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Nuclear Fuel Safety Criteria Technical Review

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FOREWORD

The goal of nuclear reactor safety is to ensure that the operation of commercial nuclear power plants does not contribute significantly to individual as well as societal health risks. Reactor safety is thus primarily concerned with the prevention of radiation-related damage to the public from the operation of commercial nuclear reactors; safety limits are introduced to avoid fuel failures during normal operation, or to mitigate the consequences of reactor 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 a reactor accident. Fuel safety criteria that relate to fuel damage are then specified to ensure that these limits are not exceeded. Numerous criteria related to fuel damage are used in safety analyses. These criteria, however, may differ from country to country. Some criteria are used to minimise cladding degradation during normal operation, and some are used to maintain cladding integrity during anticipated transients, thus avoiding fission product release. Others are used to limit fuel damage and ensure core coolability during design-basis accidents, or to limit the public risk from low probability severe accidents.

With the advent of advanced fuel and core designs, the adoption of more aggressive operational modes and the implementation of more accurate (best-estimate or statistical) design and analysis methods, the question is being raised whether safety margins have remained adequate. To begin to provide an answer, this report reviews present fuel safety criteria, examining whether - and in what way - they are affected by "new" design elements, and identifies what information (experimental or analytical) might be needed to either confirm the adequacy or to adjust or redefine nuclear fuel safety criteria.

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EXECUTIVE SUMMARY

With the advent of advanced fuel and core designs, the adoption of more aggressive operational modes and the implementation of more accurate (best-estimate or statistical) design and analysis methods, there is concern over whether safety margins have remained adequate. Most – if not all – of the currently existing safety criteria were established during the 1960s and early 1970s, and verified against experiments with fuel that was available at that time, mostly with unirradiated specimens. Verification was of course performed as designs progressed in later years, however mostly with the aim to be able to prove that these designs adequately complied with existing criteria, and not to establish new limits. The OECD/CSNI/PWG2 Task Force on Fuel Safety Criteria (TFFSC) was therefore given the mandate to technically review the existing fuel safety criteria, focusing on the "new design" elements (new fuel and core design, cladding materials, manufacturing processes, high burnup, MOX, etc.) introduced by the industry. It was also agreed that it should identify any additional efforts that may be required (experimental, analytical) to ensure that the basis for fuel safety criteria is adequate to address the relevant safety issues.

In this report, fuel-related criteria are discussed without attempting to categorise them according to event type or risk significance. For each of these 20 criteria, a brief description of the criterion is presented as it is used in several applications, along with the rationale for having such a criterion. New design elements, such as different cladding materials, higher burnup, and the use of MOX fuels, can affect fuel-related margins and, in some cases, the criteria themselves. Some of the more important effects are mentioned in order to indicate whether the criteria need to be re-evaluated. The discussion may not cover all possible effects, but should be sufficient to identify those criteria that need to be addressed. A summary of these discussions is given in Section 7.

As part of the assessment of the safety criteria, the Task Force members looked at various issues, as they relate to one or more criteria, that have become of special interest. These topics included high burnup, core management, MOX, mixed cores, incomplete control rod insertion, and axial offset anomaly.

In addition, an attempt was made to assess the current level of methods and codes, which are used to verify the criteria and margins. As code development activities are widespread, the Task Force could not identify all such activities but focused on those needed to adequately analyse the effects of new design elements.

The Task Force did not extensively review all ongoing and future research programmes. However, a few examples of research programmes are given that contribute to investigating the phenomena and mechanisms of fuel behavior under transient/accident conditions. These include hot-cell testing at Argonne National Laboratory (ANL) in the U.S., the Halden Reactor Project in Norway, research and development work by Belgonucleaire in Belgium, the Cabri test reactor and related programmes in France, and the Nuclear Safety Research Reactor (NSRR) programme in Japan.

As a result of all of the above assessments, the Task Force considers that the current framework of fuel safety criteria remains generally applicable, being largely unaffected by the "new" or modern design elements; the levels (numbers) in the individual safety criteria may, however,

change in accordance with the particular fuel and core design features. Some of these levels have already been – or are continuously being – adjusted; level adjustments of several other criteria (RIA, LOCA) 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 develop best-estimate (nominal) analysis methods, together with a suitable uncertainty analysis, in all areas of safety analysis.

- Continue to perform experimental verification (through selected experiments), for benchmarking of best-estimate methods and extending the verification basis for safety criteria (the amount of testing may be reduced as methods quality advances).

- Review and adjust where necessary, safety criteria levels based on the above methods and test data and quantify necessary margin to safety limits.

The Task Force considers that the research programmes such as those being carried out at the Halden reactor in Norway, the ANL, the Cabri reactor in France and the NSRR in Japan are necessary to support further safety developments as these will contribute to a more detailed understanding and realistic modeling of LWR accident scenarios.

Note: This report contains the results and conclusions from an evaluation by a group of experts. In various instances, the detailed results correspond to a majority opinion and not necessarily to the opinion of every single member of the group. Furthermore, the report content does not necessarily embody the opinion of the organisations that the individual group members represent.

1. INTRODUCTION

With the advent of advanced fuel and core designs, the adoption of more aggressive operational modes and the implementation of more accurate (best estimate or statistical) design and analysis methods, there is a concern if safety margins have remained adequate. Historically, fuel safety margins were defined by adding conservatism to the safety limits, which in turn were also fixed in a conservative manner; here, the expression "conservatism" expresses the fact that bounding or limiting numbers were chosen for model parameters, plant and fuel design data, and fuel operating history values. Unfortunately, as these conservatisms were not quantified (or quantifiable), the amount of safety margin available or the reduction thereof is difficult to substantiate.

For the regulator it is important to know the margins and their basis, as the utility requests approval of new fuel or methods; likewise, for the utility and vendor it is important to know what margins are available, to identify in which direction further progress may be made to optimise fuel 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 – if not all – of the currently existing safety criteria were established during the 60s and early 70s, and verified against experiments with fuel that was available at that time, mostly with unirradiated specimens. Verification was of course performed as designs progressed in later years, however mostly with the aim to be able to prove that these 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 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 toward conditions where less operational and experimental experience exists. The OECD/CSNI/PWG2 Task Force on Fuel Safety Criteria (TFFSC) was therefore given the mandate to technically review the existing fuel safety criteria, focusing on the "new design" elements (new fuel and core design, cladding materials, manufacturing processes, high burnup, MOX etc.) introduced by the industry. It should also identify if additional efforts may be required (experimental, analytical) to ensure that the basis for fuel safety criteria is adequate to address the relevant safety issues.

2.   TASK FORCE OUTLINE

2.1   Fuel safety criteria: historical perspective, background of task force

The goal of reactor safety is to ensure that the operation of commercial nuclear power plants does not contribute significantly to individual as well as societal health risks. Reactor safety is thus primarily concerned with the prevention of radiation-related damage to the public from the operation of commercial nuclear reactors; safety limits are introduced to avoid fuel failures during normal operation, or to mitigate the consequences from reactor 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 then specified to ensure that these limits are not exceeded.

Fuel safety criteria 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 60s and early 70s. The main idea in this development process was that the consequences of these postulated events, which can occur in the nuclear power plant are inversely proportional to their probability. For the sake of simplicity these events were divided into two categories: anticipated transients (or anticipated operational occurrences, AOO) 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 there are the more frequent events (classified in most countries as inherent to normal operation, or condition 1 events), and the less frequent ones (classified in most countries as faults of moderate frequency, or condition 2 events). Within the accident spectrum there are events that lead to failure of a few fuel rods (e.g. reactor coolant pump seizure, in most countries classified as condition 3 events) as well as postulated accidents of low probability (referred to as Design Basis Accidents (DBA), in most countries classified as condition 4 events) such as those which result in Loss-of-Coolant Accidents (LOCA), or Reactivity Initiated Accidents (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, safety criteria allow for only a very small number of fuel rods in the core to experience the boiling crisis. That is, the Departure from Nucleate Boiling Ratio (DNBR) for PWRs or the Critical Power Ratio (CPR) for BWRs shall be determined so that with 95% probability at the 95% confidence level the critical heat flux is not exceeded. For the 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 60s and early 70s a number of experiments were carried out, which provided information about fuel and reactor core behaviour for the more serious DBA conditions. This information was used to develop the fuel safety criteria for these accidents as well as the related analytical methods (computer codes). During the development of these criteria and methods high burnup was thought to occur around 40 MWd/kg; data up to this burnup had been included in data bases for criteria, codes, and regulatory decisions, and it was believed that some extrapolation in burnup could be made. By the mid 1980s, however, changes in pellet microstructure had been observed from a variety of data at higher burnup along with increases in the rate of cladding corrosion. It thus became clear that something new was happening at high burnup and/or new operating environments, and that continued extrapolation of transient data from the existing low burnup/traditional operating environment data base 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 USA, for example, the USNRC has licensed fuel burnup in commercial nuclear reactors up to 62 MWd/kg (average exposure of the peak rod). In Europe, high burnup verification programmes are in progress, with fuel rods in lead test assemblies attaining exposures of up to 100 MWd/kg.

As a result of the world-wide trend to increase fuel burnup well beyond the level of 40 MWd/kg and the observations regarding pellet microstructure changes and increased rates of cladding corrosion at higher burnup, a number of programmes were initiated, both of an experimental and analytical nature, to evaluate the effects of the higher burnup on fuel behaviour, especially under RIA and LOCA conditions.

The Halden programme, for instance, extended the range of fuel property investigations to the burnup range of 50-80 MWd/kg [1]. Interest peaked after two tests, related to the fuel behaviour during postulated accidents, were performed by the French in the CABRI Facility and by the Japanese in the NSRR Facility respectively. During these two tests (labelled REP Na-1 and HBO-1), 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 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.

In its 1996 report of *Nuclear Safety Research in OECD Countries* the Committee on the Safety of Nuclear Installations (CNSI) recommended that "Fuel damage limits at high burnup" is a safety research area to which priority should be assigned. Specifically the report said that "Fuel damage limits should be established for the entire range up to high burnup. Limits should be based upon appropriate parameters to ensure fuel integrity (i.e. enthalpy, DNB, cladding oxidation), and should consider the full range of possible transients, including reactivity insertion and LOCAs." Finally the CSNI and CNRA in their December 1996 meeting decided to undertake an effort involving a much broader (than only high burnup related issues) look at fuel behaviour and requirements needed to assure appropriate safety margins of modern fuels and core designs. This effort was assigned to PWG2 and a Task Force was formed to undertake this effort.

