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Analysis of differences in fuel safety criteria for WWER and western PWR nuclear power plants

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FOREWORD

In 2001 the OECD issued a report of the NEA/CSNI (Committee on the Safety of Nuclear Installations) Task Force on the existing safety criteria for reactor fuel for western LWR nuclear power plants (both for PWRs and BWRs) under new design elements. Likewise in 2001, the IAEA released a report by a Working Group on the existing safety criteria for reactor fuel for WWER nuclear power plants under new design requirements. However, it was found that it was not possible to compare the two sets of criteria on the basis upon which they had been established. Therefore, the IAEA initiated an assessment of the common features and differences in fuel safety criteria between plants of eastern and western design, focusing on western PWRs and eastern WWER reactors.

Between October 2000 and November 2001, the IAEA organized several workshops with representatives from eastern and western European countries in which the current fuel safety related criteria for PWR and WWER reactors were reviewed and compared. The workshops brought together expert representatives from the Russian Federation, from the Ukraine and from western countries that operate PWRs. The first workshop focused on a general overview of the fuel safety criteria in order for all representatives to appreciate the various criteria and their respective bases. The second workshop (which involved one western and one eastern expert) concentrated on addressing and explaining the differences observed, and documenting all these results in preparation for a panel discussion. This panel discussion took place during the third workshop, where the previously obtained results were reviewed in detail and final recommendations were made.

This report documents the findings of the workshops. It highlights the common features and differences between PWR and WWER fuel, and may serve as a general basis for the safety evaluation of these fuels. Therefore, it will be very beneficial for licensing activities for PWR and WWER plants, as it focuses on the issues of importance for the review of fuel safety cases. This report makes frequent reference to three reports which constitute the background for the workshops, two by the IAEA and one by the NEA/CSNI — these three reports have been included on a CD ROM that complements this report.

The workshops, organized under the IAEA regional Technical Co-operation Project on Licensinf Fuel and Fuel Modelling Codes for WWER Reactors, were chaired by F. Pazdera. The IAEA acknowledges W. van Doesburg for his efforts in the preparation of this publication, and E. Androssenko for her invaluable assistance with interpretation in the technical discussions. The IAEA officers responsible for this publication were J. Hoehn and F. Niehaus Division of Nuclear Installation Safety.

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1. INTRODUCTION

1.1. BACKGROUND

The OECD/CSNI/PWG2 Task Force on Fuel Safety Criteria (TFFSC) was given the mandate to technically review existing fuel safety criteria for western LWR (PWR and BWR) reactors. This became necessary because of the ‘new design’ elements (new fuel and core design, cladding materials, manufacturing processes, high burnup, mixed oxide fuel (MOX), etc.) introduced by the industry. The results from this task force were published in July 2000 [1].

In parallel, an IAEA Working Party made a similar assessment for WWER reactor fuel, the results of which are provided in a CD-ROM as a complement to this report.

The IAEA considered it necessary to initiate an appropriate assessment of the common features and differences in fuel safety related criteria between east and west, focusing on western PWR and eastern WWER reactors.

Several workshops were organized with expert representatives from the Russian Federation/Ukraine and from western countries, with the aim to review all eastern and western fuel safety related criteria and to assess the differences between them, if any. Also the rational for any differences found were to be addressed.

The first workshop focused on a review of the fuel safety criteria in order for all representatives to appreciate the various criteria and their respective basis. The second workshop (with a few experts only) concentrated on addressing and explaining the differences observed, determining whether or not any effort (analytical, experimental) should be recommended to resolve them, and documenting all these results in preparation for a panel discussion. This discussion took place during the third workshop, where previously obtained results were reviewed in detail and final recommendations were made. At the end of this workshop a final draft of the report was agreed upon among the participants.

1.2. OBJECTIVE

This report is the result of the above three workshops. It summarizes the fuel safety criteria for western PWR and eastern WWER reactors as documented in the respective reports and compares the equivalent criteria, their basis, and the common features and differences observed.

The report has the objective to make the safety level of fuel design and operation more visible for both east and west, and:

- the report helps to focus on the issues of importance for fuel safety case review,
- the report gives a broadening of insight in failure mechanisms in support of FSC review for new fuel designs,
- it offers a better understanding of FSC and their correct interpretation to avoid unexpected fuel failures or improper behaviour in incidents with negative impact on the nuclear industry as a whole,
- vendors, utilities and regulatory bodies may better understand each other in designing and licensing mixed cores (some lead assemblies are already tested in WWER reactors),
- the report helps to better and more widely understand the R&D needs in connection with new fuel designs, reload cores design and improvements in operational practice with the goal to stimulate co-operation to improve the effectiveness of R&D.

Wherever appropriate and possible, recommendations are given on further action (s).

1.3. SCOPE AND TERMINOLOGY

Generally, the safety of nuclear installations is assured through a deterministic approach, based on the principles of defence-in-depth, and through probabilistic safety assessments. Regarding NPPs, the international agreement on the more detailed approach to be followed is contained in the Safety Guide on Safety Assessment and Verification for Nuclear Power Plants [2]. The complementary nature of the deterministic and probabilistic approaches is also described in this guide.

Two different types of fuel safety related criteria exist:

(1) requirements (general, qualitative), and
(2) limits (quantitative)

Basically, all fuel safety related criteria are derived from the requirements as per the atomic law in the various countries, where dose rate limits are specified to limit the effects of radiation on the general public. Usually there are various different dose rate limits specified, the level of which depends on the likelihood (frequency) of the plant reaching a condition in which radiation may be released. Such plant conditions (Conditions I, II, III and IV), categorized according to frequency of occurrence, are defined in e.g. Appendix A of ANSI/ANS-57.5.-1996 [3]. These conditions correspond to the three categories as defined in the above mentioned IAEA safety guide [2]. Category 3 includes accident conditions III and IV.

These dose rate limits are logically translated into qualitative fuel failure requirements: “no” fuel failures for Condition I and II events, a small (limited) number of localized fuel failures for Condition III events, whereas for Condition IV (Design Basis Accidents) a larger number of fuel failures is allowed without endangering core coolability/control rod insertability. These qualitative requirements are again translated into quantitative safety criteria (=limits), which ultimately have to be met for fuel and core design and plant operation.

Figure 1 illustrates this process of criteria (requirements/limits) definition.

It is convenient to divide fuel safety related criteria into three categories:

(A) safety criteria
(B) operational criteria
(C) design criteria.

The (A) category includes all safety criteria imposed by the regulator, covering the licensing and design basis of the reactor. These criteria, most of which pertain to transient and accident conditions, have to be met at all times.

The (B) category includes operational criteria, some of which are derived from Category A, and others that are added for better coverage of normal operation and more frequent operational occurrences. These limits, many of which are specific to the fuel design and are provided by the fuel vendor as part of the licensing basis, are also mostly approved by the regulator (after appropriate safety evaluation.)

