum linear heat generation rate for various fuel safety criteria is plotted as a function of burn-up rather than core height. The figure schematically shows how these fuel (safety) limits are combined.

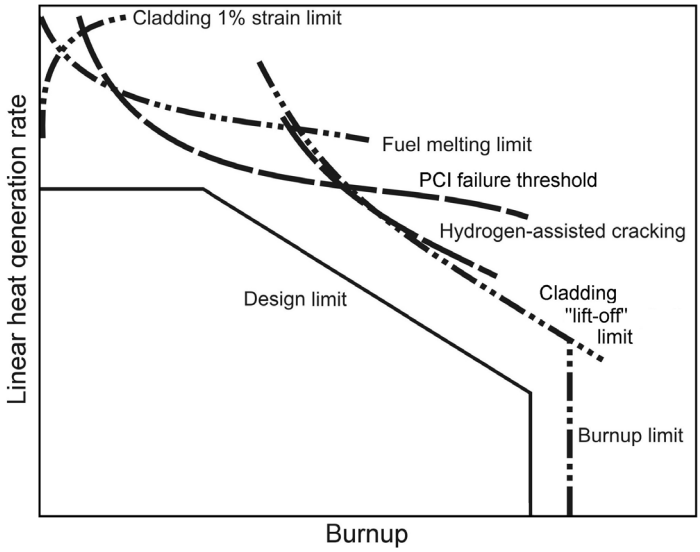
15. Advanced Nuclear Technology International, Sweden (www.antinternational.com)

Figure 5: Local LHGR limits as a function of core height



Source: USNRC.

Figure 6: LHGR as a function of burn-up for various fuel safety criteria



Source: ZIRAT15 Seminar (2011).

In this figure, the cladding 1% strain limit (Section 3.6) is lowest at beginning of life. Fuel melting and other fuel safety criteria show a modest decrease with burn-up. In deriving the LHGR limits, the most limiting of fundamental criterion (e.g. DNB) is plant, burn-up, and fuel design dependent.

3.13. RIA cladding failure

For a RIA, the number of fuel rod failures must be calculated so that the radiological doses to the public can be estimated.

In many countries, the cladding failure limit for RIA is based on the NRC standard review plan, which suggests a maximum radially averaged fuel enthalpy of 170 cal/g for BWRs and the DNB criterion for PWRs. Based on RIA experiments at CABRI and NSRR with fuel rods at a burn-up of approximately 50 GWd/t or higher, an assessment of the adequacy of this limit appeared desirable. In this respect, various limit values as function of burn-up have been proposed based either on direct experimental data (full-scale tests) renditions or on relevant parameters, such as cladding oxide thickness and mechanical properties.

Results for RIA tests (SPERT, PBF, CABRI, NSRR) with Zircaloy-cladding fuel above 5 GWd/t show failures from a pellet-cladding mechanical interaction (PCMI) rather than high-temperature failures related to critical heat flux [34]. It is believed that the reduction in gap size, reduced ductility due to irradiation and embrittlement due to hydrogen absorption are all responsible for the change in cladding failure mechanism. For irradiated fuel (in the case of UO_2-RIM or MOX clusters), fission gas induced fuel swelling is another contributing factor. Thus the effects of burn-up appear to enhance the loading to the cladding[16] and/or alter the failure mechanism and make the critical heat flux (as a stand-alone criterion) inappropriate.

In Japan, the threshold of PCMI failure in terms of enthalpy increase (cal/g UO_2) has been determined in terms of burn-up (GWd/t) based on NSRR, CABRI, SPERT and PBF data.

Table 4: PCMI failure threshold in Japan

Burn-up (GWd/t)	Enthalpy increase (cal/g UO_2)
<25	110
25-40	85
40-65	50
65-75	40

16. If the cladding ductility has not been altered by the in-reactor irradiation, then the RIA tests database shows that burn-up increase is not sufficient to alter the fuel rod failure limit (no critical burn-up threshold up to 70 GWd/t).

Since ductility will also be strongly affected by changes in cladding materials (e.g. the use of Nb as an alloying agent), the effects of new cladding materials on RIA fuel failure may also be important. For example, IGR tests [35] with Zr-1%Nb-clad VVER fuel show completely ductile behaviour without PCMI failure even at high burn-up (these tests were performed with a pulse width of about 700 ms); BIGR tests [36] show no PCMI failure up to 160 cal/g enthalpy at 48 GWd/t and up to 130 cal/g enthalpy at 60 GWd/t with very sharp pulses (pulse width < 3ms).[17]

Thus, especially in the higher burn-up range where experimental data were rare, technically-based PCMI fuel failure criteria and verification of the analytical models for fuel performance have been pursued [37]. The RIA Workshop in September 2009 [38] concluded that, despite various approaches, the international community converged towards similar PCMI failure limits. Beyond the phase where the PCMI failure mechanism takes place, especially for at power cases, the importance of the post-DNB condition has already been demonstrated in early PBF tests on fresh fuel rods, which resulted in high oxidation and embrittlement during film boiling, and cladding fracture and fuel powdering during rod quenching; therefore, further investigation for high burn-up fuel under realistic conditions appears warranted. This is the purpose of the CABRI International Project (CIP) and of the NSRR ALPS2 programme.

It must also be noted that, as identified in the recent OECD state-of-the-art report on RIA [37] and workshop on RIA [38], there is a need for further experimental investigations for a range of conditions, such as fuel with medium burn-up, MOX failure limits or RIA transients initiated from non-zero power, that were insufficiently or not at all studied in the past. The new results generated will have to be taken into account in the RIA criteria.

3.14. Fuel fragmentation and fuel dispersal

In this chapter, fuel fragmentation refers to situations for which the fuel cladding breaks into pieces and fuel dispersal to situations for which fuel particles escape from the cladding following a rupture.

Although fuel fragmentation is traditionally considered to exist only in conjunction with highly energetic events such as the reactivity-initiated accidents (RIA), recent results from the Halden test reactor show that fuel fragmentation can also occur during the loss-of-coolant accident (LOCA).

To avoid the loss of coolable geometry and the generation of coolant pressure pulses, peak fuel enthalpy criteria are used as limits for RIA. Historically, a radially-averaged fuel enthalpy value of 280 cal/g has been used in the United States and other countries based on data from early RIA fragmentation measurements prior to 1974 on fuel with a maximum burn-up of 33 GWd/t (e.g. SPERT and TREAT tests in the United States); this value corresponds to the melting of UO_2 which causes

17. In fact, due to massive DNB during a few seconds, the pulse width is no longer a governing parameter for the result of the test. Hence the similar results for both IGR and BIGR tests.

fragmentation of the cladding and expulsion of fuel particles. The expulsion of molten fuel also led to energetic fuel-coolant interactions that generated pressure pulses. Later refinements in the measurements and in the definition of the fragmentation enthalpy value, as well as PBF-RIA tests led to reductions of the 280 cal/g limit. Accordingly, various regulatory authorities use a lower value for the enthalpy limit.