Hence, the objective of this Task Force (**Task Force on Fuel Safety Criteria**, or **TFFSC**) was to review the present fuel safety criteria, whether – and in what way – they are affected by the "new" design elements, and to identify what information (experimental or analytical) might be needed to either confirm the adequacy or to adjust or redefine fuel safety criteria. The TFFSC looked at all issues (both design and operational) which could have an effect on fuel safety criteria [Chapter 2.2 of

this report lists and discusses these issues, or "new" design elements]. Moreover, the Task Force focus was on "safety related" issues rather than on "general fuel performance" issues.

Margins may be set differently in different countries, and will thus depend on the technical and regulatory interpretation of the safety criteria.

2.2 "New" elements related to fuel design and operation

The current fuel safety criteria were developed in the late 60s to early 70s and were based on tests and related analyses with the then utilised fuel and core designs, cladding materials (e.g., Zry-2 for BWRs and Zry-4 for PWRs), UO_2 fuel and burnup levels not exceeding 40 MWd/kg. In order to optimise fuel cycle cost, the nuclear industry began work in the mid 80s on new fuel and core designs with the aim of increasing the fuel burnup, e.g. for extending the cycle length or upgrading the power level. This again lead to a number of basic design changes, e.g. new cladding materials; also the use of fissile plutonium in mixed oxide fuel (MOX) was considered by some utilities.

Fuel design should be in concord with the general design criteria [2] that governs the design and operation of nuclear power stations. Thus, existing fuel safety criteria are examined against design elements as applicable to date. Table 1 lists the current fuel safety criteria against those new design elements that may affect them; a list of all new elements considered is provided in this table, while some principal elements are highlighted below.

Generally **high fuel burnup** is of great interest to nuclear operators due to their need for reducing fuel cycle cost, nowadays enhanced by the introduction of electric power deregulation. Thus, high burnup capability is very much in the centre of new design elements, and has triggered activities world-wide. This issue has already been introduced above, and will be addressed for the assessment of the individual safety criteria below; in addition, section 5.1 of this report will summarise the high burnup issue separately.

In order to reach high burnup with higher linear heat ratings, the **cladding materials** for LWR fuel rods used over the last 30 years [based on Zry-2 for BWRs and Zry-4 for PWRs] have undergone major changes during the last 10-15 years. To reduce the corrosion rate and hydrogen uptake in the metal, the concentration of Sn was reduced (low Sn alloys such as the SIEMENS ELS cladding) and Nb containing alloys (e.g. Zirlo of Westinghouse). Apart from the full cladding tubes, several inner and outer liner concepts have been introduced to cope with various performance problems (e.g. BWR inner liner for PCI resistance, PWR outer liners for corrosion reduction at high power).

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. In the PWR case the number of fuel rods per element was increased from 14x14/15x15/16x16 to 16x16/17x17/18x18 pins. The BWR fuel pin number was increased within the same trend from an 8x8 to a 9x9 or even 10X10 geometry. In parallel to these changes the cladding wall thickness was also reduced and lies today in the range of 0.6 to 0.75 mm which may include an inner liner (barrier) of about 70 μm.

MOX fuel rods differ from UO_2 fuel rods only by the 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 10 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 are identical for UO_2 and MOX rods. In most

countries the Pu comes from recycling of "burned" fuel; in addition there is the possibility of burning weapons-grade Pu in commercial reactors in both the USA and Russia. The introduction of new, advanced and/or MOX fuel leads to a **mixed core** situation, i.e. fuel assemblies of different designs jointly reside in a core. This issue will be addressed separately.

With the severe thermal duty that occurs with some of the current fuel management strategies, 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 ppm) during sometime, which could increase the corrosion rate.

Hydrogen may be added to the coolant to decrease the amount of oxygen present, which is formed by radiolysis, which consequently decreases the zircaloy oxidation rate. 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.

2.3   Task Force approach to assessing the potential effects of new elements

Numerous criteria related to fuel damage are used in safety analyses: these 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 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 for LOCA performance.

In Table 1 fuel-related criteria are therefore listed without attempting to categorise them according to event type or risk significance. We will leave the matter of relative importance of these criteria to the regulatory agencies and others who utilise this information. For each of the criteria in Table 1, we present a brief description of the criterion as it is used in several applications along with the rationale for having such a criterion.

New "elements," such as different cladding materials, higher burnup, and the use of MOX fuels, can affect fuel-related margins and, in some cases, the 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.

Table 1. **Current fuel safety criteria**

Safety related criteria	Category*	"New" elements affecting criteria	List of "new" design elements
(a) CPR/DNBR	A, B, C	1, 2, 5, 6, 7, 9	1. New fuel designs
(b) reactivity coefficient	B, C	2, 5, 6, 7, 8, 9	2. New core designs
(c) shutdown margin	A, B, C	1, 2, 5, 6, 7, 8, 11	3. New cladding materials
(d) enrichment	A, B, C	1, 2, 5	4. New manufacturing procedures
(e) crud deposition	A	1, 2, 3, 4, 5, 7, 10	5. Long fuel cycle
(f) strain level	A, B	1, 3, 4, 7, 8	6. Uprated power
(g) oxidation	A, B, C	3, 4, 7, 8, 10	7. High burnup
(h) hydride concentration	A, B, C	3, 4, 7, 8, 10	8. MOX
(i) internal gas pressure	A, B, C	1, 5, 6, 7, 8	9. Mixed core
(j) thermal-mechanical loads	A, B	1, 3, 4, 7	10. Water chemistry changes
(k) PCI	A, B, C	1, 2, 3, 4, 6, 7, 8, 11	11. Current/new operating practices
(l) fuel fragmentation (RIA)	C	7, 8	
(m) fuel failure (RIA)	C	1, 3, 4, 7, 8	
(n) cladding embrittlement/PCT (non-LOCA run away oxidation)	C	3, 4, 7, 8	
(o) cladding embrittlement/oxidation	C	3, 4, 7, 8	
(p) blowdown/seismic loads	C	3, 7	
(q) assembly holddown force	A, B, C	1, 11	
(r) coolant activity	A, B, C	5, 6, 7, 8	
(s) gap activity	C	5, 6, 7, 8	
(t) source term	C	5, 6, 7, 8	

* A – normal operation, B – anticipated transients, C – postulated accidents

3.   REVIEW OF SAFETY CRITERIA

In this Chapter, the possible implications from new design elements 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 [2].

Research is being conducted by various organisations around the world on the effects of new design elements such as different cladding materials, higher burnup, and the use of fissile plutonium in MOX fuels. The TFSSC 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   CPR/DNBR

The most widely used safety criteria for cladding integrity are related to critical heat flux. These are the critical-power ratio (CPR) for BWRs and a departure-from-nucleate-boiling ratio (DNBR) for PWRs. In a PWR the critical heat flux occurs when the bubble density from nucleate boiling in the boundary layer of the hot rod is so great that adjacent bubbles coalesce and form a vapour film across the surface of the rod. Heat transfer across the film is relatively poor such that, if the heat flux is further increased (or the coolant flow is reduced), the cladding temperature would rise rapidly and substantially. Cladding melting or rapid oxidation could then take place and result in failure of the cladding. Similarly, in a BWR the critical heat flux at the onset of transition boiling must not be exceeded.

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 margin during normal steady-state operation, an adder 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.

The CPR/DNBR safety limit is derived from a statistical analysis, in which the fuel assembly specific heat flux characteristics are accounted for with a specific core loading. Historically generic bounding assumptions were made on the core loading (using a so-called "reference core"); in this way, the safety limit depends only on the fuel assembly characteristics and is not re-evaluated unless the fuel type changes. In view of advanced fuel and core designs, mixed core situations and to reduce unnecessary conservatism, the safety limit is nowadays often re-evaluated based on the cycle specific core loading and thus becomes a cycle specific limit. This way, all necessary details of the actual fuel and core design are accounted for.

The statistical method to establish the safety limit is usually based on a Monte Carlo technique, that calculates the critical heat flux for each assembly in the core, at multiple exposure

points during the cycle, while introducing random variations in the input variables (manufacturing data, plant measured data, critical heat flux correlation) based on their known uncertainties; also the model uncertainties are treated statistically.

The critical heat flux correlation links the critical heat flux and the operational parameters; this correlation is a mathematical fit to data from full-scale tests (Note: PWR tests typically employ a smaller rod array) that the fuel supplier performs for every assembly design specifically. Thus, the critical heat flux correlation is fuel assembly type specific. The correlation parameters include pressure, flow, subcooling, power peaking within the assembly and axial power shape; tests are performed while varying each of these parameters separately. To reduce the amount of (time consuming and costly) testing, hydraulic (subchannel) models such as VIPRE or COBRA are utilised to form a "response function" to variations in certain variables; after sufficient validation of these models against test data, further tests may then be reduced. The same critical heat flux correlation is used for core monitoring, to derive critical heat flux for the actual operating point (heat balance etc.).

As a consequence, CPR/DNBR safety limits 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) that are successfully applied world-wide; to date, no fuel has failed due to inadequacies in establishing these safety limits.[1] Although it appears that there is no need to change either the safety criteria or the methods to establish them, some testing seems to be needed. This includes full scale testing to establish the proper thermal-hydraulic modelling of new assembly designs. Also, statistical methods are to follow method improvements such as the detailed pin power calculation capability of modern 3-dimensional steady-state methods.

In addition, a concern that is related to high burnup ought to be addressed. 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 burnup, or by heavy crud layers. CPR and DNBR correlations are, in general, developed from data on unoxidized, or lightly oxidised, fresh cladding tubes and may not be accurate for high-burnup cladding. Material and fabrication variations may make small changes in heat transfer characteristics, but the effect of oxidation on surface conditions could be an important effect.

3.2   Reactivity coefficient

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 in the 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 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).

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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 4 rods operated around 20-30% in excess of the safety limit CPR for several months prior to failure.

The reactivity coefficients depend on the following four 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

The fuel temperature or Doppler coefficient $d\rho/dT_f$, where ρ is reactivity, responds promptly to the enthalpy deposited in the fuel, whereas the other coefficients are delayed. The fuel time constant, 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 on the fuel burnup. The higher the burnup the harder the spectrum, so in general the change of the fuel temperature coefficient with burnup 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. 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 higher fuel burnup, 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. Depressurization 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 any safety problem as far as reactivity is concerned.

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

High fuel burnup usually implies the loading of more reactive fresh fuel bundles. This additional reactivity is compensated by fuel (addition of burnable poison) and core design, keeping in mind that the basic safety criterion (negative total reactivity coefficient) must be fulfilled.