The (C) category includes design limits that for the most part have not been approved by the regulator. They are part of the design basis for the fuel with the aim to be able to meet the (A) or (B) category criteria.

These three categories are represented in Figure 2. Tables I–III show the criteria in each of the three categories.

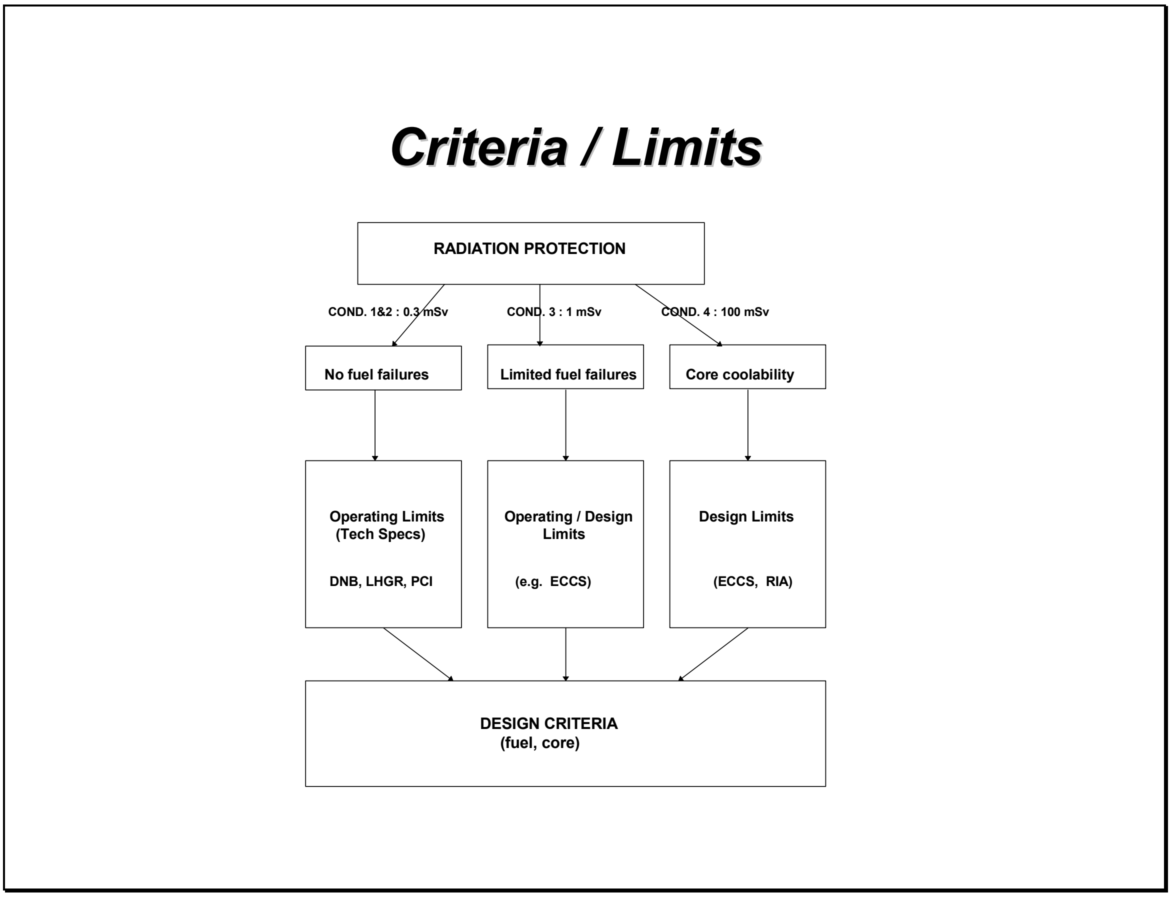

FIG. 1. Process of criteria (requirements/limits) definition.

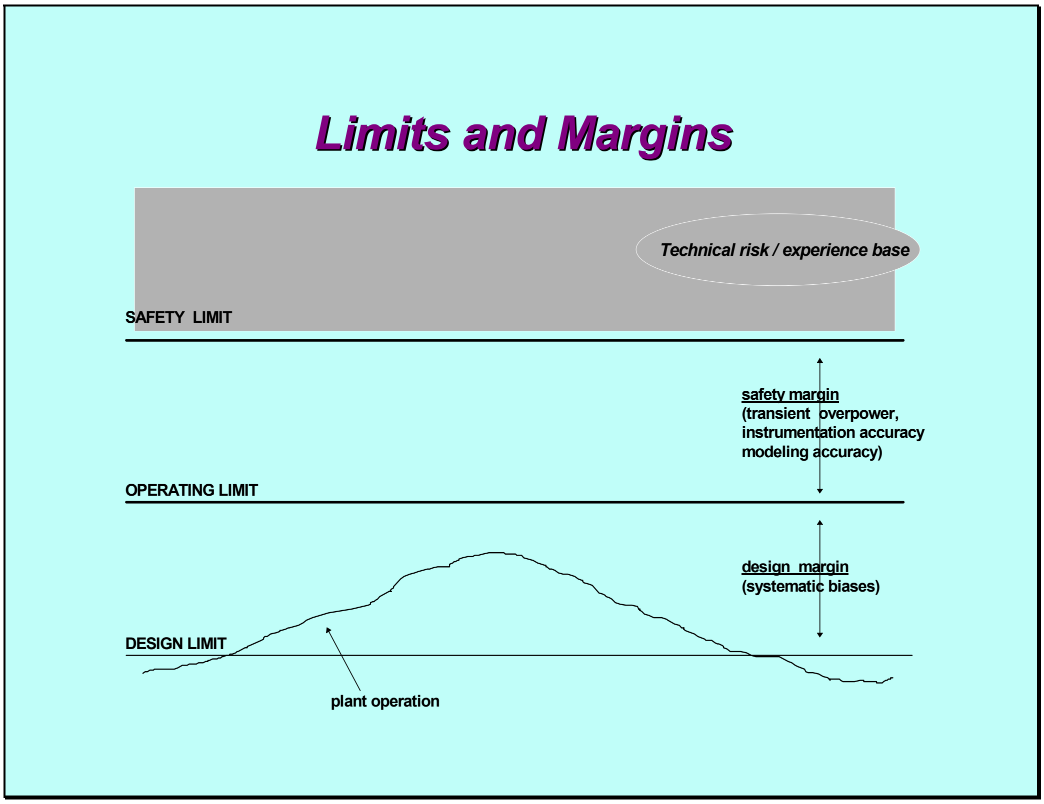

FIG. 2. Categories of fuel safety related criteria.