The original SPERT and TREAT data indicated that the 280 cal/g total energy deposition was conservative to ensure minimal core damage and to maintain core coolability. This limit was subsequently incorporated into NRC Regulatory Guide 1.77 [31]. However, some of the tests also indicated that a fuel rod subject to a radial average peak fuel enthalpy of 280 cal/g will be severely damaged, loose its original geometry and impair post-accident cooling; on this basis a revised criterion of 230 cal/g was recommended [39] but not incorporated into the NRC regulatory guide although it has been adopted in most European countries. At the same time, the question whether the limit should be identical for unirradiated and irradiated fuel was brought up (ref. results from the PBF RIA 1-1 test). In Europe, around 220 cal/g is used for fresh fuel and 200 cal/g is used for irradiated fuel. An international industry working party, led by EPRI, suggested a value of around 240 cal/g for fresh and low burn-up fuel.

Experiments in the French CABRI test reactor [40] and the Japanese NSRR test reactor [41] using high burn-up fuel samples have resulted in fuel particle dispersal for deposited energies well below 200 cal/g. It is clear that, for high burn-up fuel, a mechanism other than fuel melting is producing particle dispersal at low deposited energies. This new mechanism may possibly be related to the large accumulation of fission gas bubbles on grain boundaries of the fuel and the rapid expansion of that gas during the power pulses, with special emphasis on MOX clusters. Entrainment of particles in escaping fission gas may also be involved. Various effects (pulse width, cladding type, coolant type, internal pressure, coolant temperature), some of which are not yet well understood, may play a role: however the burn-up effect in UO_2 is evident. For RIA, the current practice is often to define criteria intended to ensure that there is no fuel cladding failure, thus preventing from fuel dispersal and fuel fragmentation.

Of special interest is the formation of a peripheral zone in the fuel material with high plutonium content and consequently high reactivity, porous structure and high content of fission products, the so-called RIM, which grows as a function of exposure: at about 45 GWd/t the RIM-zone is of the order of 60 µm. Studies are continuing to clarify what role the RIM-zone plays in transient accident situations and when grain fragmentation occurs during the RIA transient: during the pulse or during the cooling phase when the pellet becomes less constrained by the cladding. This point is crucial: if the grain fragmentation occurs during the pulse, all the fission gases that were trapped in the RIM are suddenly released and may add a pressure considered in PCMI loading.

The role of the RIM-zone has also been questioned in the loss-of-coolant accident. In April 2006, a LOCA test, IFA-650.4 was run in the Halden reactor on a fuel rod segment with a very high local burn-up of 91.5 GWd/t. Results from this test showed gross loss of fuel material from above the rupture opening [42]. Online instrumentation indicated that this fuel loss occurred during the temperature

transient rather than after the test was over. In this very high burn-up fuel specimen, more than 40% of the fuel material was RIM-zone material. This material was able to flow freely under the influence of gravity and pressure differences within the cladding tube and through the rupture opening. This result was later replicated with a sibling rod in Halden LOCA test, IFA-650.9. Fuel fragmentation mechanisms need additional investigation but it appears that the pellet burn-up (and the related level of pellet restructuration) aggravates the effect. A lower burn-up rod (IFA-650.10) tested under similar conditions, exhibited lower fuel fragment relocation and dispersal.

For reactivity events, it may be correct that the fuel dispersal limit is in the range of around 230 cal/g. This may be sufficient to ensure a coolable geometry for fresh and very low burn-up fuel during this energetic event. However, for both LOCA and fuel at high burn-ups, there is a need for further understanding of the fuel dispersal process and the effects of high burn-up (in particular the effect of the RIM-zone and the MOX clusters).

3.15. Non-LOCA cladding embrittlement/temperature

Certain non-LOCA accidents are analysed to estimate radiological doses to the public and to demonstrate that coolability of the core is maintained.

For accidents like the PWR locked rotor accident, DNB is used to indicate cladding failure for dose calculations, and the peak cladding temperature of 1 480°C (2 700°F) is sometimes used to demonstrate coolability. The 1480°C limit was taken from early data estimates of the fuel failure boundary for LOCA conditions (1 480°C and 17% of cladding thickness oxidised by metal-water reaction). This limit was established in the 1969-1971 time period prior to the ECCS hearings in the United States, which resulted in a lower temperature limit for LOCA analysis (2 200°F or 1 204°C). The rationale for retaining a higher temperature limit for non-LOCA transients was that those transients were of brief duration and fuel rods could withstand brief periods of DNB without suffering serious damage. The 1 482°C limit corresponds to one fuel vendor's opinion of the maximum cladding temperature beyond which the oxidation is thermally self sustained (due to endothermic reaction). Other (different) limits were also proposed.

This peak cladding temperature criterion is a measure of the amount of oxidation that can take place during the transient and the related loss of ductility. Because oxidation and hydrogen absorption also take place during normal operation, this will cause a further reduction of ductility at high burn-up. Therefore, the peak cladding temperature during the transients may have to be adjusted to accommodate normal corrosion. Since cladding ductility is also affected by cladding materials (e.g. the use of Nb as an alloying agent), an effect of cladding materials would also be expected for this criterion. However, the high temperature oxidation kinetic, which is the leading mechanism susceptible to altering the cladding residual ductility, does not strongly depend on the nature of

the cladding. Results from the KIT[18] QUENCH experimental programme show this dependence of the oxidation kinetic on cladding material for various oxidation temperatures ranging from 600°C to 1 600°C [43]. The behaviour of highly burnt fuel under this condition is relatively unknown. The relevance of the above criterion should therefore be confirmed experimentally.

3.16. LOCA cladding embrittlement

For LOCA analysis, it is generally assumed that a certain amount of fuel rods fail and release fission products, but that emergency core cooling systems (ECCS) operate in such a way that fuel rod fragmentation is avoided, thus preserving a coolable geometry, and moreover provide long-term core cooling.

Based on many laboratory quenching and ductility tests (or strength-based tests) with unirradiated Zircaloy tubes, it was found that cladding would not become embrittled enough to fragment if the peak cladding temperature remained below 1 204°C (2 200°F) and the total oxidation did not exceed 17% of the cladding thickness prior to transient based on the Baker-Just oxidation correlation. These cladding embrittlement criteria (ref. 10CFR50.46) are used widely, although in some cases the oxidation limit is placed at 15% (e.g. Japan).

In addition there is a LOCA limit on core-wide hydrogen generation, however this is for containment integrity and not against embrittlement (limit is usually 1% related to total possible cladding oxidation).

The cladding embrittlement criteria were developed in the 1960s and early 1970s; experimental verification and validation included tests with zero or low burn-up fuel. Nowadays fuel operation exhibits typical oxidation levels of up to 100 microns or more and hydrogen concentrations up to or even higher than 600 ppm at the time of discharge (these levels are usually employed as criteria for fuel mechanical design, which frequently become licensed limits). Hence the 17% criterion is now often interpreted as “total” oxidation level. As the oxidation process at LOCA temperatures differs from that at normal operating temperatures, this interpretation may be considered as being very conservative; the question whether the oxidation during normal operation should be accounted for when comparing against the 17% LOCA-limit is unsettled. In Europe, the total oxidation must be considered. In the United States, consideration of pre-transient oxidation is suggested but not required. A different criterion, that might be more suitable especially at high burn-up, could also be envisaged.