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

3.3   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 (ARI) 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 TechSpec 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 start-up; 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.

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. 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. 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 TechSpecs, but are 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 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 and boron worths 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);

- 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 elements. 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. For the assessment of modelling adequacy and necessary verification, see Chapter 4.

3.4   Enrichment

Enrichment limits around 5 wt% [235]U are used in connection with criticality considerations for fabrication, handling, and transportation. For some high-burnup 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 less than 5%. Neither benchmarks of code performance nor the bases for extrapolating code performance in the enrichment range of 5-10% have been well established. Moving into this range will require care because the physics of criticality begins to change as enrichments reach 6% and beyond, where single moderated assemblies can go critical and criticality of weakly moderated or unmoderated systems becomes possible. Enrichments above 5% 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.

3.5   Crud deposition

Crud deposition on the fuel is usually taken into account for fuel design purposes. The amount of crud deposited, sometimes as a function of burnup but at least at the end of the fuel lifetime, is a conservatively assumed value which is verified against data from measurements (e.g. crudscrape). 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, CILC).

New design elements, such as cladding materials and their manufacturing processes, 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. 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 for one aspect may go in the opposite direction for the other. New fuel and core designs, high burnup 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.

Criteria on crud deposition are considered "derived" criteria, and only indirectly safety related. No firm limits are likely to be needed here as criteria relative to the limiting phenomena (oxidation, hydriding, PCI) are already in place.[2]

There is a concern with large crud depositions in PWRs leading to boron pick-up, thereby causing distortion of the core axial power profile and reduced SDM: this issue is discussed in section 5.6 of this report.

Finally, 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 [3].

3.6   Strain level

Generally conservative design limits are taken for stress (e.g. around 1% yield or tensile strength at operating temperature) or strain (e.g. 1% max. circumferential elastic and plastic strain, and max. 2.5% permanent axial and tangential strain – caused by fuel swelling – at end of fuel life.) 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 burnup range.

These limits, together with others such as PCMI (section 3.9) and fuel rod internal pressure (section 3.8), are used to define the fuel specific thermal-mechanical limit; this limit is expressed as a burnup dependent linear heat rate curve (in W/cm). The curve conservatively bounds all thermal-mechanical phenomena; it is set to cover for transient thermal/mechanical overpower, which ranges from about 10 to 50%. For a further discussion of the thermal-mechanical limit, see section 3.9.

Stress and strain analyses are performed by the fuel vendor, with models that are constantly being benchmarked against available experimental test data. Benchmarking can also be obtained from sophisticated fuel performance codes (e.g. the COMETHE or the ENIGMA code.) The use of such codes allows the expected fuel duty to be modelled and therefore there is no particular problem investigating the effects of new core designs or unusual operating practices.

Because these mechanical properties depend on material composition, fabrication, fluence, and hydrogen content, they will clearly be affected by new design elements, in particular by high burnup. Hence, continuous verification [4] of fuel design models is essential to ensure that the proper basis for design and operation exists.

3.7   Oxidation and hydriding

Oxidation and hydriding are directly related to fuel performance for normal operation, transients and accidents. Oxidation degrades material properties, most importantly the cladding thermal conductivity (with a consequential increase in the stored energy of the fuel), whereas hydriding leads to embrittlement; these phenomena are increasingly important at higher exposures, as the dependence on burnup is not linear. For these reasons, Zircaloy cladding materials have been

2. Unexpected large amounts of crud have recently been observed at the River Bend NPP (USA) associated with a number of fuel failures. The root cause appears to be thermally-induced accelerated corrosion, due to elevated iron and copper deposits associated with a chemistry excursion early in the operating cycle.

highly optimised during the past 10-20 years. For BWRs, the optimisation was directed mainly towards reducing nodular corrosion, due to the CILC related fuel failures which occurred during the 70s and early 80s; a too high degree of optimisation (with very small sized secondary phase particles) may however adversely affect uniform corrosion, which makes the choice of materials and manufacturing processes a complicated balance act. In recent years, other materials have been developed for PWRs in addition to the standard Zircaloy 4 (e.g. Zr-Nb alloys).

Uniform corrosion rates differ between PWRs and BWRs. With the much lower operating coolant temperature and the more corrosion resistant Zry-2 as basic cladding material, uniform corrosion is much less critical for BWRs; in contrast, PWRs are less susceptible to nodular or local phenomena (e.g. CILC, enhanced shadow corrosion) due to much less oppressive heat transfer and flow conditions.

For design purposes, oxide thickness and hydride concentration limits are normally assumed at end of fuel life. Values are usually in the range of 100 micron and 500-600 ppm, respectively; these values are taken from "experience", and represent upper bounds on data measured from fuel exposed in commercial reactors. The 100 micron oxide thickness also represents the level at which there is a steep increase in the likelihood of oxide spalling, which will unfavourably influence hydriding growth and hence further oxidation.

In several countries the design limit of a average cladding oxide thickness at end of fuel life of 100 micron, and of average hydride concentration of 500-600 ppm, have effectively become licensed/approved safety criteria via the approval of fuel vendor design methodologies.[3] 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 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. At increasing burnup 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 can have a deteriorating effect on fuel rod behaviour under irradiation. It prevents the axial transport of fission gases in a fuel rod and induces a severe pellet cladding mechanical interaction. Due to bonding internal oxidation of the fuel cladding is becoming increasingly important as a function of fuel burnup. At high fuel burnup the bonding effect is important; full bonding occurs at fuel burnup of about 50 MWd/kg. Bonding causes diffusion of fission products such as I, Cs and Cd into the cladding. These effects cause internal oxidation and embrittlement of the fuel cladding and should be considered when assessing the effects of oxidation.

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. Therefore, consideration should still be given to impose limits on oxide thickness and hydride concentration.

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3. Unfortunately the interpretation of these criteria is not unambiguous, because the clad region over which the average is taken is often ill defined.

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. Also, from a LOCA performance point of view, a high hydrogen content (2000 ppm and above) may lead to severe quench resistance degradation (reference the French HYDRAZIR test data.)

Extensive research, including tests on corrosion rates, and fuel inspection programmes in commercial NPP has led to a basis for burnups up to about 50 MWd/kg; also at even higher burnups some amount of data is available. Cladding containing new zirconium alloys and multi-layer type fuel claddings have been developed in recent years, and tested out under irradiation in commercial NPPs (lead test assemblies or rods, some of these programmes with subsequent destructive testing). Such tests will produce data to cover normal core operation, and will have to be pursued to extend fuel burnup substantially beyond the 50 MWd/kg level. In addition, tests to cover transient and accident fuel performance will have to be made.

In summary, as corrosion of Zircaloy 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.

3.8 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 (see also section 3.9.) The fission gas release is dependent on:

- the fuel microstructure and chemistry;

- its development with time; and

- the fuel temperature, which is strongly influenced by the power rating and the burnup.

At high burnup (higher than 40-50 MWd/kg) fission gas release tends to increase rapidly. Also available experiments involving fission gas release under transient conditions, indicate very high fission gas releases in the high burnup region of the fuel; furthermore, fission gas release is strongly influenced by the formation of the peripheral fuel rim at high burnup which is especially important for transient/accident conditions. These phenomena are not yet well understood, nor can existing analytical tools predict them satisfactorily.

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 and, more importantly, of the heat transfer between the pellets and the cladding due to the resulting gap size modification. The fission gas 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, the fission gas release in various rods can be highly irregular. The high internal rod pressures can have an important effect on fuel cladding behaviour (ballooning, burst, etc.) under transients and postulated accidents. 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 other option the rod internal pressure may exceed the RCS pressure, but is limited so that the instantaneous cladding creep-out rate due to an internal rod pressure greater than the reactor coolant system pressure is not expected to exceed the instantaneous fuel swelling rate, i.e. the fuel to cladding gap does not open (this is the so-called "no lift-off" criterion.) At high fuel burnup this could, in transient and accident conditions, lead to a very high internal pressure of the fuel rod with subsequent high stored energy, cladding ballooning and bursting, which could challenge core coolability, and thus the level/limits resulting from this safety criterion.

These criteria themselves should not be affected by new design elements, although methods to demonstrate compliance will be affected.

Furthermore, MOX fuel has been found to produce a higher fission gas release as compared to UO_2 fuel. An acceleration of fission gas release with exposure is observed, also because of the higher linear heat generation rate in MOX due to the higher reactivity level. More than the criteria themselves, the issue here is to demonstrate the compliance with the criteria. The development of rod internal pressure as function of burnup on MOX fuel needs to be well characterised, also in consideration of the production method and plutonium content in the MOX fuel. The consequence of rod internal pressure build up must also be carefully studied. One such study, a lift-off test series (IFA 610) with UO_2 and MOX fuel rods of 50-60 MWd/kg is currently being performed at the Halden research reactor. For the UO_2 rod, lift-off occurred at an overpressure of around 130 bar; first results for the MOX fuel rod indicate an even higher lift-off pressure. On the basis of the Halden findings a refined "no lift-off" criterion has been proposed by some vendors, considering outward creep rate/strain and tensile stress due to overpressure.

For a discussion on possible effects of new design elements on the thermal-mechanical (LHGR) limit, see section 3.9.

3.9 Thermal mechanical loads (PCMI)

Pellet-to-cladding mechanical interaction (PCMI) refers to the stress on the cladding from an expanding pellet, especially during a transient. Pellet expansion results mainly from thermal expansion, and if the stress is large enough it can result in cladding failure. PCMI differs from the related PCI phenomenon (see section 3.10) inasmuch as the latter refers to power ramps where the stress is held for a long period of time and corrosion is necessary for cracking to take place.

The avoidance of mechanical fracture of the clad during transients due to PCMI, which is the basic safety criterion, is traditionally covered by the limit on uniform cladding (plastic and elastic) strain of 1%, as already described in section 3.6.

A range of power-increasing transients where PCMI may be important are addressed in FSAR and reload licensing safety analyses (e.g. loss of feedwater heating in a BWR and steamline break in a PWR). 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 burnup that could result in early cladding failure by PCMI. First, the pellet-to-cladding gap closes at higher burnup, eliminating 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 power increase, which would contribute to the cladding strain. Third, cladding ductility is reduced significantly by radiation embrittlement already at intermediate exposure such that a mechanical failure becomes more likely. Fourth, cladding hydriding further reduces the ductility at high exposure mainly at lower cladding temperatures. Under these circumstances PCMI failures could also 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.