TABLE I. SAFETY CRITERIA

	Criterion	OECD ref (Table 1)	IAEA ref (Table 2.1)
1	DNB safety limit	A	2.1
2	Reactivity coefficients	B	3.2
3	Shutdown margin	C	3.3
4	Enrichment	D	
5	Internal gas pressure	I	1.9
6	PCMI	J	
7	RIA fragmentation	L	1.17, 1.18
8	Non-LOCA runaway oxidation	N	1.14
9	LOCA-PCT	O	1.14
10	LOCA-Oxidation	O	1.15
11	LOCA-H release	Ch. 3.14 (O)	1.16
12	LOCA-long term cooling	Ch. 3.14 (O)	
13	Seismic loads	P	3.6
14	Hold-down force	Q	2.4
15	Criticality		

TABLE II. OPERATIONAL CRITERIA

	Criterion	OECD ref (Table 1)	IAEA ref (Table 2.1)
1	DNB operating limit	A	3.4, 3.5 (indirect)
2	LHGR limit	Ch. 3.9 (J)	1.4,1.8, 1.10
3	PCI	K	1.1, 1.11
4	Coolant activity	R	3.1
5	Gap activity	S	
6	Source term	T	
7	Control rod drop time		2.2
8	RIA fuel failure limit	M	

TABLE III. DESIGN CRITERIA

	Criterion	OECD ref (Table 1)	IAEA ref (Table 2.1)
1	Crud deposition	E	
2	Stress / strain / fatigue	F	1.2, 1.4
3	Oxidation	G	1.12, 1.13
4	Hydride concentration	H	1.12
5	Transport loads		2.3
6	Fretting wear		2.5
7	Clad diameter increase		1.7
8	Cladding elongation		1.6
9	Radial peaking factor		3.4
10	3D peaking factor		3.5
11	Cladding stability		1.3

1.4. STRUCTURE

In Section 3 all these criteria will be reviewed. In this section, first a description will be given of the criteria as defined in the west (for PWRs) and east (for WWERs), then the differences between these definitions will be summarized; lastly, conclusions and recommendations will be given as appropriate.

Section 2 includes a description of WWER and PWR fuel design characteristics as made available by some of the fuel vendor representatives that participated in the workshops. *Please note that the design descriptions presented here are intended as examples and do not imply any judgment whatsoever towards a particular fuel design or vendor.*

2. FUEL DESIGN DESCRIPTION

2.1. WWER FUEL ASSEMBLIES: WWER-440

The Working Assembly (WA) shown below (Fig. 2) consists of the fuel rod bundle, cap, tailpiece and jacketed tube. The fuel rods within the bundle are arranged in a triangle and are connected by the «honeycomb-type» spacing grids being mechanically mounted on the central tube and by the lower support grid mounted on the tailpiece. The lower support grid is welded to the tailpiece intended to install the WA within the reactor basket bottom. The WA tailpiece is installed into the basket bottom seat resting by its ball surface upon the seat conic part.

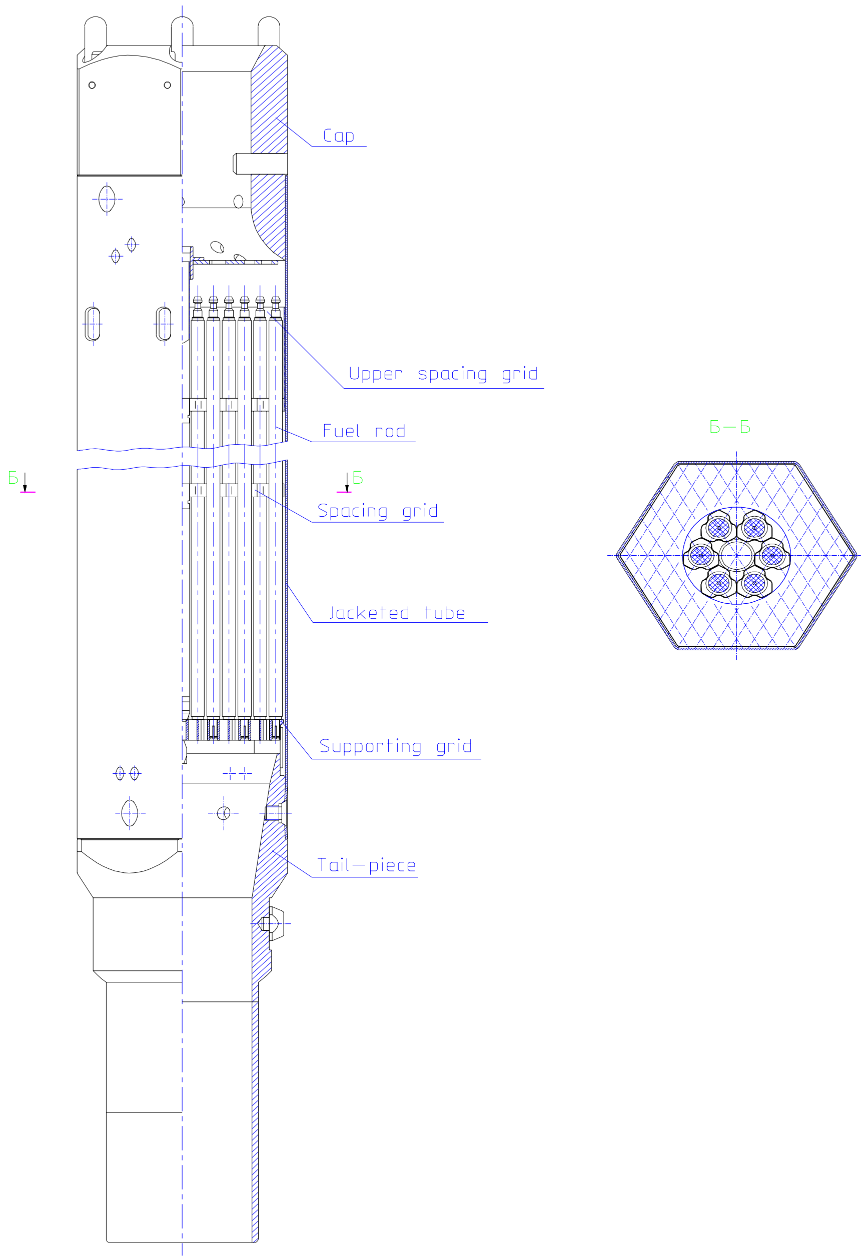

FIG. 2. The working assembly for a WWER-440.

The WA cap is rigidly attached (over the hexahedral surface) to the jacketed tube. In the WA cap there are two fingers for the transport grip of the fuel handling machine and six spring-loaded stops used to prevent the working assembly from floating and to compensate for thermal expansions and technological tolerances of the reactor internals. The bottom end of the cap is attached to the protective grid. The fuel rods are fixed in the support grid by the pin wire. To compensate for thermal expansion and radiation growth of the fuel rod bundle with respect to the support grid, the WA ensures possible elongation of the fuel rods for at least 25 mm.