There is a number of issues and concerns that necessitate additional verification and subsequent justification or adjustment of the current LOCA limits.

Some of these are related to high burn-up:

- radiological consequences: extent of rods burst, and fission product release;

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- consequence from hydrogen-induced β-phase stabilising effect on cladding strain;
- consequences from fine fragmentation of the fuel, filling the space in the ballooned cladding (also the timing of slumping of the fuel column into the ballooned area is important, as well as the consequences of additional decay heat related to cladding temperature and oxidation);
- cladding behaviour during quench after high temperature oxidation, and long-term cooling: changes in oxidation rate and cladding embrittlement, heating and cooling rates;
- UO_2-RIM or MOX clusters fission gas induced fuel swelling (losses in mechanical strength may become important if cladding wall strength is significantly weakened during irradiation);
- modelling accuracy, e.g. cladding thickness calculation after burst or adequacy of Baker-Just oxidation correlation.

Whereas some are of a more generic nature:

- fuel relocation in the ballooned region and its impact on the calculated peak cladding temperature (PCT);
- potential subchannel blockage (interaction between ballooned areas, axial extent of the ballooned areas).

Many NEA member countries have been examining the LOCA requirements, which impose conditions governing LOCA analysis [44]. For future verification and review of LOCA safety criteria, the results from this examination should be taken into account.

3.17. Blowdown/seismic/transportation loads

During a seismic event the fuel assemblies are subjected to dynamic, structural loads which could cause fuel assemblies to sway back and forth, causing impacts with each other and with the vessel wall. Jet forces associated with blowdown from one side of the vessel through a broken pipe could also accelerate the vessel in the lateral direction, resulting in similar impacts between fuel assemblies and vessel wall.

Analyses usually include the consideration of mechanical and hydraulic loads in horizontal and vertical directions; critical crushing loads are used to determine if such impacts cause grid deformation that reduce coolant flow and degrade ECCS performance. Other mechanical properties are used to ensure that fuel rods do not fragment, thereby loosing coolable geometry, and that guide tubes and channel boxes do not fracture and prevent control rods from being inserted.

Most countries follow the safety criteria as per NUREG-0800, SRP 4.2, Appendix A which require core coolability and control rod insertability to be assured under the combined seismic and LOCA loads. These criteria are often translated into design requirements such as:

a) fuel rod fragmentation shall not occur (can be met by verification that fuel rod stresses are within limits);

b) control rod insertion shall not be impaired (verify that combined loads do not displace the fuel assembly from the support piece);

c) limit spacer distortion to ensure rod coolability (verify that spacer distortion or failures either do not occur or do not decrease the hydraulic section of the grid cells).

Verification is performed analytically to compare design loadings with fuel assemblies performance. Fuel assembly capacities are experimentally determined by using various specific tests (fuel assembly drop tests, fuel assembly bend tests, grid buckling tests, etc.). As fuel designs may have different dynamic properties, this analysis is not only fuel design dependent but also core design dependent; in particular, the mixed core situation should be addressed explicitly. (Actually, cores are never homogeneous as the actual fuel assemblies dynamic properties are continuously changing during their irradiation from beginning to end of life.)

Also, design requirement changes on allowable structural loads for earthquakes during and after a LOCA may be needed at high burn-up, because the strength and ductility of high burn-up cladding, guide tubes, grid spacers (PWRs), and channel boxes (BWRs) will not be the same as for fresh material. Analyses for fresh fuel usually show ample margins, and the increased strength at high burn-up would seem to enlarge those margins. But the method of review presumes that the material being analysed is ductile, whereas a substantial loss in ductility occurs at high burn-up for some materials. It should be noted that the alteration at high burn-up of the buckling strength of the spacer grids is not related to the properties changes of the grid materials but to the grid-to-rod spring relaxation under irradiation. By intentionally relaxing the cell springs of an as-received spacer grid, it is then possible to define a test protocol that allow to simulate the effect of irradiation on the spacer grid buckling resistance.

Nevertheless, altered materials properties (growth, creep, ultimate stress and strain, etc.) for high burn-up cores and for new core materials may well affect the results of this structural analysis; thus, adequate treatment of these properties is needed, which implies that material properties verification at high burn-up is of importance.

Safety criteria in this area are not directly affected by the new design changes. Considering the fact that compliance with criteria is demonstrated analytically, methods used to analyse the seismic/LOCA event should be well verified and validated.

3.18. Assembly holddown force

LWR fuel assemblies are equipped with holddown springs in the top piece. They have to provide sufficient forces to prevent fuel assembly lift-off due to hydraulic loads during normal operation and anticipated operational occurrences, with the exception of the hot pump overspeed transient (for the hot pump

overspeed transient some lifting is tolerated; the holddown springs shall again prevent fuel assembly lift-off after the transient has subsided).

Safety criteria are usually defined following NUREG-0800, SRP 4.2, App. A: vertical lift-off forces must not unseat the lower fuel assembly tieplate from the fuel support structure.

The required holddown force is calculated by:

$$F_{HD} = F_{HY} + B - W$$

where

F_{HD}	required holddown force
F_{HY}	hydraulic force
B	buoyancy force
W	fuel assembly weight.

The hydraulic force on the fuel assembly depends on the coolant flow rate and the fuel assembly pressure loss coefficient. A conservatively high flow rate (mechanical design flow rate) is used for calculating the required holddown force. The uncertainties and tolerances are taken into account differently by the fuel vendors, but in general the following uncertainties and tolerances are taken into account:

- tolerance of axial spaces between lower and upper reactor core plate;
- tolerance of assembly length;
- tolerance of fuel assembly weight;
- uncertainty of the coolant flow rate;
- uncertainty of pressure loss coefficients;
- uncertainty of holddown spring deflection curve and spring constant;
- uncertainty of guide tube axial growth;
- uncertainty of spring relaxation.

The analytical evaluation of required holddown force is done for cold startup conditions and hot full power conditions, during the whole life of the fuel assemblies within the core.

The fuel assembly holddown force leads to compressive forces on the guide tubes, which forces can give high fuel assembly bow due to irradiation induced guide tube creep. Vice versa, high compressive forces can result from excessive guide tube growth.

Guide tube growth is correlated to the fast neutron fluence and hydrogen pickup. Therefore a major consideration at high burn-up levels, where high corrosion and hydrogen pickup of the guide tube accelerate guide tube growth above the fast neutron irradiation induced rate. Corrosion and hydrogen pickup highly depend on the coolant temperature and on the guide tube material and its

condition. Thus, to ensure acceptable guide tube corrosion and hydrogen pickup, guide tube design and material has to be selected adequately.

3.19. Fretting wear

Fretting wear at contact points on the fuel assembly structural members should be limited.

Fretting wear tests and analyses that demonstrate compliance with this design basis should account for grid spacer spring relaxation. The allowable fretting wear should be stated in the safety analysis report, and the stress strain, and fatigue limits should presume the existence of this wear.