Experimental data on PCMI for LWR fuel have been obtained from e.g. the Halden reactor project and the international R&D programme from Belgonucleaire, covering a range of burnup up to 60 MWd/kg; so far, none of these results point towards PCMI effects being prohibitive at high burnup. However, as these experiments usually aimed at investigating other high burnup effects such as fission gas release and thus PCMI data were obtained "on the side", it appears warranted to perform more tests focusing on PCMI directly.

In summary, some concerns regarding the effect of high burnup 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.

The **thermal-mechanical limit** (a burnup dependent curve) is established while including the PCMI phenomenon, as well as various other phenomena (fuel rod internal pressure, stress/strain, fatigue, fuel melting, clad corrosion and ballooning) that have been discussed elsewhere in this report. 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. The parameters in the areas mentioned above are treated as distributions, with best-estimate uncertainties, and a Monte Carlo model varies all these parameters for the multiple calculation of important design features (e.g. internal rod pressure). With a known design/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, should generally be applied in order to reduce unnecessary conservatism. This implies, however, that such methodologies need to be well validated and verified; thus, experimental tests are to continue to provide the basis for such verification and validation.

The basic safety criterion – the avoidance of mechanical fracture of the clad – is not affected by new design elements, however the current limit (1% strain) may change.

3.10 Pellet cladding interaction (PCI)

Pellet cladding interaction (PCI) failures are due to stress corrosion cracking in the cladding material, which is associated with local power ramps during reactor startup or maneuvering (e.g. rod adjustments/swaps, load follow.) Both the stress from the power increase and the corrosion level in the cladding are necessary conditions for PCI. A crack, initiated at a microscopic defect in 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 do not fail by PCI, neither do fuel rods operated at constant power.

The PCI phenomenon has been extensively investigated after multiple PCI failures during the 70s. To control the PCI phenomenon, operating rules (also called 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 higher burnup the fuel is less able to withstand ramping) and differ between various fuel types. To establish and validate these rules, extensive power ramp tests were performed – by basically each fuel vendor – notably in the STUDSVIK and PETTEN test reactors; thus, the failure threshold of the cladding is known very well up to 40-50 MWd/kg and for power ramps well beyond normal operation. A certain amount of ramp testing has also been performed at higher burnups (up to 60-70 MWd/kg) later on, e.g. in France. Verification of these rules was performed in various commercial NPPs.

The PCI limits/rules typically contain a maximum ramp rate for power increase (in 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.

During the 80s and 90s, PCI resistant fuel types were developed based on a small layer of zirconium ("barrier" or "liner", with or without small additives like Sn or Fe) at the inner part of the clad as a more permanent remedy. Also, the modern fuel assembly designs 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 PCI.

The PCI mechanism is sensitive to the gas composition in the gap. At high burnup (higher than 40 MWd/kg) fission gas release becomes increasingly important and due to increased releases in iodine, cadmium and cesium the gap gas composition is more aggressive and can enhance PCI induced fuel cladding failure. Since fission gas release from MOX fuel pellets will differ from that of UO_2 pellets, a MOX effect is also expected. Also the mechanical interaction between the fuel pellet cladding (bonding) is becoming an increasingly important phenomenon in the high burnup area. For these reasons it is important to continue to perform ramp tests especially in the high burnup region, in order to establish PCI threshold values properly. This is equally important in view of the introduction of new cladding materials.

During transient conditions PCI fuel failure mechanisms must be taken into account. The PCI failure mechanism may be significant in transients like the control rod withdrawal error (RWE), and in subcooling (cold water) transients generally. It should however be remembered that such failures pertain to previously intact fuel, and that therefore the radiological consequences of such transients are normally minor.

By and large, PCI limits (rules) are not licensed; each NPP is designed to cope with a certain number of small fuel failures, and TechSpec limits (particularly the ^{131}I concentration level in the primary coolant) will bound plant operation.

Nevertheless PCI rules do pertain to safe fuel performance, and regulators will maintain that for non-PCI-resistant fuel these limits be adequate and that NPPs obey these rules for core operation. The 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 PCI limits up to about 50 MWd/kg burnup. A continuation of ramp testing is recommended, to improve the basis at the higher burnups and as appropriate to the fuel design adopted. At the same time, fuel performance models should be further developed and benchmarked against these ramp tests; finally, with sufficient modelling, the amount of testing could be reduced.

3.11   Fuel fragmentation (RIA)

To avoid the loss of coolable geometry and the generation of coolant pressure pulses, peak fuel enthalpy criteria are used as limits for reactivity-initiated accidents (RIA). To date, an enthalpy value of 280 cal/g has been used in the USA and other countries based on data from early RIA fragmentation measurements prior to 1974 (e.g. SPERT and TREAT tests in the USA); this value corresponds to the melting of UO_2 which causes 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 [5], as well as later FRAP-T calculations and 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. Some of those 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 [5].

At the same time, the question whether this limit should be identical for unirradiated and irradiated fuel was brought up (ref. results from the PBF-RIA 1-1 test). Similarly an international industry working party, led by EPRI, suggested a value of around 240 cal/g for fresh and low burnup fuel.

Recent experiments in the French CABRI test reactor and the Japanese NSRR test reactor using high burnup fuel samples have resulted [5] in fuel particle dispersal for deposited energies well below 200 cal/g. It is clear that, for high burnup 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 in UO_2-RIM and 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 burnup effect in UO_2 is evident.

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, that grows as a function of exposure: at about 45 MWd/kg the RIM-zone is of the order of 200 µm. Fundamental studies are being performed to clarify what role the RIM-zone plays in transient/accident situations.

Because pellet structure and fission gas release from MOX fuel pellets will differ from those in UO_2 pellets, an additional MOX effect is also apparent [17,18].

To summarize, from a safety point of view, it is considered that the analytical verification of meeting the limit in the range of around 230-280 cal/g may well be sufficient to ensure a coolable geometry for fresh and very low burnup fuel. For the assessment of this limit at high burnups there is a

need for further understanding of the fragmentation process and the effects of high burnup (in particular the effect of the RIM-zone and the MOX clusters) thereon. Verification against more realistic RIA experiments (planned with the CABRI – water loop) is therefore desired. This improved understanding should also contribute to better modelling with fuel performance codes (see Chapter 4).

3.12   Fuel failure (RIA)

For a RIA, the number of fuel rod failures must be calculated so that the radiological doses to the public can be estimated. In most countries the current fuel failure limit is based on the definition per section 4.2 of the US Standard Review Plan [7] as a maximum radially averaged fuel enthalpy of 170 cal/g for BWRs and as a DNB criterion for PWRs. Based on some of the RIA experiments at CABRI and NSRR during the 1990s, with fuel rods at a burnup of approx. 50 MWd/kg or higher, an assessment of the adequacy of this limit appeared desirable. In this respect, various limit values as function of burnup have been proposed based either on direct experimental data renditions or on relevant parameters, such as cladding oxide thickness.

Results from all RIA tests (SPERT, PBF, CABRI, NSRR) with Zircaloy-clad fuel above about 5 MWd/kg show failures from a pellet-cladding mechanical interaction (PCMI) rather than high-temperature failures related to critical heat flux [8]. It is believed that the reduction in gap and ductility due to radiation creep and embrittlement/hydrogen absorption is 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 burnup appear to alter the failure mechanism and make the critical heat flux criteria inappropriate. 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 with Zr-1%Nb-clad VVER fuel show completely ductile behaviour without PCMI failure even at high burnup (these tests was performed with a pulse width of about 700 ms); BIGR tests show no PCMI failure up to 160 cal/g enthalpy at 48 MWd/kg (here the pulse width is 3 ms) [9].

Thus, especially in the higher burnup range where experimental data are lacking, technically based safety criteria and verification of the analytical models for fuel performance should be pursued. Here the future experimental CABRI programme, which test facility will be modified to include a PWR water-loop, is likely to provide very valuable results. The importance of the post-DNB condition has already been demonstrated in early PBF tests [5] 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 burnup fuel under realistic conditions appears warranted, and the CABRI water loop could be very useful for such investigation.

Also NSRR and separate effect tests in facilities like PROMETRA, PATRICIA, SILENE (France) and from the ANL – test programme [13] (USA) as well as the JAERI – test programme [17] (Japan) should provide worthwhile information.

3.13   Cladding embrittlement/PCT (non-LOCA run away oxidation)

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 2 700°F is sometimes used to demonstrate coolability. The 2 700°F limit was taken from early data estimates of the fuel failure boundary for LOCA conditions (2 700°F and 17% of clad thickness oxidised by metal-water reaction). This limit was established in the 1969-1971 time period prior to the ECCS hearings in the

U.S., 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.

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 burnup. Therefore, the peak cladding temperature during the transient 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.

The behaviour of highly burnt fuel under this condition is relatively unknown. The relevance of the above criterion should therefore be confirmed experimentally.

3.14   Cladding embrittlement/oxidation (LOCA)

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 with unirradiated zircaloy tubes, it was found that cladding would not become embrittled enough to fragment if the peak cladding temperature remained below 2 200°F (1204°C) and the total oxidation did not exceed 17% of the cladding thickness before oxidation. These embrittlement criteria (reference 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 hydrogen generation, however this is rather for containment integrity than against embrittlement (limit is usually 1% related to total possible clad oxidation).

The embrittlement/oxidation criteria were developed in the 60s and early 70s; experimental verification and validation included tests with zero or low burnup fuel. Nowadays fuel operation exhibits typical oxidation levels of up to 100 microns and hydrogen concentrations up to 500 ppm at the time of discharge (these levels are usually employed as criteria for fuel mechanical design, which frequently become licensed limits, see section 3.7). 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, and will hopefully be resolved from the ongoing French [10] and Japanese [17] as well as the planned ANL [13] experiments. A different criterion, that might be more suitable especially at high burnup, 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 burnup:

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

- consequences from hydrogen-induced ß-phase stabilising effect on clad strain;

- consequences from fine fragmentation of the fuel, filling the space in the ballooned clad (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);

- clad behaviour during quenching and long-term cooling: changes in oxidation rate and clad embrittlement, UO_2-RIM or MOX clusters fission gas induced fuel swelling (losses in mechanical strength may become important if clad wall strength is significantly weakened during irradiation);

- modelling accuracy, e.g. clad 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;
- potential subchannel blockage.

Some of the concerns and issues mentioned above could well be addressed by performing out-of-pile (e.g. hotcell) experiments on ballooning, corrosion and fracturing. A number of research programmes are already ongoing (e.g. in France: TAGCIS, TAGCIR, HYDRAZIR [11], CINOG, EDGAR [11] and in Japan at JAERI [17]) or have been initiated (e.g. in the USA at ANL [12], with a very comprehensive test programme) which include integral as well as separate effect tests.