In the lower and upper parts of the WA jacket in the regions of the cap and tailpiece there are holes (two on each flat) intended for radial off-loading of the jacketed tube from coolant pressure differential.

The reactor control assembly consists of the fuel assembly (FA) (Fig. 3) and the absorber (Fig. 4) connected between each other by the intermediate mast. The fuel rods are triangle-arrayed in the fuel assembly. The absorber presents itself a welded structure made of stainless steel with the hexahedral inserts of boron steel located inside.

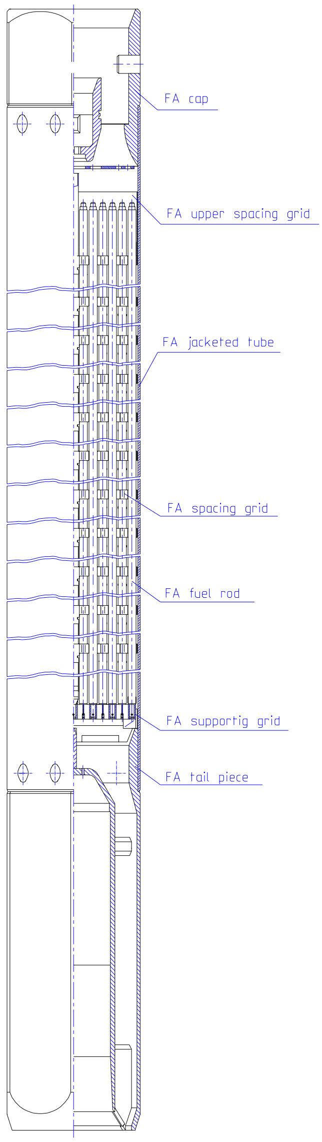

FIG. 3. Fuel assembly.

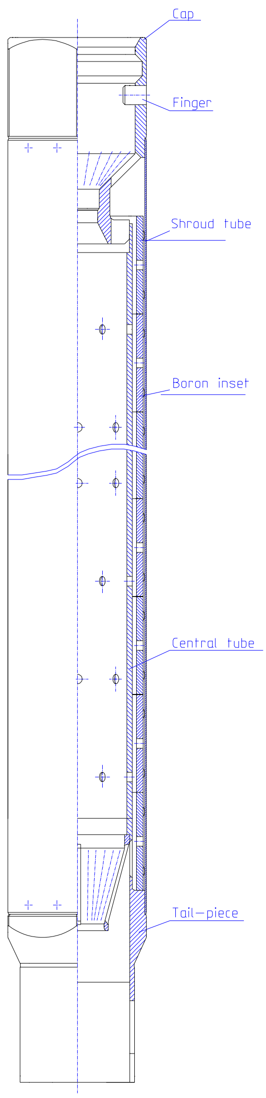

FIG. 4. Absorber.

The fuel assembly design is identical to the WA design except for the following features: special tailpiece, FA jacket has no perforations and the FA cap has no spring-loaded stops.

The FA cap is equipped with the grip device of bayonet type with a seat for a triangle catch used to provide engagement with the intermediate mast. The intermediate mast passing through the absorber centre for its full height is engaged with the grip bayonet device located in the FA cap; in this case the fixing triangle rod of the intermediate mast enters the FA cap seat thereby avoiding rotation and subsequent disengagement of the FA with the intermediate mast.

In the FA tailpiece there is a damper device (thimble) used to provide assembly damping during its movement (drop) by gravity under the accident condition related to a break of the intermediate mast. The damping principle consists in coolant (water) throttling through the gaps formed between the rod located in the reactor cavity bottom and the FA tailpiece thimble at a moment when the assembly drops and the thimble seats upon the rod.

In addition, water throttling occurs through two holes or more of a 3 mm diameter located in the FA tailpiece thimble bottom.

2.2. WWER FUEL ASSEMBLIES: WWER-1000

The WWER-1000 fuel assembly (Fig. 5) consists of the following components:

- cap
- bundle of fuel rods
- tail-piece.

The cap consists of the following parts: upper shell, supporting plate, spring unit, lower shell, collets and components connecting the assembly units of the cap in the common structure.

The bundle of fuel rods is assembled of 312 fuel rods in a frame consisting of 15 spacing grids, a central tube, 18 guiding channels, and the lower supporting grid.

The fuel rod consists of the following parts: upper plug, cladding, lower plug, fuel core made of pellets UO2 and a lock. The material of the fuel rod cladding and plugs is alloy Zr + 1 % Nb. The spacing grid provides support in pairs between a cell - fuel rod, and a cell - FA guiding channel.

The FA tail-piece is a supporting welded construction - the body with the system of ribs. The ribs, welded to the shell, form the supporting grid, containing two parallel ribs crossing the third rib in a transverse direction. The ribs are enclosed into a hexahedron with transition to the cylinder. The inside of the lower part of the tail-piece body is made in the form of a diffuser, and from the outside has a supporting spherical part with transition to the cylinder. The lock is installed on the cylinder. The bundle of fuel rods through the lower supporting grid rests on the parallel ribs of the tail-piece.

2.3 PWR FUEL ASSEMBLY DESIGN EXAMPLE: FRAMATOME-ANP

The assembly (Fig. 6) with a 17×17 array consists of 264 fuel rods, 24 control rod guide tubes, one instrumentation tube, a bottom end piece, a top end piece and eight axially arranged spacer grids in the case of an active core height of 12 ft. Optionally, the fuel assemblies are equipped with a debris filter, and for the increase of thermal-hydraulic margins, with three intermediate flow mixers (IFMs).