It is noted that wear cannot be analytically predicted at this time. The fretting wear phenomenon depends on fuel design of course (and experimental relative assessment can be successfully done to identify and select the most robust designs) but in reactor grid-to-rod fretting wear will depend on various parameters, including the cross flows signature, which is plant dependant and difficult to quantify.

As a consequence, compliance with the fretting wear limit (typically 10% of the cladding thickness) is checked *a posteriori*, through post-irradiation examination.

For WWERs, a first fretting wear design criterion requires that no fretting (due to rapid movement such as vibration e.g. in lower or upper tie plates) shall occur after minimum of 3 000 hours endurance testing [45]. A second design criterion for avoiding fretting wear limits the cladding reduction (due to creepdown) to 0.10 mm; this criterion is in place due to the different spacer grid design of WWER fuel, which does not include any springs — the contact between grid and fuel rods is controlled only by the grid construction and must be warranted also after cladding creepdown.

Spacer grid structural tests, control rod structural and performance tests, fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping), fuel assembly hydraulic flow tests and endurance tests (lift forces, control rod wear, vibration, fuel rod fretting) are the necessary tests to determine if a specific fuel assembly design is sensitive (or not) to the fretting wear phenomena. To simulate the effect of irradiation those tests should account for spacer spring relaxation.

The Electric Power Research Institute Fuel Reliability Programme includes guidelines on grid-to-assembly fretting wear [46].

3.20. Coolant activity

In most countries, limits are specified in the plant technical specifications on the concentration of I-131 (sometimes also of Cs-137) in the primary coolant; numbers are typically around $2 * 10^9$ Bq/t. Thus NPP operation with a certain (small) number of fuel failures is tolerated; the plant systems have been designed to cope with fuel failures of this magnitude. Aside from this technical specification limitation, no fuel safety criteria on coolant activity exist.

Usually, as soon as larger I-131 concentrations are measured that would challenge the technical specification limit, the plant operational staff prepares for plant shutdown to identify and replace the leaking bundles, so that plant operation within technical specification limits may be continued.

From large cracks in the fuel rods (direct contact between fuel and coolant) washout of fuel material from the pellet may occur, subsequently leading to a high concentration of Neptunium. Even after the leaking fuel has been removed, it may take a long time (several years/cycles) for this concentration to decrease as fuel material is plated out throughout the primary system; this implies that small amounts of washout from later fuel failures cannot be observed from the Np-concentration due to the large background already available. Thus, a pure correlation between actual Np-concentration and fuel failures does not appear to be possible; nevertheless, most fuel vendors analytically associate Np-concentration with the size and number of fuel leaks from. A future technical specification limit on the Np-concentration may be needed to avoid operation with large fuel failures with substantial amounts of washout.

No change of the above limit(s) is expected in conjunction with new design changes.

3.21. Fuel gap activity

Fuel gap activity is of interest for accident scenarios that may result in cladding failures but that do not involve melting of the fuel. It determines the potential release of fission products to the primary circuit.

During normal reactor operation, some fission products come out of the UO_2 fuel matrix and collect in the gap between the fuel pellet and the cladding. Fixed values of release to the gap, like up to 10% of the rod inventory for noble gases and 1-6% for halogens and alkali metals, are assumed in safety analyses. These gap activities are then assumed to be released from failed fuel rods for the purpose of off-site dose calculations for postulated accidents. The release fractions assumed are not safety criteria, but represent conservative numbers used for assessment of the design.

Fission gases are not very soluble in the UO_2 matrix, and most of these fission products take up residence in the form of gas bubbles that become attached to grain boundaries. At very high burn-up the grain size becomes smaller in the RIM zone, and also leads to the formation of high number of micro-sized pores in which the fission gas is supposed to be contained; yet, these pores are not interconnected and do not significantly contribute to the gap inventory since the gap is closed. However, gas bubbles become interlinked along the boundaries in the fuel centre, providing easier pathways for release to the gap. Volatile fission products such as iodine and cesium partly evaporate from the grain surfaces to the interlinked porosity and are eventually released to the gap. Hence fission product release to the gap is found to increase at high burn-up; a similar enhancement compared with UO_2 is seen for MOX fuel. These increases in release may require the modification of assumptions about gap activity that are used in safety analyses.

These increases in release may require the modification of assumptions about gap activity that are used in safety analyses. The separate effect tests which have been performed so far on highly irradiated UO_2 and MOX fuel pellets (up to 70 and 65 GWd/t, respectively) are showing that there are still significant margins between the actual gap activity and the gap activity used in the safety analyses [47].

3.22. Source term

During and immediately following an accident, the part of the fission products inventory, released into the containment, potentially available for release to the environment is called the source term. In most countries, a severe-accident source term (associated with core melting) is defined deterministically to estimate radiological releases to the public for beyond design basis accidents. Source terms are also used in probabilistic risk assessments to estimate plant releases and accident consequences.

Source terms are based on measured releases from irradiated fuel, tested under accident conditions, in combination with assumptions or analyses of the effects of retention or enhancement during the course of an accident sequence.

Source terms related to design basis accidents are calculated regularly, in conjunction with safety analyses for licensing of new fuel designs/core loading strategies, to evaluate radiological consequences. Thus, changes due to new design changes are accounted for, which may lead to changes in source term levels. However, the assumptions or analytical procedures themselves are not expected to change.

There are no safety criteria directly associated with source terms. Various assumptions are made for the analysis of accident scenarios and for retention effects, etc.; in part these are rooted in the basic reactor design philosophy, and can vary significantly between various regulatory frameworks. Although these differences are known, attempts to unite the analytical procedures and assumptions have not been very successful thus far.

The effect on source terms from new design changes – especially from high fuel burn-up – is estimated as follows.

The main effects that could impact source terms as well as core melt progression at high burn-up are (a) a modification in the amount of unoxidised zirconium in the core, (b) embrittlement of the fuel cladding, (c) an increase in the release of fission gases from fuel pellets during normal operation, (d) fragmentation of fuel pellets, and (e) a shift in the spectrum of fission products produced as plutonium fission becomes more important. Effects (c), (d), and (e) could, in principle, also impact source terms for MOX fuel.

The amount of unoxidised zirconium may be lower because higher burn-ups are associated to longer residence time of the fuel in the core, or higher because new cladding materials used have a much lower propensity to oxidise during normal operation. However, the amount of preoxidation of the cladding will be less than 15-17% of the wall thickness because of regulatory limits related to LOCA, and is likely to be much lower than that for newer cladding alloys; therefore, this

beneficial effect would be small. Non-molten fuel relocation may occur due to cladding embrittlement, particularly for scenarios involving delayed reflood or depressurisation, but this is not expected to significantly affect the overall outcome of uninterrupted core melt accidents. Gap activity comprises only a small part of the source term so that even large changes in gap activity would not have a big effect on the source term. Fuel fragmentation has been observed at high burn-up, but it appears that dispersal of fragments occurs by washout or gas entrainment and there may be no means to get that material into the atmosphere as aerosol particles. In contrast, particulate releases included in the source term are lifted from the core as high temperature gases that condense as aerosol particles. The source term itself is defined by release fractions and therefore would not be affected by isotopic shifts. Those shifts would be accounted for in the generation analysis (e.g. with an analysis code such as ORIGEN): In any case changes are expected to be small, and experimental programmes are being undertaken to provide an adequate validation basis for these analysis codes at higher burn-ups.