The USNRC has been examining the 10CFR50.46 Appendix K requirements [13], which impose conditions governing LOCA analysis, in recent past. For future verification and review of LOCA safety criteria, the results from this examination should be taken into account.

On the whole, LOCA safety criteria are still considered adequate for modern fuels to meet the basic limitations on core coolability and radiological release. It is believed, that the results of the above tests along with complementary ductility tests for long-term LOCA behaviour, should be sufficient for verifying the current safety criteria, especially for the effect of burnup, and to further develop and validate LOCA modelling.

3.15   Blowdown/seismic 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 losing 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, App. 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:

- fuel rod fragmentation shall not occur (can be met by verification that fuel rod stresses are within limits per ASME III, App. F);

- control rod insertion shall not be impaired (verify that combined loads do not displace the fuel assembly from the support piece, that stresses are within the above ref. limits etc.);

- limit spacer distortion to ensure rod coolability (verify that spacer distortion or failure do not occur).

Verification is performed analytically. As fuel designs may have different dynamic properties, this analysis is not only fuel design but also core design dependent; in particular, the mixed core situation (see also section 5.4) should be addressed explicitly.

Safety criteria in these areas are not directly affected by the new design elements; however, considering the analytical verifications just mentioned, methods used to analyse the seismic/LOCA event should be well verified and validated.

Also, design requirement changes on allowable structural loads for earthquakes during and after a LOCA may be needed at high burnup, because the strength and ductility of high-burnup cladding, guide tubes (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 burnup 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 burnup for some materials. Altered materials properties for high-burnup 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 burnup is of importance.

3.16   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 start-up conditions and hot full power conditions, at BOL and EOL.

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 therefore a major consideration at high burnup 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.

In summary, safety criteria are not considered to be affected by new design elements. For the analytical verification of these criteria, it is important that sufficiently well validated and verified models are in place; also, material properties need to be available, especially at high burnup, to be able to choose and analyse materials adequately.

3.17   Coolant activity

In most countries, limits are specified in the plant Technical Specifications on the concentration of ^{131}I (sometimes also of ^{137}Cs) in the primary coolant; numbers are typically around 2×10^9 Bq/t. This allows the NPPs to operate with a certain (small) number of fuel failures; the plant systems have been designed to cope with fuel failures of this magnitude. Aside from this TechSpec limitation, no fuel safety criteria on coolant activity exist.

Usually, as soon as larger ^{131}I concentrations are measured that would challenge the TechSpec limit, the plant operational staff prepares for plant shutdown to identify and replace the leaking bundles, so that plant operation within TechSpec 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 TechSpec 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 elements.

3.18   Gap activity

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 design purposes.

Volatile fission products 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 burnup 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. Hence fission product release to the gap is found to increase at high burnup; 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.

3.19   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. Also, 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 PRAs 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 elements 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 elements – especially from high fuel burnup – is estimated as follows:

The main effects that could impact source terms as well as core melt progression at high burnup are:

a) a reduction in the amount of unoxidized 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.

A reduction in the amount of unoxidized zirconium metal in the core could diminish the severity of core melt and subsequent ex-vessel phenomena by lowering the reaction heat from metal oxidation. 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 depressurization, 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 burnup, 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): anyhow changes are expected to be small, yet experimental programmes are being undertaken to provide an adequate validation basis for these analysis codes at higher burnups.

Considering the above factors, it is unlikely that high burnup will have a significant effect on source terms or core melt progression. A similar statement can be made about MOX fuel, particularly in light of the diminishing difference between MOX fuel and UO_2 fuel as burnup increases. It should be noted, though, that this conclusion is based on a limited assessment and might be altered by a more thorough evaluation. Also, the implementation of a revised source term [14] could affect the dependence on new designs and materials.

4.   ASSESSMENT OF ANALYSIS METHODS

Code development activities are wide spread, and the models and correlations involved in these codes are numerous in comparison to the fuel safety criteria discussed above. Therefore, the Task Force will not attempt to identify all development programmes and the many models and correlations involved. However, it is of interest to assess which models and correlations would be affected by new design and operational elements. In this section, a general attempt will be made to identify the weak areas with the use of several examples of ongoing development work.

Analytical methods (computer codes) are extensively used in safety analysis, either as stand-alone codes or in a coupled manner. Thus the methods emphasis in safety analysis is on transients and accidents; however, steady-state analysis is also needed to establish initial conditions. For example, for LOCA analysis it is very important to have an accurate value for the stored energy of the fuel pellets at the time the accident is initiated which can be released during the accident: this stored energy comes from a steady-state calculation for normal reactor operation. Also, for deeper understanding of RIA failure mechanismus it is very important to have accurate values for the fission gas content in grain boundaries and porosities at the time of accident initiation, especially for UO_2-RIM and MOX clusters which may promote fuel swelling and grain separation during the accident; this fission gas content is obtained from a steady-state calculation at normal operating conditions.

Several types of computer codes that are used in safety analysis are sensitive to fuel-related parameters. In the previous Chapter, the need for further code development and verification has been stated on many occasions; new design elements, such as different cladding materials, higher burnup, and the use of MOX fuels, can affect the performance of these codes. In the following paragraphs some of the more important impacts on these codes are mentioned in order to assess if the codes might need further verification or even modification. Examples are given rather than systematically covering all possible models and effects; this should still be sufficient to identify analytical areas that need to be addressed in more depth.

4.1   Steady-state fuel rod codes

These single-rod codes, like FRAPCON, COMETHE, TRANSURANUS, METEOR and TOUTATIS calculate thermal quantities such as radial temperature profile and fission gas release to the gap, and mechanical quantities such as creep deformation and irradiation growth. Results are used for many purposes like axial clearance between rods and end fittings, internal gas pressure to compare with system pressure, cladding oxide thickness to compare with established limits or to initiate transient calculations, stored energy for LOCA analysis and fission gas repartition between grains, grain boundaries and porosities for RIA fuel failure mechanisms studies. These codes consist of numerous models and correlations to describe gap conductance, material properties such as thermal conductivity and specific heat, radial power profiles, stress-strain equations, mechanical properties, creep properties, fuel-swelling, fuel-densification, waterside corrosion, and hydrogen absorption.

In recent work in the U.S., the FRAPCON steady-state fuel rod code was modified [15] for burnup up to 65 MWd/kg. Previously the code had been validated up to about 40 MWd/kg. Nine

models within the code were found to need modification because burnup effects on the phenomena being modelled were large enough to warrant a change. Those models were:

- fission gas release;

- fuel thermal conductivity (including effects of burnable absorber additions);

- fuel swelling;

- fuel pellet cracking and relocation;

- radial power distribution;

- solid-solid contact gap conductance;

- cladding corrosion and hydriding;

- cladding mechanical properties and ductility;

- cladding axial growth.

It is expected that this experience would be typical for other fuel rod codes at high burnup. For different cladding materials, of course the cladding-related models like corrosion, mechanical properties, and growth would be affected. Similarly for MOX fuel pellets one would expect changes in radial power distribution, fission gas release and thermal conductivity, and perhaps also in swelling, cracking and relocation (see e.g. the COMETHE model development for high burnup MOX fuel based on Halden experiments).

4.2   Transient fuel rod codes

These single-rod codes, like FALCON/FREY, FRAPTRAN, and SCANAIR also calculate thermal quantities and mechanical quantities.[4] The range of models and correlations included in these codes is quite similar to that for the steady-state codes. The major differences between the transient and the steady-state codes are:

- the steady-state codes do not include transient heat-transfer terms in their solution equations; and

- the transient codes do not include long-term phenomena like creep.

However, the transient codes need to incorporate models, correlations, and properties for cladding plastic stress-strain behaviour at elevated temperatures, effects of annealing, behaviour of oxides and hydrides during temperature ramps, phase changes, and large cladding deformations such as ballooning. The mechanical description of the cladding is two-dimensional ideally, but models of lower dimension are used as well. Other differences also come into models like fission gas release, which can have long-term and short-term components. The transient codes are used for analysing fuel rod response to transients and accidents like RIA and LOCA and may include failure models.

In principle, the nine models affected by high burnup for the steady-state fuel codes also identify the important phenomena that would need to be modified in the transient codes. In some cases, like fission gas release, the transient models would be quite different from the steady-state

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4.   The EPRI FALCON – new FREY – code also includes the effect of bonding.

models. For burnup higher than 40-50 MWd/kg, special attention should be devoted to the proper modelling of thermal characteristics of the so-called RIM-zone with its structural change and the consequent degradation of the fuel thermal conductivity. The phenomena of non-transient swelling and axial growth may not be modelled in the transient code because the transient time period is too short for significant non-transient swelling or axial growth. Different cladding materials and MOX fuel pellets will have similar effects as described for the steady-state codes.

In summary, the need for a revision of the transient fuel codes is based on the same observations as applied to the steady-state fuel codes. Especially for the analysis of fast transients, the proper modelling of the RIM-zone and the MOX clusters should be included in these revision efforts.

4.3   Reactor kinetics codes

Reactor kinetics codes, like RAMONA, PARCS, SIMULATE-K, CORETRAN or SAPHYR, are used to calculate assembly averaged neutron flux and power and sometimes also peak values (pin power) in a reactor core during transient events. Whole-core events like coolant temperature changes in a PWR cause global power changes that can be analysed properly using point-kinetics models. In the BWR, similar transients require the use of 1-D kinetic models due to the important axial changes of the of the power distribution. However, localised events like rod ejection in a PWR, control rod drop or regional power oscillations in a BWR require multidimensional neutron kinetics analysis. Point-kinetics models need reactivity coefficients, effective delayed neutron fraction, generation time and reactivities of control rods as input. One- and 3-D kinetics models need neutronic input parameters like assembly averaged neutron cross-sections (typically condensed into two energy groups), delayed neutron fractions (typically on a nodal basis) that are obtained from reactor static codes.

Modern (static) nodal codes using two energy groups are more capable of handling the sharp flux gradients that appear if an assembly containing MOX is placed beside an assembly with UO_2. This technique is also used for the transient codes. For accident conditions such as RIA, it is hypothesised that more neutron energy groups may be needed to handle the harder neutron spectrum resulting from the increasing fraction of plutonium in the fuel (this would as a consequence also lead to a multi-group formalism for the reactor static codes.)