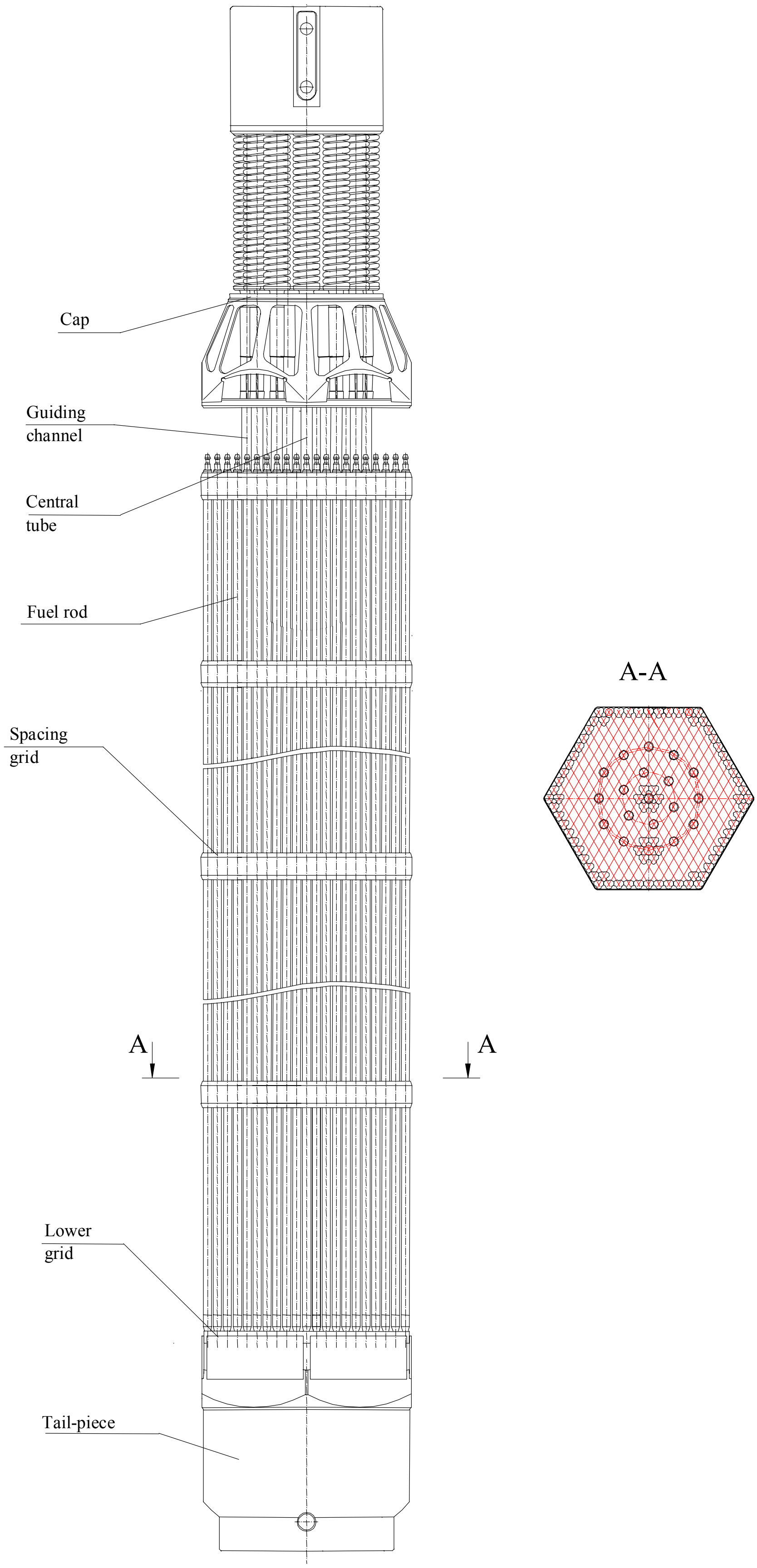

FIG. 5. WWER-1000 assembly.

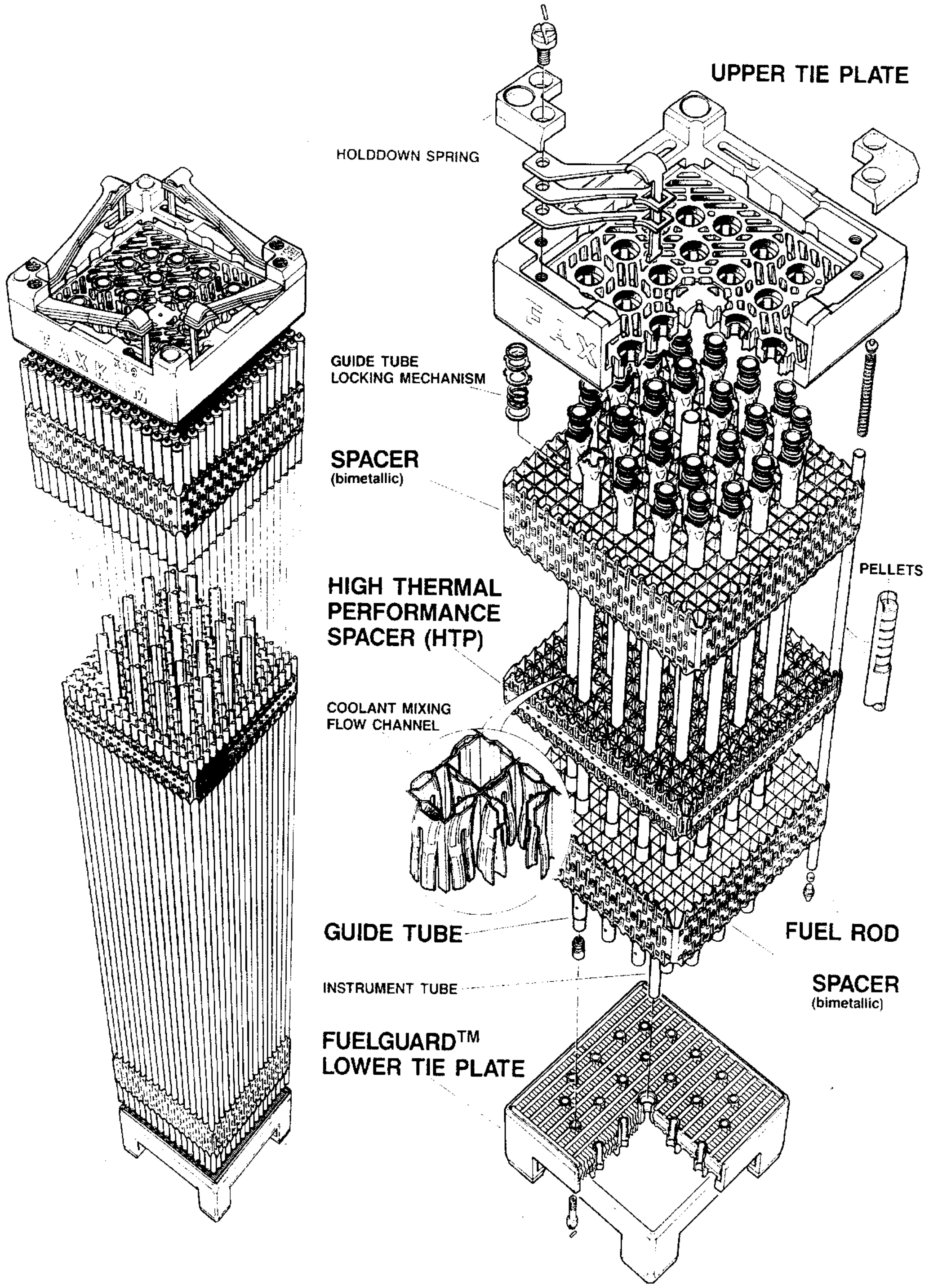

FIG. 6. Fuel assembly design for a PWR.