Considering the above factors, it is unlikely that high burn-up will have a significant effect on source terms or core melt progression. A similar statement can be made about MOX fuel: indeed, with respect to fission product release, the governing parameter is the local burn-up. Thus, because of the heterogeneity in burn-ups due to the initial presence of plutonium, MOX fuel may be seen as equivalent to a UO_2 fuel with higher burn-up. Also, the implementation of a revised source term [48] could affect the dependence on new designs and materials.

It is considered unlikely that new design changes or high burn-up will have a significant effect on source terms or core melt progression.

3.23. Burn-up

First, a short summary of the situation regarding the high burn-up issue (licensed burn-up limits, burn-up levels achieved today and expected burn-up extensions) is given. Licensed burn-up limits depend on the type of fuel and fuel vendor; licensed limits may refer to local (sometimes referred to as “peak pellet”) burn-up levels and/or rod average burn-up levels and/or assembly average burn-up levels. Examples of licensed burn-up limits are as follows:

- a maximum rod-average burn-up of 62 GWd/t for some fuels in the United States;
- a generic limit for maximum fuel assembly average burn-up of 52 GWd/t exists in France for UO_2 and MOX fuel. MOX fuel was previously limited to 47 GWd/tM and 3 one-year cycles of insertion;[19]

19. Since 2007, the “MOX Parity” fuel management is now deployed in 21 French 900 MWe NPPs and allows 4 one-year cycles of insertion, and the same assembly discharge burn-up for UO_2 and MOX fuel (enrichment of UO_2 is 3.7% U-235 and the average Pu content of the MOX assemblies is equivalent to 3.7% U-235).

- the need for very high burn-up limits is somewhat alleviated in France by the use of fuel recycle technology. High burn-ups degrade the isotopic composition and "quality" of reprocessed fuels (MOX and enriched reprocessed uranium);[20]
- maximum assembly average burn-up is 57 GWd/t for the fuel type used in operating VVERs and 50 GWd/t for the fuel types used in operating BWRs in Finland;
- generic limits for the maximum assembly average burn-up of 55 GWd/tU for UO_2 in PWRs and BWRs, 45 GWd/tHM for MOX in PWRs and 40 GWd/tHM for MOX in BWRs, respectively in Japan;
- maximum assembly average burn-up of 65 GWd/t for PWRs and 53 GWd/t for BWRs, for some fuels in Germany;
- maximum assembly average burn-up of 50 to 70 GWd/t, or maximum local burn-ups of 59 to 82 GWd/t for various different fuels in Switzerland.

Previously, the high burn-up issue and its possible consequences have been brought up many times. In some countries, the industry greatly focuses on high burn-up, which is claimed to be the biggest key to better fuel economy; therefore, this issue continues to receive a lot of attention, especially with respect to transient/accident behaviour [49-50].

In the EPRI report "Licensing Criteria for Fuel Burn-up Extension Beyond 62 GWd/tU" [51], the investigators aimed at providing an industry-wide consistent approach to the licensing of burn-up extensions for LWR fuel and at simplifying the burden of demonstrating adequate performance at targeted burn-up limits. In their approach, the investigators established a systematic process to review and assess the applicability of existing design bases above the current burn-up limits. The review process relied on design approaches specified in licensee fuel design topical reports as well as guidelines outlined in the NRC's standard review plan (SRP), Section 4.2 to identify important fuel response behaviours that require design limits. To ensure a sufficiently broad spectrum and to capture all important burn-up effects that may influence fuel rod or assembly behaviour, the review also included the fuel-related general design criteria specified in 10CFR50 Appendix A.

This report identifies burn-up-dependent design criteria as well as key design parameters or performance measures used to demonstrate compliance with the criteria. It takes into account an evaluation of whether current design limits need to be modified to account for burn-up or whether the effects of burn-up could be addressed through an expansion of current methods and performance data. Research showed that most current design limits could be retained when supported with data at the targeted burn-up levels. Criteria relating to the

20. A limit of 62 GWd/t has been licensed for the "GALICE" fuel management in France in the 1 300 MWe NPPs, which has been implemented in only one NPP, to gather in-reactor feedback. There is no plan to extend this type of high burn-up fuel management to other 1 300 MWe NPPs.

response of fuel to reactivity initiated accidents (RIAs) or to loss-of-coolant accidents (LOCAs) require a more complex evaluation.

In recent years more information has become available on the behaviour of highly burnt fuel. This has provided additional basis for the fuel/core operation for burn-up level up to those currently licensed. However, the working group also considers that there is a need for further research to (a) experimentally verify the validity of safety criteria for high burn-up, in particular for burn-up levels beyond those currently licensed, and (b) further develop and benchmark the analytical models used in the safety design studies to comply with the high burn-up safety criteria.

Clearly, one of the main benefits of high burn-up has been to decrease the fuel cycle cost. Another benefit has been the increased operational flexibility that high burn-ups allow. However, the OECD/NEA Nuclear Science Committee has noted that the question for which there is no definitive answer at present is whether this historic trend will continue indefinitely or whether there will be a technological limit to PWR burn-ups [52]. The desire for higher burn-ups has been tempered in recent years in the United States as the migration to longer cycle lengths often results in the discharge of some fuel assemblies at lower burn-up. That is, longer fuel cycle length reduces fuel utilisation flexibility. The question is very important for utilities and fuel fabricators, as it is probably the single most important technical unknown affecting the future LWR fuel cycle. The Nuclear Science Committee has also examined the worldwide issue of research facilities available to address this question [53].

In addition, it is to be noted that the economic balance is more complicated to establish where fuel reprocessing is performed. Indeed, when the burn-up increases, reprocessing may become more and more difficult. Because of this, the French utility EDF has chosen to suspend its plans for increasing burn-up beyond values currently authorised.

As pointed out throughout this report, it is important that all aspects related to high burn-up are now covered (fuel and core design, choice of materials, goodness of analytical methods). In this respect the fuel vendors will bear most of the responsibility, for basic qualification of their respective fuels; independent verification by the utilities (probably as a joint effort, via internationally sponsored research and development programmes at national or international research centres) will have to be added, while selecting the experimental test cases appropriately and carefully. The idea of independent verification of design changes/new fuel designs/generic issues by the utilities has been adopted in the United States through the use of an industry review team. Such review teams are formed on a need basis and managed by EPRI.

4. Other considerations

As part of the assessment of the fuel criteria, the Working Group on Fuel Safety looked at various other issues and considerations, as they relate to one or more fuel criteria, that have become of special interest. These topics are separately discussed below.