The reduction in delayed neutron fraction and number of neutrons released per absorption in the fuel will be handled by the assembly burnup code and are hence factored automatically into the transient cross section sets.

Fuel models in these codes are normally not very detailed; model extensions could be considered (e.g. thermal conductivity) for better modelling of high burnup fuel.

4.4   Reactor static codes

Reactor static codes (fuel assembly burnup programmes) are used to generate the neutronic cross sections used in the multi-dimensional kinetic codes in function of burnup, other history parameters and the instantaneous thermal-hydraulic parameters such as fuel temperature and moderator density temperature.

The 3-D (static) core simulators are also fed with this data for the core calculations. Power distributions, burnup-distributions reactivity worth and core-wide reactivity coefficients are

determined with these codes. 1-D cross sections are generated from the steady-state flux-distributions using auxiliary collapsing programmes.

These results are used to select the proper burnup-dependent thermal properties in the transient codes such as thermal conductivity. Accurately predicting the burnup therefore is an important feature of the reactor static codes (the proper prediction of the isotopic composition of the fuel for a given burnup is also important for an accurate prediction of the decay heat.)

Currently, modern static codes have successfully been applied for the analysis of cores which contain MOX-fuel as part of the core loading. As no "new" reactor physics is expected at high burnup, an extension of the validation base of these codes to higher burnup – especially for improving the quality of the prediction of the isotopic composition for exposed fuel – appears the only issue of importance here (with benchmarking data being provided by experimental programmes.)

4.5 Thermal-hydraulic codes

Large reactor systems codes like TRAC, RELAP, CATHARE, ATHLET, and RETRAN are used to calculate flow, temperature, and pressure during normal operation and transients. These codes typically contain point kinetics models to model the reactor power in PWRs and 1-D kinetics for BWRs, as well as simple 1-D fuel rod models. They may, in some cases, be coupled with 3-D neutron kinetics codes or even more detailed fuel rod codes. As a minimum, thermal-hydraulic codes generally have at least one coolant channel that models a few individual fuel rods, typically an average rod and a hot rod. The fuel rod models in the thermal-hydraulic codes will usually include heat transfer correlations (cladding to coolant), a constant or variable (dynamic gap model) value for gap conductance, and average values for thermal conductivity and heat capacity. For LOCA analysis, these codes typically contain ballooning, burst and oxidation models.

Although simpler in practice, the fuel models in the thermal-hydraulic codes describe the same basic phenomena as those in the transient fuel rod codes and would thus be affected in a similar way by high burnup, changes in cladding material, and MOX fuel pellets. A proper selection of the thermal properties of both the fuel pellet and the cladding for the average and the hot rod (mainly burn-up dependence) is considered of high importance. In general, these simplified models are benchmarked against the detailed transient fuel rod models.

It is judged that there is no obvious need for a revision of this type of large reactor system codes due to new design elements, however the balloning and rupture models may be refined in the future.

4.6 Subchannel codes

This class of codes is used to analyse the flow distribution inside a fuel assembly. Normally a 3-D model for the two-phase flow, 1-D models of the different fuel rods and detailed models for the heat transfer between the cladding surface and the coolant are included. Models for the calculation of critical heat flux are built-in. This type of code is used to demonstrate compliance with DNBR/CPR-requirements.

Modern fuel bundle designs with part length rods and large water holes pose new challenges to the subchannel codes. The proper prediction of the void distribution within the bundle, especially near the non-heated water rods, may require improvements in the so-called lateral void drift modelling.

Margin to CPR/DNB is determined using correlations specific to individual fuel-design; thus, new fuel designs come with revisions of the CPR/DNBR-correlations (see also section 3.1). These correlations have fuel-design specific formulations, in which burnup-dependence is normally not considered. CPR/DNB is determined based on channel averaged parameters. Here, only the conditions at the cladding surface appear to depend on high burnup or other new design elements.

It is hypothesised that a highly oxidised cladding surface could have different boiling characteristics than a fresh surface; thus, some minor influence on the critical heat flux may not be excluded. Reliable data on this possible effect are currently unavailable.

In summary, there is a need for a revision of the subchannel codes in the areas of the geometrical representation of modern fuel bundles and the modelling of the void distribution.

4.7   Reactor core structural analysis codes

Codes like ANSYS and ABACUS are used to model the linear and non-linear response of fuel assembly and core components, e.g. during vessel accelerations from earthquakes and asymmetric blowdown following a LOCA pipe break. Elastic analyses are usually performed, and calculated loads or stresses are compared with ASME allowable values.

Irradiation and oxidation alter the strength of core materials as burnup increases, and these changes in materials properties can be input to the structural analysis codes. However, the codes that are used for this purpose usually presume that the material has ductility so that plastic deformation would occur beyond the elastic range. This is included into the safety factors incorporated in the analyses. The validity of this analytical approach may be questioned for some core materials that loose ductility and become brittle at high burnup.

5. SPECIAL TOPICS

As part of the assessment of the safety criteria the TF members looked at various issues, as they relate to one or more criteria, that have become of special interest. These topics are separately discussed below.

5.1 High burnup

First, a short summary of the situation regarding the high burnup issue (licensed burnup limits, burnup levels achieved today and expected burnup extensions) is given.

Licensed burnup limits depend on the type of fuel and fuel vendor; licensed limits may refer to local (sometimes referred to as "peak pellet") burnup levels and/or rod average burnup levels and/or assembly average burnup levels. Examples of licensed burnup limits are as follows:

- a maximum rod-average burnup of 62 MWd/kg for some fuels in USA;

- a generic limit for the maximum rod-average burnup of 48 MWd/kg in Finland;

- a generic limit for the assembly average burnup of 52 MWd/kg in France (recently increased from 47 MWd/kg) for UO_2 fuel (MOX fuel is currently still limited to 3 one-year cycles of insertion);

- a generic limit for the assembly average burnup of 40 MWd/kg in Finland;

- generic limits for the assembly average burnup of 48 MWd/kg for PWRs and of 50 MWd/kg for BWRs in Japan;

- maximum assembly average burnup are 65 MWd/kg for PWRs and 53 MWd/kg for BWRs, for some fuels in Germany;

- maximum assembly average burnup of 48 to 60 MWd/kg, or maximum local burnups of 65 to 70 MWd/kg for various different fuels in Switzerland.

The general trend to increase burnup levels is likely to continue in the near future. In the US, rod average burnup levels up to 62 MWd/kg have been achieved and requests for a peak rod average burnup to about 75 MWd/kg have been discussed with the USNRC. In France, Japan and Germany, utilities aim to increase the maximum assembly average burnup from 42-50 MWd/kg to 52-55 MWd/kg with a corresponding peak pellet average burnup of about 65 MWd/kg within the next few years. In Switzerland, peak pellet average burnup up to 65 MWd/kg have already been attained. All of these numbers differ substantially from the 40 MWd/kg burnup level originally expected during the development of the criteria.

In Chapter 3 the high burnup issue and its possible consequences have been brought up many times. The industry greatly focuses on high burnup, 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 [16].

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 burnup level up to those currently licensed, However, the Task Force also considers that there is a need for further research to:

- experimentally verify the validity of safety criteria for high burnup, in particular for burnup levels beyond those currently licensed; and

- further develop and benchmark the analytical models used to define and monitor high burnup safety criteria.

This is particularly important as the economical incentive for high fuel burnup still prevails; industry studies to increase fuel burnup to the 100 MWd/kg level are already being attempted.

In terms of research programmes, the Task Force particularly encourages the Halden [1], ANL [13], the NSRR [17] as well as the French [11,12] test programmes which hopefully will provide more information on the LOCA and RIA behaviour of high burnup fuel. Also the extension of the CABRI test programme, with tests in a PWR-water-loop that is planned to be constructed during the next years, is likely to contribute to development of more realistic RIA-limits [18].

Generally, as pointed out throughout this report, it is important that all aspects related to high burnup are adequately 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 R&D programmes at national or international research centres) will have to be added, while selecting the experimental test cases appropriately and carefully.

5.2   Core management

The fuel cycle costs (FCC) is an important part of the cost for plant operation. Utility strategies to reduce FCC 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.

About 15-20 years ago 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 toward 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 burnup.

Modern loading strategies with a smaller amount of fresh reload bundles on account of the higher fuel discharge exposures will reload a smaller number of fresh bundles, leading to higher power peaking due to higher reactivity of fresh fuel bundles. Safety criteria, notably LHGR, SDM and DNB/CPR, must however still be met; as a consequence, fuel bundles with very high burnup may now have to be loaded into a centre of the core adjacent to fresh fuel bundles. This implies that reaching maximum fuel burnup 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) leads to having fuel with high burnup in the core centre.

This situation may influence the behaviour of the high burnup fuel during transients/accidents. As an example, during a small/medium size LOCA the cladding of the fresh fuel

may collapse due to low clad internal pressure[5] and the high burnup 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 the high burnup fuel clad 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 burnup and fresh fuel bundles. The effect of the fuel failure mechanisms for high burnup 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 burnup 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 verified/benchmarked.

Traditionally the codes used in transient and accident analysis are 1-D. Currently state of the art modelling includes the use of 3-D neutron kinetics codes, though the thermal-hydraulic modelling remains 1-D. With reactor cores becoming more heterogeneous, the capability of present codes for analysing the transient behaviour of high burnup 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 verification 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.

5.3   MOX

In the past reprocessing appeared a viable option in some countries. 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 (RepU) which may be used as carrier material or blended with regular UO_2. Also, during the past few years, the option of using weapons grade high enriched Uranium (HEU) and plutonium has triggered activities in e.g. the USA.

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 of MOX fuel is less characterised than for UO_2 fuel, especially at high burnup. Experiments will continue to be needed to confirm the operational regimes in which MOX fuels are compliant with safety criteria, considering

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

also that MOX performance can be affected by the fabrication route and by the total plutonium content in the fuel (and Pufissile /Pututotal ratio).

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 temp. coefficient is generally observed.

- Decay heat characteristics are slightly different (smaller short-term, but larger long-term effects).

This potentially results in a lower SDM 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 [19] 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 burnup, remain. The different MOX isotopics and pellet (grain) structure could lead to differences in e.g. the fission gas release characteristics, and thus indirectly affect criteria such as RIA, (for example, the MOX test CABRI-REP-Na7 /55 GWd/t, clad-corrosion 50 μm/, led to fuel dispersal and severe fuel-coolant interaction at 120 cal/g, whereas the UO_2 tests CABRI-REP-Na3 /53 GWd/t, clad-corrosion 40 μm/ did not fail despite a narrower pulse of 120 cal/g. [18]). The review of the individual criteria should therefore include MOX fuel, as appropriate.