The detailed features of this fuel assembly design are:

- Corrosion resistant Duplex cladding, capable of high burnup without loss of rod integrity. This cladding tube is proposed for rod burnups over 55 MWd/kg (U).
- Options are fuel rods with natural uranium axial blankets, which increase neutron economy by an enrichment saving of about 0.06 $^{w}/_{o}$ U^{235}.
- All-Zircaloy High Thermal Performance (HTP) spacers with integrated curved flow channels, utilized for all but the bottom spacer position, increase the coolant mixing and enhance the DNB performance (see detailed Fig. 7 below).
- The Inconel HTP spacer at the lowermost position provides improved fuel rod support throughout life at the bottom of the fuel rod region and minimizes the possibility of flow-induced fretting failures.
- The debris-resistant FUELGUARD bottom end piece with curved blades provides almost complete protection against debris-induced fretting failures.
- The readily removable top end piece allows quick and easy fuel assembly repair, reconstitution or surveillance from the topside.
- The dismountable FUELGUARD bottom end piece allows fuel assembly repair, reconstitution or surveillance also from the bottom side, should the need arise.
- Gadolinium burnable neutron absorber with optimized gadolinium absorber length provides operating and fuel cycle design flexibility. When incorporated in the UO_2 pellets of selected rods, Gd avoids the cost of separate encapsulation required for B_4C or borosilicate glass and its residual parasitic absorption. The integration of gadolinium-bearing fuel rods minimizes radial neutron leakage which, together with the reduced residual reactivity penalty, would decrease batch average enrichments.

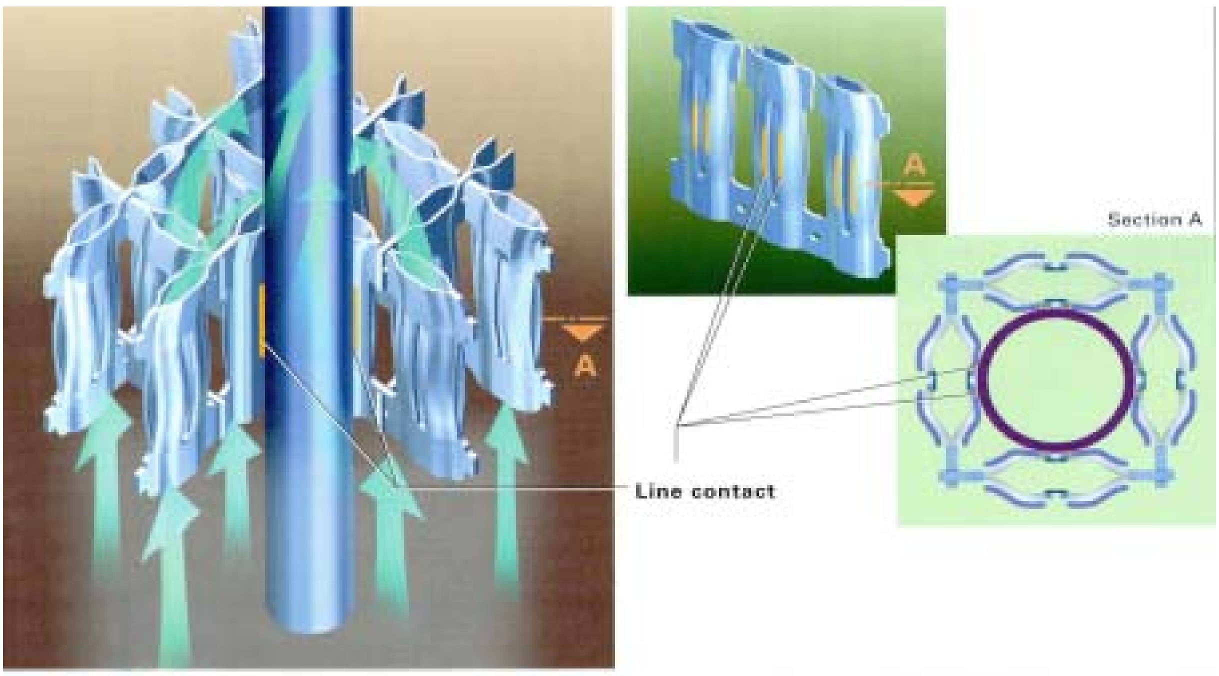

FIG. 7. High thermal performance (HTP) spacers.

3. DESCRIPTION OF THE CRITERIA APPLIED IN THE EAST AND IN THE WEST

This section contains a description of all fuel safety related criteria as documented in Ref. [1]. The background (rationale) for any differences will be given, and recommendations to resolve them will be mentioned, as appropriate.

The cross-reference to these criteria is given in Appendix I.

3.1. ITEM A-1 DNB SAFETY LIMIT

3.1.1. Description

This safety criterion which covers the cladding integrity relates to the so-called 'boiling crisis', i.e. departure from nucleate boiling (DNB) due to cladding overheating. The rapid and substantial rise of cladding temperature and subsequent fast oxidation (or even melting) of the cladding will cause the cladding to fail.

For western reactors the relevant parameter is the heat flux; the critical heat flux (CHF) is where the boiling crisis occurs. Critical heat flux correlations (e.g. Doroschuk, W-3) relate the critical heat flux with bundle parameters such as pressure, flow, axial peaking, etc. and are based on data from full-scale fuel bundle (array) testing.

The limiting DNB-ratio (or DNBR, critical to actual heat flux) is a safety limit statistically defined such that with 95% confidence and 95% probability, fuel rods will not experience CHF during an Anticipated Operational Occurrence (AOO, or Condition II event). This limit is also used to indicate fuel failure for some postulated accidents (Condition III and IV events) for evaluating off-site dose rates. Typically, DNBR safety limits are around 1.15. For more detailed information on the definition of DNB(R), see Ref. 4.

For WWERs, with 95% confidence and 95% probability DNB shall not occur for maximum-powered fuel rods under steady-state and AOO conditions. Again, critical heat flux correlations (e.g. Bezrukov) are used, based on thermal-hydraulic testing.

3.1.2. Differences: Description and rationale

There is no difference in the (statistical) approach for defining the criterion. For PWRs, the criterion is defined for transients, whereas for WWERs the criterion also includes steady-state operation. The WWER criterion is, however, not likely to be less conservative because it applies to the highest power rods only.

As the definition of the operating limit DNBR / radial peaking factor (see Item B-1) includes a substantial amount of margin to the safety limit, the probability for any fuel rod to experience DNB is likely to be similar for both PWRs and WWERs.

Differences exist between the various CHF correlations; this is clearly due to the different fuel designs, as the correlations are based on fuel-specific experimental data.

3.1.3. Conclusions

There is no principal difference in basis between east and west for defining DNB safety limits.