4.1. Core management

The fuel cycle costs are an important part of the costs for plant operation. Utility strategies to reduce costs have increased the activity in the core management area; as a result of optimised core management, such as higher fuel discharge exposure, the loading strategies have changed.

In the past, the loading strategy included the loading of fresh fuel into the centre of the core and then, as a function of exposure, to move the fuel towards the edge of the core with each reload ("low leakage" loading pattern, or "in-out-out"). For this type of loading strategy the LHGR power history curves showed a monotonous decrease against fuel burn-up.

Modern loading strategies use a smaller number of fresh reload bundles on account of the higher fuel discharge exposures. This leads to higher power peaking due to higher reactivity of those bundles. Safety criteria, notably LHGR, SDM and DNB/CPR, must however still be met; as a consequence, fuel bundles with very high burn-up may now have to be loaded into a centre of the core adjacent to fresh fuel bundles. This implies that reaching maximum fuel burn-up levels is no longer limited to those bundles at the core periphery. Also other modern core management features such as the control cell core cycle design for BWRs (movement of only a few selected rods for reactivity control during the cycle) lead to having fuel with high burn-up in the core centre.

This situation may influence the behaviour of the high burn-up fuel during transients/accidents. As an example, during a small/medium size LOCA the cladding of the fresh fuel may collapse due to low cladding internal pressure[1] and the high burn-up fuel may balloon due to high fission gas release during normal operation prior to the transient and during the transient itself. In the collapsed cladding case strong mechanical interaction between the fuel pellet and the cladding dominates internal oxidation of the cladding, and together with diffusion of the pellet material and fission gases into the cladding will cause fuel to fail; in

1. This will not happen for a large LOCA, because of the rapid decrease of coolant pressure: internal overpressure occurs even for fresh fuel.

the high burn-up fuel cladding ballooning, burst and double side steam oxidation are dominating mechanisms. Also during quenching and cooling down of the fuel the failure mechanisms are different between high burn-up and fresh fuel bundles. The effect of the fuel failure mechanisms for high burn-up fuel may be enhanced by the larger reactivity (power) level in the adjacent fresh fuel; in return, the effects in highly burnt fuel could adversely affect the failure mechanisms in fresh fuel. Also, the different behaviour of fresh and high burn-up fuel bundles has an impact of flow redistribution during the accident, which can challenge the fuel coolability criterion.

The above example may serve to illustrate the importance of having good physics models that are adequately validated[2]/benchmarked.

Traditionally the codes used in transient and accident analysis are one-dimensional. Currently state-of-the-art modelling includes the use of three-dimensional neutron kinetics codes, though the thermal-hydraulic modelling remains one-dimensional. With reactor cores becoming more heterogeneous, the capability of present codes for analysing the transient behaviour of high burn-up and fresh fuel bundles should be verified. It is equally important to further develop codes, which can analyse complicated thermal-hydraulic phenomena between adjacent fuel bundles such as flow blockage induced cross-flows between collapsed and ballooned fuel. It is important that this validation includes experimental test data on these phenomena.

In summary, changes in core management do not directly upset safety limits or margins; as long as satisfactory modelling is available to describe the phenomena occurring in currently designed and operated cores, safety limits are not affected.

4.2. Mixed-oxide fuel

Some countries have chosen the option of reprocessing spent fuel. Thus, contracts with reprocessing companies were put in place resulting in a certain quantity of fissile Pu, which can be used together with UO_2 to manufacture so-called mixed-oxide fuel (MOX), as well as some amount of reprocessed uranium, which may be used as carrier material or blended with regular UO_2. Also, the option of using weapons grade high enriched uranium and plutonium has been considered.

Thus, MOX insertion is taking place (to date mainly in Europe) or is being planned in a number of countries, and is therefore of concern with respect to safety criteria. Various designs were and are being considered; presently the “all-MOX” type of design with the largest possible amount of Pu in the smallest possible number of assemblies appears economically to be the most attractive (with burnable absorber still blended with UO_2 only). In general, the performance

2. In standard terminology for verification, validation and uncertainties quantification, “verification” stands for numerical verification and “validation” stands for physics validation.

of MOX fuel is less characterised than for UO_2 fuel, especially at high burn-up. Experiments will continue to be needed to confirm the operational regimes in which MOX fuels are compliant with safety criteria, considering also that MOX performance can be affected by the fabrication route and by the total plutonium content in the fuel.

Safety related effects of MOX (as compared to standard UO_2) insertion may be summarised as follows:

- In general, a lower boron and control rod worth is to be expected due to the different isotopic and spectral characteristics.
- For the same reason, a more negative Doppler and moderator temperature coefficient is generally observed.
- Decay heat characteristics are slightly different (smaller short-term, but larger long-term effects).

This potentially results in a lower shut-down margin and faster transient response; radiologically, the different decay heat response will mitigate the accident response but aggravate long-term (e.g. storage) behaviour.

These effects are mainly counteracted by fuel and core design, analogous to the introduction of new fuel types. In particular, the design and subsequent safety analysis takes the specific characteristics of MOX into account, and ensures that the existing safety limits are met. Results of transient/accident analysis reported [54] indicate only minor differences between acceptably designed UO_2 and MOX cores, as long as the amount of MOX fuel remains below about 50% of the total core loading. In some cases, utilities may have to make plant changes such as raising the boron concentration (by increasing the boron content in the injection tank, the tank capacity, or the boron enrichment level).

Modelling difficulties associated with the insertion of MOX have been encountered, e.g. larger than normal differences between calculated and measured detector signals in MOX cores indicate that the modelling accuracy of steady-state methods may not be as good as in the case of UO_2 fuels/cores. In some cases, the modelling has been or is being improved; the verification and validation of physics modelling for MOX remains an important issue, that is closely coupled to the issue of uncertainty analysis.

Although the assumption that safety criteria of UO_2 and MOX fuel are identical appears to be generally accepted, some questions on a possibly different behaviour of MOX, especially at high burn-up, remain. The different MOX isotopes and pellet microstructure could lead to differences in e.g. the fission gas release characteristics, and thus indirectly affect criteria such as RIA. The review of the individual criteria should therefore include MOX fuel, as appropriate.

MOX fuel offers similar changes in fuel pellet material; UO_2 is replaced by PuO_2-UO_2 mixed oxide in which the PuO_2 content can vary from 2 to 13 wt% according to the rod position within the fuel assembly and the design criteria. For the case where MOX fuel has been used, the geometry, the dimensions, and the cladding material may be identical for UO_2 and MOX rods. In most countries the plutonium comes from recycling of “burnt” fuel; in addition, some effort has been applied to

burning weapons-grade plutonium in commercial reactors in both the United States and Russia. The introduction of new, advanced and/or MOX fuel can lead to a mixed core situation, i.e. fuel assemblies of different designs jointly reside in a core.

4.3. Mixed assembly cores

With the introduction of new fuel types (advanced designs, MOX, etc.) a "mixed core", i.e. a core consisting of more than one particular design, automatically comes to pass. The fuel and core design must ensure that the newly introduced fuel is compatible with the residing fuel from a physics and thermal-hydraulic point of view; fuel and core safety limits are principally unchanged, but may have to be adapted to the mixed core situation.