5.4   Mixed 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. In particular, core monitoring (during plant operation) must be addressed at this point: as an example, in-core detector signals are based on the neutron flux from bundles of various different fuel types, and the models must be adjusted to correctly unfold the individual fluxes or, if the total signal is uniformly divided over all bundles, an appropriate bias must be introduced for those bundles that produce a higher flux.

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 (FSAR type) 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, are already considered separately (see Chapter 3.)

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.

5.5 Incomplete control rod insertion

During the past few years, a malfunctioning rod scram was observed in several PWRs (Europe, USA) due to an incomplete rod insertion (IRI). The changed scram reactivity may affect the fulfilment of the SDM 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 IRI 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 and/or RCCA are responsible for IRI; adjusting/improving the mechanical design is expected to lead to final resolution of this problem.

The safety criteria themselves are thus considered unaffected by IRI.

5.6   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 lithium metaborate (LiBO2) being absorbed into the crud layer. As a result, the power distribution shifts toward 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, burnup 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 (ECP) 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 US [20]. 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 experts 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 Task Force recognises that finding remedies against AOA during an operating cycle may be rather difficult, and hence the operator may not have any other option then 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 [21].

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.

6.   TEST PROGRAMMES

In this Chapter, a few examples of research programmes are given that contribute to investigating the phenomena and mechanisms of fuel behaviour under transient/accident conditions.

6.1   ANL test programme

Because the expected outcome of a LOCA is strongly dependent on flow blockage and cladding embrittlement, the large amounts of corrosion, enhanced oxidation rates, and reduced cladding ductility experienced at high burnup are likely to have a significant effect.

Besides the regulatory criteria, fuel behaviour computer codes are used to help understand experimental results and to perform safety analyses. Codes, such as the NRC's FRAPCON code, have been updated for high-burnup effects insofar as possible. These code updates have been largely confined to the thermal properties of fuel pellets (thermal conductivity, fission gas release, etc.) because the mechanical properties of the cladding have not yet been measured at high burnups – especially under the transient conditions needed.

A test programme at Argonne National Laboratory (ANL), sponsored by the NRC with EPRI co-operation, has two main objectives:

- determine the behaviour of high-burnup fuel under simulated LOCA conditions; and

- establish a database of mechanical properties of high-burnup cladding needed to analyse transients that are important in licensing safety analyses.

There are two subsidiary objectives of a practical nature:

1. establish an analytical or empirical method of estimating the behaviour under LOCA (or RIA) conditions. Since it is impractical to perform integral tests on all cladding varieties, a method will be developed to use information of one cladding type to determine the properties of another cladding types, under the assumption that the cladding types are not too different; and

2. make suitable benchmark tests and measurements on fresh cladding (viz., archive material from earlier NRC tests and sibling material for the high-burnup specimens) to establish low-burnup properties of today's materials and to check for consistency with previous results.

Three types of tests will be performed:

1. oxidation studies, which are needed to develop or validate the kinetics models used in evaluation models;

2. quenching tests under simulated LOCA conditions to evaluate the current acceptance criteria, or provide a database for new criteria, if that is necessary; and

3. structural response tests to evaluate whether external mechanical loads could affect coolable geometry or control rod insertion.

6.2 Halden reactor project (HRP)

The OECD Halden reactor project in Norway is a joint undertaking of national and industry organisations in 20 countries, which co-sponsor jointly financed research. By virtue of advanced in-core instrumentation capabilities, the Halden tests represent the source of critical information, such as fuel temperatures, fission gas release and PCI during service. In recent years the focus of the programme has been more and more placed on high burnup fuel performance both at normal and abnormal power conditions. To this end, high burnup fuels retrieved from commercial power reactors are transferred to Halden and re-fabricated into instrumented test rodlets. These are inserted into test sections and depending on test requirements, in LWR loops. Besides high burnup UO_2, the Halden project is a key source of in-reactor data for MOX fuels. The current Halden programme is intended to explore burnups ranges between 50 and 80 MWd/kg and contemplates a variety of tests at normal operation and in power/coolant transients.

6.3 Belgonucleaire R&D programme

Since 1965, a large part of the R&D work at Belgonucleaire is devoted to supporting LWR burnup increase, with special emphasis on MOX fuel. In this respect, the international programmes managed by Belgonucleaire provide a common database for all participating organisations. Special attention is given to the evolution with burnup of fuel neutronic characteristics and of in-reactor rod thermal-mechanical behaviour. Some highlights from this programme are given below; for a more detailed outline, see e.g. the 1999 programme overview [22].

Burnup of Pu in MOX is characterised essentially by a drop of ^{239}Pu content. The other Pu isotopes have an almost unchanged concentration, due to internal breeding. The reactivity drop of MOX fuel versus burnup is consequently much less pronounced than in UO_2 fuel. The concentration of minor actinides Am and Cm becomes significant with burnup increase: these nuclides start to play a role for total reactivity and helium production.

The thermal-mechanical behaviour of MOX fuel is found very similar to that of UO_2. There are, however, some specific differences. The better PCI resistance recognised for MOX fuel has recently been confirmed. Three PWR MOX segments pre-irradiated up to 58 MWd/kg were ramped at 100 W/cm/min to 430-450-500 W/cm, respectively, followed by a hold time of 24 hours: no fuel failures were observed.

MOX and UO_2 fuels appear to have a different reactivity level which generally leads to a different power history. Moreover, radial distribution of power in MOX pellet comes out less depressed at high burnup than in UO_2, leading to a higher fuel central temperature for a same power rating. Also, the thermal conductivity of MOX fuel is found to decrease with Pu content (typically 4% for 10% Pu.) The combination of these three elements (power history, power profile, and conductivity) leads to larger FGR at high burnup as compared to UO_2.

Helium production is seen to remain low compared to fission gas production (ratio < 0.2). As the faster diffusing element, the helium fractional release is much higher than that of fission gas, leading to rod internal pressure increase comparable that resulting from fission gas.

6.4 CABRI

The CABRI test reactor in Cadarache, France consisting of a driver fuel core in a water pool with a sodium (Na) cooled loop, is operated by the Institute for Protection and Nuclear Safety (IPSN) and is used to test nuclear fuel rods under reactivity-initiated accident (RIA) conditions. The facility is planning to implement a water loop, which would be more representative of PWR designs. The reactor provides pulse widths of about 10 ms, which may be broadened to about 80 ms. Studies of highly irradiated UO_2 and MOX fuels were begun in 1993. Seven tests have been completed with UO_2 fuel and three with MOX fuel. One of the three MOX fuel tests experienced cladding failure (the MOX rod with the highest burnup), whereas an UO_2 rod tested at comparable conditions did not fail (see section 5.3). Fuel dispersal from the failed rod and sodium ejection from the vicinity of the rod were more pronounced than with the UO_2 rods that failed. It is believed that the enhanced fission gas migration to grain boundaries and porosities from the high concentration Pu agglomerates caused the MOX fuel to behave differently from the UO_2 fuel under these conditions. IPSN plans to replace the sodium loop with a water loop to more accurately simulate PWR core conditions. Additional tests are planned with very high-burnup UO_2 fuel and MOX fuel in this facility.

6.5 TAGCIS/TAGCIR/HYDRAZIR

A test programme in the Grenoble and Chinon laboratories (France), sponsored by the IPSN and EdF, studying LOCA embrittlement limits for high burnup Zircaloy4, is in progress. The main results to date are:

- there is no protective effect of the initial (in-reactor) oxide layer against transient (LOCA) oxidation;

- an acceleration in the transient oxidation kinetics of the irradiated or hydrided Zircaloy4 against fresh Zircaloy4 is observed;

- there is no violation of the 17% limit for axially unconstrained tests [11].

These tests do not cover the long-term phase of the LOCA under hydraulic, seismic or handling loads.

6.6 CINOG

A test programme similar to TAGCIS in the Grenoble laboratory (France), sponsored by FRAMATOME and EdF, for fresh M4 and M5 cladding, is in progress [12].

6.7 EDGAR

A test programme in the Saclay laboratory (France), sponsored by FRAMATOME and EdF, studying α/β phase change kinetics, ballooning and burst during a LOCA, for fresh M4 and M5 cladding, is in progress [12].

6.8 NSRR

The Nuclear Safety Research Reactor (NSRR) of the Japan Atomic Energy Research Institute (JAERI) in Tokai, a TRIGA type reactor with an annular core in a water pool, is being used to

investigate the behaviour of fuel rods under RIA conditions. The reactor pulse width is around 5 ms. Studies to determine the behaviour of fresh MOX fuel were initially performed to provide baseline information. These initial tests repeated the conditions of previous tests performed with uranium-based fuel. It was found that the cladding failure mechanism and threshold for fresh MOX fuel were consistent with those for fresh UO_2 fuel and that an effect of plutonium particles (inhomogeneities) was not detected. More recently, JAERI has tested four irradiated MOX fuel rods (20 MWd/kg). Up to the limits of energy deposition tested to date (140 cal/g fuel enthalpy), no cladding failures have occurred in these relatively low-burnup specimens. However, the NSRR tests with irradiated MOX have shown higher fission gas release and larger fuel swelling compared with UO_2 fuel (in agreement with CABRI results), indicating the existence of MOX effects for RIAs.

7. RECOMMENDATIONS AND SUMMARY

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

7.1 CPR/DNBR

There appears to be no need for changing either the safety criteria or the methods to establish them. Some testing seems to be needed, including full scale testing to establish the proper thermal-hydraulic modelling of new assembly designs. Also, statistical methods are to follow method improvements, such as the detailed pin power calculation capability of modern 3-D steady-state methods.

CPR and DNBR correlations are, in general, developed from data on unoxidised, or lightly oxidised, fresh cladding tubes and may not be accurate for high-burnup cladding. Thus, the effect of oxidation on surface conditions ought to be addressed.

7.2 Reactivity coefficients

Although the reactivity coefficients may be affected by new design elements, the effects are not considered to affect the corresponding safety criteria themselves.

7.3 Shutdown margin

Existing SDM criteria are considered unaffected by, and adequate for, the new design elements. 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.

7.4 Enrichment

Enrichment limits around 5 wt% ^{235}U are used in connection with criticality considerations for fabrication, handling, and transportation. For some high-burnup applications, higher enrichments may be needed. Higher enrichment levels will require the assessment of the validity of criticality safety codes and of the equipment for fuel fabrication, handling, transportation and storage. Also, the possibility of recriticality during accidents, in particular in severe accident core melt sequences should be addressed.