3.1.4. Recommendations

Mixed cores may contain various assemblies each with a different CHF correlation for calculating the distance to DNB; the DNB safety limit however is a core limit, and is therefore established by analysing the whole core. Thus the DNB safety limit could become a cycle specific limit.

3.2. ITEM A-2 REACTIVITY COEFFICIENTS

3.2.1. Description

Reactivity coefficients are intrinsic to the reactor and fuel design and guarantee an overall negative reactivity feedback from the reactor core during transients / accidents.

For western reactors, the general criterion is that the total of all reactivity coefficients be negative when the reactor is critical. An individual reactivity coefficient may be positive, however the effect of such positive feedback must be inconsequential.

For WWERs in the Russian Federation, each reactivity coefficient must be negative under all power conditions. This Russian requirement is also accepted in the Ukraine and Bulgaria for WWER-1000 reactors. In some countries which operate WWER plants the requirement is identical to that for western reactors, with some additional licensing requirements; for example, in Finland, the isothermal temperature coefficient must be negative under all conditions.

3.2.2. Differences: Description and rationale

In the west only the sum of reactivity coefficients must be negative. The Russian requirement was imposed after the Chernobyl accident.

3.2.3. Conclusions

The criteria for WWERs appear to be more conservative in certain countries.

3.2.4. Recommendations

The Russian and Ukrainian approach is limiting for plant operation; a positive moderator coefficient needs to be considered at certain (low) power conditions.

3.3. ITEM A-3 SDM

3.3.1. Description

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

For western reactors, the *control rod* shutdown margin (SDM) is defined as the margin to criticality ($k_{eff} = 1$) in the situation with all control rods inserted, but with the most effective control rod withdrawn. SDM needs to be sufficient for achieving hot zero power using control rods only. Control rod SDM limits (typically around 0.5% $\Delta k/k$) are mostly established including the assumed envelope of uncertainties in the determination of k_{eff} and control rod manufacturing tolerances.

The *boron* SDM is the margin to criticality ($k_{eff}=1$) for the situation in which the emergency boron injection system is activated; the increase of boron concentration (usually a 2000 to 2500 ppm boron concentration is required) shall be sufficient to achieve cold shutdown without control rod movement. Boron SDM limits are established similar to control rod SDM limits, i.e. based on calculational and system uncertainties.

Both these SDM limits are verified analytically as part of the safety analysis and reload licensing process; the analysis usually assumes a Xenon-free core, for conservatism. The limits are verified at least at reactor startup after refuelling.

For WWERs, criteria are defined in a similar way including the condition that the most effective control rod is stuck out.

The requirement on boron concentration in the Russian Federation is 16 g/kg boric acid concentration which is equivalent to about 2700 ppm boron concentration: this compares to western concentrations.

3.3.2. Differences: Description and rationale

For new generation NPPs in the Russian Federation, there is an additional requirement that no recriticality shall occur down to 100°C with control rods only (i.e. without boron injection, which normally initiates around 280°C).

3.3.3. Conclusions

Same criteria, with one additional requirement in the Russian Federation for new generation NPPs.

3.3.4. Recommendations

None.

3.4. ITEM A-4 ENRICHMENT

3.4.1. Description

An administrative enrichment criterion is imposed in consideration of possible criticality during fuel fabrication, handling and transport.

For western reactors, an enrichment limit of 5 wt% U-235 is in effect. This limit is largely based on a historic decision and on the validation / benchmarking of criticality safety codes and associated cross section libraries for LWR fuel.

3.4.2. Differences: Description and rationale

This issue was omitted in the IAEA report as not operationally related. However, the same limit applies as in the west (i.e. 5 wt%), administratively for WWER fuel vendors in the Russian Federation.

3.4.3. Conclusions

No differences.

3.4.4. Recommendations

At this point in time, the industry does not appear to have a need for increasing this limit. If in the future such an increase becomes necessary, action(s) required to verify a higher limit for criticality evaluations need to be co-ordinated.

3.5 ITEM A-5 INTERNAL GAS PRESSURE

3.5.1. Description

A safety criterion is applied to prevent clad distension and run-away fission gas release.
For western reactors, two different types of criteria exist:

(1) the rod internal pressure must be less than the nominal reactor coolant system (RCS) pressure,
(2) the instantaneous cladding creep-out rate shall not exceed the instantaneous fuel swelling rate (i.e. the fuel-to-clad gap does not open): this is the so-called "no lift-off" criterion.

3.5.2. Differences: Description and rationale

For WWERs, the first criterion applies as for some western countries. A margin of 10% is used in some WWER operating countries to cover uncertainties in fission gas release (FGR) evaluation methods. In other WWER operating countries these uncertainties are included in a different way, as is the case for PWRs. Sometimes a statistical treatment of uncertainties is performed.

3.5.3. Conclusions

Depending on the country, the first or the second criterion applies in the case of PWRs. For WWERs, only the first, more conservative, criterion applies.

The uncertainties in FGR evaluation methods are treated differently in the various countries.

3.5.4. Recommendations

None.

3.6. ITEM A-6 PCMI

3.6.1. Description

PCMI (pellet–cladding mechanical interaction) refers to the stress due to expansion of the fuel pellet during a transient without the effect of Iodine (no stress-corrosion cracking (SCC): thus, not to be confused with pellet–cladding interaction (PCI), see Item B-3).

For western reactors, the safety criterion is defined to avoid mechanical fracture during this type of transient due to PCMI.

This safety requirement has traditionally not been quantified. Instead, the requirement is considered to be met by applying the existing 1% design limit on elastic + plastic strain, see Item C-2.

Because of the conservative requirement that the stress is below the standard yield strength for Condition II events for WWERs (see Item C-2), there is negligible plastic deformation and thus no need for a strain criterion.

3.6.2. Differences: Description and rationale

The Western 1% strain criterion is to be compared with the more conservative stress criterion for WWERs.

3.6.3. Conclusions

As for western reactors, existing WWER criteria cover the safety requirement as formulated for PWRs. Experiments for WWER fuel show similar behaviour as compared to PWR fuel, with similar resistance to PCMI.

3.6.4. Recommendations

It is considered that the speed of the transient/accident has to be taken into account for a proper definition of PCMI limits. Thus, a review of the PCMI safety criteria appears appropriate.

3.7. ITEM A-7 RIA, FRAGMENTATION

3.7.1. Description

Peak fuel enthalpy criteria are used as limits for reactivity-initiated accidents (RIA), in order to avoid the loss of coolable geometry and the generation of coolant pressure pulses.

For western reactors, an enthalpy limit of 280 cal/g has mainly been used. This limit is based on data from early RIA fragmentation measurements prior to 1974 (e.g. SPERT and TREAT tests in the USA); the value corresponds to the melting of UO_2 which causes fragmentation of the cladding and expulsion of fuel particles which also leads to energetic fuel-coolant interactions that generated pressure pulses.