Each fuel type comes with a set of specific safety criteria, such as LHGR, oxidation or PCI. These limits are established by the respective fuel supplier, and must be met whether the core is mixed or not. Other limits, such as the safety limit CPR or SDM, that relate to the entire core, must be analysed by the responsible safety analysis engineer (usually at the fuel supplier).

The mixed core situation is thus basically covered by the safety analysis that the fuel and core design responsible suppliers perform. If utilities do not change fuel vendor, the various analyses are internally coherent; as long as the supplier design and monitoring methods are approved, no additional action is needed.

If however more than one fuel vendor is involved, the utility must take appropriate action to ensure that the different methods and correlations do not carry over any inconsistencies or mismatches.

The mixed core situation is thus basically covered by an appropriate design and analysis, which should cover the following areas:

- Neutronic and thermal-hydraulic compatibility, examples: local and global reactivity level, bundle flow characteristics (e.g. risk of flow starvation in neighbouring bundles by low pressure drop for BWRs, or axial flow variations due to local flow redistribution for PWRs).
- Development of safety limits, both for each individual fuel type and for the mixed core.
- Safety analysis in which the mixed core features and incompatibilities are taken into account as appropriate.

There may be an influence on safety limit settings, on account of the mixed core specific features; this influence is comparable to differences in limit settings due to cycle specific features, and does not in itself constitute a basic change in safety criteria. The influence on safety criteria from fuel type specific features, corresponding to the changes in design and materials, is already considered separately.

When performing verification and validation of physics/thermal-hydraulic models, the mixed core features should be accounted for. Of particular concern are

data that cannot be shared between fuel vendors; here, special arrangements need to be made to warrant a conservative setting and monitoring of safety limits.

4.4. Slow or incomplete control rod insertion

During the past few years, a malfunctioning rod scram was observed in several PWRs due to a slow or incomplete rod insertion. The changed scram reactivity may affect the fulfillment of the shutdown margin requirement, as well as the general transient/accident response.

As a temporary measure, the effect of changed scram reactivity on SDM and transient response, based on the observed behaviour, is taken into account for safety analysis and core design; the cycle specific design and reload safety analysis are adapted as appropriate.

Root cause analyses have shown that the mechanical properties of fuel assembly (leading to excessive fuel assembly distortion) and/or rod cluster control assembly (leading to rod swelling) are responsible; adjusting/improving the mechanical design is expected to lead to final resolution of this problem.

The safety criteria themselves are thus considered unaffected.

4.5. Axial offset anomaly

When substantial CRUD build-up occurs in the upper part of a PWR core, especially in high-power assemblies, fission rates are reduced due to boron-containing species ($LiBO_2$, Ni_2FeBO_5) being absorbed into the CRUD layer. As a result, the power distribution shifts towards the bottom of the core, causing a reduction in SDM and an increase in local peaking. During plant operation an anomalous, bottom peaked, power distribution is observed; should the power shift persist, burn-up effects will eventually reverse the power shift setting off a top peaked power distribution near the end of the cycle. The bottom peaked power distribution will tend to reduce SDM, thereby causing deviations in the estimated critical position of control rods, and will also tend to increase local peaking.

This phenomenon, called axial offset anomaly (AOA), has been observed mainly in high energy cores at several PWRs in the United States [55]. Power reductions in the hope of releasing the lithium metaborate from the CRUD proved not to be very successful; utilities could however continue plant operation within the licensing basis by reducing power and/or introducing operating restrictions. Later on, as the amount of subcooled boiling at the fuel rod surfaces in the top of the core was identified as the most significant condition for AOA to occur, methods to evaluate and limit nucleate boiling were implemented for high energy cores: since then, few AOA incidents have been reported.

The utilities and vendors are still continuing their investigation of this phenomenon. An EPRI-sponsored group of industry specialists was asked to address this issue and make operations management recommendations in case of AOA.

Without attempting to make recommendations on this issue, the working group recognises that finding remedies against AOA during an operating cycle may be rather difficult, and hence the operator may not have any other option than to perform safety evaluations as soon as an AOA is observed, to confirm the validity of the licensing basis and to predict the possible change in SDM and peaking factors. From reload design analyses the amount of nucleate boiling can actually be evaluated ahead of time, and hence there may be a possibility to control the effect of AOA by design. In some plants, removing heavy CRUD deposits with advanced ultrasonic methods has been successfully employed [56].

It is not expected that AOA will directly affect any of the fuel safety criteria. The actual numbers of some safety criteria, notably SDM, may change for those power plants (i.e. PWRs with high energy cores) affected.

4.6. Cladding diameter increase

For WWERs, it was observed experimentally that single event PCI criteria no longer protect against stress corrosion cracking beyond a creep and cyclic accumulation of plastic deformation of 0.4%. Thus, a design (strain) criterion limiting cladding diameter increase of 0.4% was put in place, covering creep and cyclic accumulation of plastic deformation. For practical purposes, this design criterion is transformed into an operational recommendation to limit the number of significant power transients (including scram, start-ups, etc.).

For western reactors, no such limit is defined; the requirement is considered to be covered by existing PCI criteria. On the other hand, the cladding diameter change during base irradiation (including gaseous swelling of the fuel pellets and creep of the cladding) has to remain below 1% strain in western reactors. This limit has been set up to be maintained during the whole irradiation, to consider the hydraulic section of the fuel assembly channels and thus the DNBR margins.

4.7. Cladding elongation

Following a general fuel design requirement, the fundamental mechanical and hydraulic functions of the assembly shall not be impaired due to irradiation growth of fuel rods and channel; in particular, the fuel assembly shall give sufficient space for differential rod growth to occur without it becoming restrictive. For western reactors, no explicit elongation (axial growth) design limits are defined. The vendor design process includes verification of the general design requirement against values obtained from experimental data (in-pile and out-of-pile) with suitable uncertainty analysis.

4.8. Radial peaking factors

The radial peaking (Fr for WWERs or enthalpy rise hot-channel factor $F\Delta h$ for PWRs) is sometimes used as a limit to prevent DNB and for WWERs also to prevent reaching saturation temperature of the coolant on the assembly outlet under normal operating conditions and AOOs. A radial peaking factor (Kr or Fxy) is derived by including the uncertainties in measurements, design methods and

fabrication tolerances; this becomes one of the limits for reload design purposes. The limit is also verified during operation with the use of core monitoring programmes. For most western reactors, the radial peaking is employed to indirectly verify the DNBR criterion not only for core design but also during plant operation; for this reason, it is sometimes specified in the technical specifications of the plants.

For WWERs with Russian legislation, this is a licensed limit as no operating limit DNB is defined.

4.9. 3D peaking factor

A total peaking or "hot spot" factor is defined for design purposes to limit local power peaking during normal operation. The limit is also verified during operation with the use of core monitoring programmes.

For western reactors and for WWERs, the three-dimensional (3-D) peaking factor is employed to indirectly verify LHGR as well as the DNBR operating limit not only for core design but also during plant operation; for this reason, it is sometimes specified in the technical specifications of the plants.