7.5   Crud deposition

Criteria on crud deposition are considered "derived" criteria, and only indirectly safety related. No firm limits are likely to be needed here as criteria relative to the limiting phenomena (oxidation, hydriding, PCI) are already in place.

7.6   Strain level

Here, design limits as well as the methods to verify them appear to be well established. However, because fuel mechanical properties depend on material composition, fabrication, fluence, and hydrogen content, they will clearly be affected by new design elements, in particular by high burnup. Hence, continuous verification of fuel design models is essential to ensure that the proper basis for design and operation exists.

7.7   Oxidation and hydriding

As corrosion of zircaloy is 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.

In some countries, there are no formal criteria related to oxide thickness and hydride concentration. However, oxide and hydride influence stress and strain performance, and ultimately the fracture toughness, of the cladding material. Therefore, consideration should be given to impose limits on oxide thickness and hydride concentration.

7.8   Internal gas pressure

High internal rod pressure can have an important effect on fuel cladding behavior under transients and postulated accidents. Two alternative criteria – absolute (vs. RCS pressure) or relative (lift-off) – for acceptable internal gas pressure are currently used in various countries by their regulatory authorities. For MOX fuel, the fission gas release is higher as compared to UO_2 fuel, and the adequacy of the relative (lift-off) criterion for acceptable internal fuel rod gas pressure should be studied.

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

7.9   Thermal-mechanical loads (PCMI)

The existing experimental data on PCMI for LWR fuel do not point towards PCMI effects being prohibitive at high burnup. However, as these experiments usually aimed at investigating other high burnup effects such as fission gas release and thus PCMI data were obtained "on the side".

Some concerns regarding the effect of high burnup 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, for this

purpose. Also, some more testing of PCMI for benchmarking these codes and verifying their results appears to be justified.

The basic safety criterion – the avoidance of mechanical fracture of the clad – is not affected by new design elements, however the current limit (1% strain) may change.

7.10   Thermal-mechanical operating limit

Best-estimate or statistical methods to determine thermal-mechanical operating limits should generally be introduced covering modern fuel and core designs.

7.11   PCI

Although PCI limits are not licensed in most countries, they do pertain to safe fuel performance. A continuation of ramp testing is recommended to improve the basis at the higher burnups and as appropriate to the fuel design adopted. At the same time, fuel performance models should be further developed and benchmarked against these ramp tests.

7.12   RIA – fuel fragmentation and fuel failure

The fuel **fragmentation limit** in the range of around 230-280 cal/g may well be sufficient to ensure a coolable geometry for fresh and very low burnup fuel. For the assessment of this limit at high burnups there is a need for further understanding of the fragmentation process and the effects of high burnup (in particular the effect of the RIM-zone and the MOX clusters) thereon. Verification against more realistic RIA experiments (planned with the CABRI-water loop) is therefore desired. This improved understanding should also contribute to better modelling with fuel performance codes.

In most countries the current **RIA fuel failure limit** is based on the definition per section 4.2 of the US Standard Review Plan [23] as a maximum radially averaged fuel enthalpy of 170 cal/g for BWRs and as a DNB criterion for PWRs. Based on some of the RIA experiments at CABRI and NSRR during the 1990s, with fuel rods at a burnup of approximately 50 MWd/kg or higher, an assessment of the adequacy of this limit appeared desirable. In this respect, various limit values as function of burnup have been proposed based either on direct experimental data renditions or on relevant parameters, such as cladding oxide thickness.

Thus, especially in the higher burnup range where experimental data are lacking, technically based safety criteria and verification of the analytical models for fuel performance should be pursued. Here the future experimental CABRI programme, which test facility will be modified to include a PWR water-loop, is likely to provide very valuable results. Also NSRR and separate effect tests in facilities like PROMETRA, PATRICIA, SILENE (France) and from the ANL – test programme [13] (USA) as well as the JAERI – test programme [17] (Japan) should provide worthwhile information.

7.13   Cladding embrittlement/oxidation

Non LOCA: 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 2 700°F is sometimes used to demonstrate coolability. The behaviour of highly burnt fuel under this condition is

relatively unknown. The relevance of the above criterion should therefore be confirmed experimentally.

LOCA: The current criteria state that the cladding peak temperature should remain below 2 200°F and that the cladding oxidation does not exceed 17% (15% in some countries) in order to avoid clad embattlement/fragmentation in a LOCA. These criteria are based on tests that were performed with unirradiated cladding. For high burnup applications, the 17% is now often interpreted as total oxidation level, which maybe a conservative assumption. The question on whether the oxidation during normal operation should be accounted for and how, is unsettled. This is rather important since it is not unusual that uniform oxidation reaches ~100 μm in high burnup fuel. Ongoing French [11] and Japanese [12] tests as well as the ANL [13] tests will hopefully be sufficient to verify the current safety criteria and further develop and validate LOCA modelling for new designs and high burnup.

On the whole, LOCA safety criteria are still considered adequate for modern fuels to meet the basic limitations on core coolability and radiological release. It is believed, that the results of the above tests along with complementary ductility tests for long-term LOCA behaviour, should be sufficient for verifying the current safety criteria, especially for the effect of burnup, and to further develop and validate LOCA modelling.

7.14   Blowdown/seismic loads

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

7.15   Assembly holddown force

Safety criteria in this area are not considered to be affected by new design elements. Again, for the analytical verification of these criteria, it is important that sufficiently well validated and verified models are in place; also, material properties need to be available, especially at high burnup, to be able to choose and analyse materials adequately.

7.16   Coolant activity

No change of the limit(s) in this area is expected in conjunction with new design elements.

7.17   Gap activity

The release fractions assumed in safety analyses are not safety criteria, but represent conservative numbers used for design purposes. Fission product release to the gap is found to increase at high burnup; 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.

7.18   Source term

It is considered unlikely that new design elements or high burnup will have a significant effect on source terms or core melt progression. This conclusion is however based on a limited assessment and might be altered by a more thorough evaluation.

7.19   Analysis methods

With reactor cores becoming more heterogeneous, the capability of present codes for analysing the transient behaviour of high burnup and fresh fuel bundles should be verified. Code development activities are widespread, and the models and correlations involved in these codes are numerous in comparison to the fuel safety criteria discussed above. Therefore, the Task Force did not attempt to identify all development programmes and the many models and correlations involved. Several types of computer codes that are used in safety analysis are sensitive to fuel-related parameters. In the previous sections of the report, the need for further code development and verification has been stated on many occasions; new design elements, such as different cladding materials, higher burnup, and the use of MOX fuels, can affect the performance of these codes. Some of the more important impacts on these codes were mentioned in order to assess if the codes might need further verification or even modification. Examples were given rather than systematically covering all possible models and effects; this should still be sufficient to identify analytical areas that need to be addressed in more depth.

To adequately cover modern fuel and core designs, the already mentioned best-estimate methods, along with associated uncertainty analysis, should generally be applied in order to reduce unnecessary conservatism. This implies, however, that such methodologies need to be well validated and verified; thus, experimental tests are to continue to provide the basis for such verification and validation.

7.20   High burnup fuel programmes

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 burnup level up to those currently licensed. However, the Task Force also considers that there is a need for further research to:

- experimentally verify the validity of safety criteria for high burnup, in particular for burnup levels beyond those currently licensed; and

- further develop and benchmark the analytical models used to define and monitor high burnup safety criteria.

A few examples of research programmes were given that contribute to investigating the phenomena and mechanisms of fuel behaviour under transient/accident conditions. These includes Halden Reactor Project in Norway, hot-cell testing at the Argonne National Laboratory in the U.S., research and development work by Belgonucleaire in Belgium, the Cabri test reactor and related programmes in France, and the Nuclear Safety Research Reactor programme in Japan. Continuation of these programmes is desirable.

7.21   Conclusion

It is considered that the current framework of fuel safety criteria remains generally applicable, being largely unaffected by the "new" or modern design elements; the levels (numbers) of the individual safety criteria may, however, change in accordance with the particular fuel and core design features. Some of these levels have already been – or are continuously being – adjusted; level adjustments of several other criteria (RIA, LOCA) also appears 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 (nominal) analysis methods, together with a suitable uncertainty analysis, in all areas of safety analysis.

- Continue to perform experimental verification (selected experiments), for benchmarking of such best-estimate methods and extending the verification basis for safety criteria (the amount of testing may be reduced as methods quality advances).

- Review, and adjust where necessary, safety criteria levels based on the above methods and test data; define (quantify) necessary margin to safety limits.

The Task Force considers research programmes such as the Halden, the ANL, the French and Japanese research programmes with many test facilities including the CABRI and NSRR test loops, necessary to support the industry developments as these will contribute to a more detailed and realistic representation of LWR accident scenarios.

8. ACKNOWLEDGEMENT OF TASK FORCE PARTICIPANTS AND REPORT CONTENT

This report summarises the various, sometimes very comprehensive and detailed, contributions from the various experts that participated in the TFFSC. These contributions included reviews in a variety of different areas, according to the expertise of the individual experts.

In various instances, the detailed results reported here correspond to a majority opinion and not necessarily to the opinion of every single expert member of the group. However, the overall conclusions and recommendations are fully supported by all TF participants.

Furthermore, the report content does not necessarily embody the opinion of the organisations which the individual group members represent.

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9. GLOSSARY

Acronym/abbreviation	Meaning
AOA	Axial Offset Anomaly
AOO	Anticipated Operational Occurrence ("normal" transient)
BOL	Beginning of Life
BWR	Boiling Water Reactor
CILC	Crud Induced Localised Corrosion
CPR	Critical Power Ratio
DBA	Design Basis Accident
DNB(R)	Departure from Nucleate Boiling (Ratio)
ECCS	Emergency Core Cooling System
EOL	End of Life
FSAR	Final Safety Analysis Report
IRI	Incomplete Rod Insertion
LHGR	Linear Heat Generation Rate
LOCA	Loss Of Coolant Accident
LWR	Light Water Reactor (BWR, PWR)
MOX	Mixed (i.e. U and Pu) OXide fuel
NPP (NPS)	Nuclear Power Plant (Station)
PCI	Pellet Clad Interaction (ref. stress corrosion cracking)
PCMI	Pellet Cladding Mechanical Interaction
PCT	Peak Cladding Temperature
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assembly
RCS	Reactor Coolant System
RIA	Reactivity Initiated Accident
SDM	ShutDown Margin
TF(FSC)	Task Force (on Fuel Safety Criteria)

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