Special attention must be paid to WWER-440 fuel due to potentially large local power peaking (up to 70%) in the fuel surrounding the connecting part between the absorber and the fuel follower of the control rod. Recently Hf containing absorber segments have been introduced to handle this problem.

4.10. Cladding stability

Cladding stability limits are defined to prevent cladding collapse due to ovalisation. For western reactors these are normally design limits, constraining elastic and plastic deformation, which are verified analytically. The maximum as-built fuel tube ovality is typically defined in the tubing specification and 100% inspected to ensure analytical basis is valid.

For WWERs, deformation is also verified against design limits and ovality is traced analytically during the expected lifetime of the fuel rod. As the integrity of the plant primary circuit is checked every four years at a higher than normal operating pressure, it must also be verified that the cladding does not collapse during this test. Thus, an ultimate pressure is calculated at which the cladding would collapse and compared against the pressure operating limit associated with such tests; if the ultimate pressure is below this operating limit, the fuel design must be changed.

5. Observations

In considering the fuel criteria discussed in this report, some are of greater interest to the nuclear industry (PCI) than to the regulatory authorities, and vice versa (CHF). In the same manner, some criteria are well established and well documented. Other criteria are evolving and may be impacted by emerging issues or the better understanding of these issues.

Complete or sufficient information is not available for a number of issues discussed in this report. These include CRUD deposition, cladding oxidation and hydriding, rod internal gas pressure, pellet-cladding and thermal-mechanical loads, fuel melting, fuel fragmentation, cladding embrittlement, gap activity, radioactive source term, high burn-up, mixed-oxide fuel, slow or incomplete control rod insertion, axial offset anomaly, cladding elongation and cladding stability. Under the auspices of the OECD Nuclear Energy Agency, active research is being conducted in many of these areas through programmes including the Halden Reactor Project in Norway, the Studsvik Cladding Integrity Project in Sweden, and the CABRI International Project in France. These issues have been, and will continue to be addressed by the Working Group on Fuel Safety, as directed by the NEA Committee on the Safety of Nuclear Installations.

The process of including an issue in this list is quite subjective. However, the existence of such a list suggests the need for further research and investigation in a number of areas, as identified above, and suggests themselves for further study and attention of the working group.

6. Summary and conclusions

The following is a synopsis of the review of the individual safety related criteria and issues of special concern, together with recommendations for further action.

To some extent, the current framework of fuel safety criteria remains applicable, being largely unaffected by the "new" or modern design changes; the numeric values of the individual safety criteria may, however, change in accordance with the particular fuel and core design features. Some of these values have already been – or are continuously being – adjusted. However, adjustments to or revisiting of several other criteria (RIA, LOCA, PCMI) also appear to be needed, on the basis of experimental data and the analysis thereof.

For this (re)assessment of fuel safety criteria, the following process is recommended:

- Continue to further develop best-estimate analysis methods, together with a suitable uncertainty analysis, in all areas of safety analysis.
- Continue to perform experimental studies for benchmarking of best-estimate codes and extending the verification validation basis for safety criteria and the codes (the amount of testing may be reduced as code quality advances).
- Review, and adjust or change where necessary, safety criteria based on the above codes and test data; define or quantify necessary margin to safety limits.

The working group considers international research programmes necessary to support the industry developments as these will contribute to a more detailed and realistic representation of LWR accident scenarios.

7. Glossary

3D	Three-dimensional
AEKI	*Atomenergia kutatóintézet*/Atomic Energy Research Institute (Hungary)
AOA	Axial offset anomaly
AOO	Anticipated operational occurrence ("normal" transient)
BWR	Boiling water reactor
CABRI	Test reactor in France
CHF	Critical heat flux
CNRA	Committee on Nuclear Regulatory Activities (OECD/NEA)
CPR	Critical power ratio
CRUD	Chalk river unidentified deposits (on cladding)
CSN	*Consejo de Seguridad Nuclear* (Spain)
CSNI	Committee on the Safety of Nuclear Installations (OECD/NEA)
DBA	Design basis accident
DHC	Delayed hydride cracking
DNB(R)	Departure from nucleate boiling (ratio)
ECCS	Emergency core cooling system
EDF	*Électricité de France*
ENSI	Swiss federal nuclear safety inspectorate
EPRI	Electric Power Research Institute (United States)
GRS	*Gesellschaft für Anlagen und Reaktorsicherheit* (Germany)
IAEA	International Atomic Energy Agency
INPO	Institute for Nuclear Power Operations (United States)
IRSN	Institut de radioprotection et de sûreté nucléaire (France)
JAEA	Japan Atomic Energy Agency
JNES	Japan Nuclear Energy Safety Organisation
KAERI	Korea Atomic Energy Research Institute
LHGR	Linear heat generation rate
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MOX	Mixed-oxide fuel (U and Pu)
NEA	Nuclear Energy Agency (OECD)
NPP	(NPS) Nuclear power plant (station)

NRC	Nuclear Regulatory Commission (United States)
NRI	Nuclear Research Institute (Czech Republic)
NSC	Nuclear Science Committee (OECD/NEA)
NSRR	Nuclear safety research reactor (Japan)
OECD	Organisation for Economic Co-operation and Development
OTS	Operating technical specification(s)
PCI	Pellet cladding interaction (see stress corrosion cracking)
PCIOMR	Pre-conditioning interim operating management recommendation
PCMI	Pellet cladding mechanical interaction
PIE	Post-irradiation examination
ppm	Parts-per-million
PWR	Pressurised water reactor
RCCA(S)	Rod cluster control assembly(s)
RCS	Reactor coolant system
RIA	Reactivity initiated accident
SCC	Stress corrosion cracking
SCIP	Studsvik Cladding Integrity Project (SCIP)
SDM	Shutdown margin
STUK	Radiation and Nuclear Safety Authority (Finland)
TÜV NORD	*Technischer Überwachungsverein* (Germany)
VNIINM	Bochvar All-Russian Scientific Research Institute for Inorganic Materials
VVER	Russian-designed pressurised water reactor (WWER)
WGFS	Working Group on Fuel Safety (OECD/NEA/CSNI)
WPNCS	Working Party on Nuclear Criticality Safety (OECD/NEA/NSC)

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Nuclear Fuel Safety Criteria Technical Review

Most of the current nuclear fuel safety criteria were established during the 1960s and early 1970s. Although these criteria were validated against experiments with fuel designs available at that time, a number of tests were based on unirradiated fuels. Additional verification was performed as these designs evolved, but mostly with the aim of showing that the new designs adequately complied with existing criteria, and not to establish new limits.

In 1996, the OECD Nuclear Energy Agency (NEA) reviewed existing fuel safety criteria, focusing on new fuel and core designs, new cladding materials and industry manufacturing processes. The results were published in the *Nuclear Fuel Safety Criteria Technical Review* of 2001. The NEA has since re-examined the criteria. A brief description of each criterion and its rationale are presented in this second edition, which will be of interest to both regulators and industry (fuel vendors, utilities).

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