Later RIA measurements and subsequent analyses sometimes led to a redefinition of the enthalpy limit, see e.g. the discussion in Ref. [5]. Thus, in some countries limit values lower than 280 cal/g are in effect today.

Similarly, for WWERs, a limit of 230 cal/g is in place for avoiding fuel fragmentation. This limit is derived from the 280 cal/g limit as mentioned above; this limit was applied originally, then set down conservatively several years ago based on data from the first RIA experiments with WWER fuel. Recently, experiments have been performed with fuel from 0 to 60 MWd/kg burnup showing no fragmentation in the whole range of experiments with enthalpies up to 265 cal/g, demonstrating that sufficient margin to the 230 cal/g limit is available. Fuel fragmentation was observed on earlier experiments well above 300 cal/g.

3.7.2. Differences: Description and rationale

One extra requirement for WWERs applies: no global fuel melting resulting in PCMI shall occur. This is a general criterion for all design basis accidents (DBA), which is mentioned here, as it appears to be most applicable to a RIA.

3.7.3. Conclusions

No basic differences between eastern and western limits.

3.7.4. Recommendations

For PWRs, criteria are under review supported by R&D programmes especially in terms of burnup dependence. A criterion related to fuel melting is currently being considered (ref. the EPRI Robust Fuel Programme). Also for WWER fuel, more experiments are foreseen; however, the 230 cal/g limit is not expected to decrease.

3.8. ITEM A-8 NON LOCA RUNAWAY OXIDATION

3.8.1. Description

For non-LOCA (loss of coolant accident) transients of brief duration (e.g. the locked rotor/pump seizure accident), the fuel is not seriously damaged due to DNB, but may still fail due to a significantly increased oxidation rate and subsequent loss of ductility.

For western reactors, a criterion is in effect to indicate fuel failure for estimating radiological dose rates to the public, while at the same time assuring that core coolability is still maintained. A 2700°F (1482°C) limit temperature is mostly utilized; this is based on early experimental data on the fuel failure boundary for LOCA type conditions — for an actual LOCA, a lower temperature limit was deemed necessary (see Item A-9) while for non-LOCA fast transients the higher value was considered adequate.

3.8.2. Differences: Description and rationale

No explicit criterion exists for WWER fuel. However, the general requirement for cladding temperature in accident conditions to remain below 1200°C applies (see Item A-9). For WWER-1000 reactors, the locked rotor/pump seizure event leads to maximum cladding temperatures below 700°C and should therefore not be limiting.

3.8.3. Conclusions

The 1482°C (2700°F) criterion pertains to 2- and 3-loop PWRs; the 1200°C limit is the equivalent criterion for 4 and 6-loop WWERs.

3.8.4. Recommendations

From a western point of view, the behaviour of highly burnt fuel under this condition is relatively unknown. It is necessary that the relevance of the above criterion be experimentally confirmed.

3.9. ITEM A-9 LOCA — PCT

3.9.1. Description

During a LOCA, a certain amount of fuel rods may fail and release fission products. However, criteria are defined to limit cladding embrittlement in order to prevent fragmentation and maintain a coolable geometry. To achieve this, emergency core cooling systems (ECCS) must operate to provide sufficient and long-term core cooling.

For western reactors, 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 2200° F (1204°C) and the total oxidation did not exceed 17% of the cladding thickness before oxidation (see Item A-10 below). These embrittlement criteria were established during the early 1970s, and are still widely used.

For WWERs, the same (1200°C) limit is in place; this limit generally applies to each DBA.

3.9.2. Differences: Description and rationale

No difference between criteria values, but are derived from different bases.

Basis for the WWER criterion is avoidance of the self-sustaining steam-zirconium reaction; peak cladding temperature (PCT) <1200° C. Also, no embrittlement leading to fragmentation shall occur (see Item A-10).

For LWRs, the basis is to guard against post quench embrittlement. The value of 1200°C was chosen as the temperature where the ductile to brittle transition temperature is de-coupled from the level of oxidation. This is based on the observation that samples oxidized at >1200°C were more brittle than samples oxidized at <1200°C with similar levels of oxidation.

3.9.3. Conclusions

No basic differences between east and west.

3.9.4. Recommendations

None.

3.10. ITEM A-10 LOCA — OXIDATION

3.10.1. Description

During a LOCA, a certain amount of fuel rods may fail and release fission products. However, criteria are defined to limit cladding embrittlement in order to prevent fragmentation and maintain a coolable geometry.

For western reactors, as explained under Item A-9, the total oxidation of the cladding during a LOCA must not exceed 17% of the cladding thickness before oxidation. In some countries an even lower limit value has been imposed (e.g. 15% in Japan).

At the time this limit was established, experimental validation included tests with zero or low burnup fuel. Today’s fuel operation at high burnup exhibits typical steady-state oxidation levels of up to 100 microns and hydrogen concentrations up to 500 ppm at the time of fuel discharge (EOL). Hence the 17% criterion is now interpreted as ‘total’ oxidation level, i.e. including both pre-transient and transient oxidation. As the oxidation process at LOCA temperatures differs from that at normal operating temperatures, this interpretation may be considered as being very conservative.

For WWERs, a 18% criterion was implemented in 1977 based on the western 17% limit described above, taking into account the (relatively small) differences in oxidation kinetics between Zircaloy and Zr-Nb materials. Afterwards, experiments were performed (in the Russian Federation as well as other eastern countries) with fresh and irradiated fuel (up to 45–50 MWd/kg) to verify this criterion, with satisfactory results. This criterion is interpreted as total oxidation, i.e. including pre-transient oxidation. Because today’s fuel operation of WWER fuel exhibits almost negligible steady-state oxidation levels and hydrogen concentrations even at high burnup, this interpretation does not pose a real problem.

3.10.2. Differences: Description and rationale

Limit values are similar, however they are derived on a different basis. For WWERs, the remaining ductility must be sufficient in order to avoid fragmentation; this has been experimentally confirmed. Confirmatory experiments have demonstrated that, with the 18% equivalent cladding reacted (ECR) criterion, the required ductility is maintained.

The physical phenomenon of embrittlement of Zircaloy is similar as for Zr-Nb alloys for oxidation in the beta phase. However, if part of the oxidation goes through the alpha-beta transition phase, Zr-Nb alloys display different oxidation characteristics leading to higher embrittlement.

Cladding plasticity will be lower, however still sufficient to prevent cladding fragmentation from thermal shock; this has been confirmed by Russian experiments with both fresh and irradiated cladding (up to about 50 MWd/kg). In reality, values evaluated for licensing of WWER-1000 fuel are about 5–6% ECR, which implies that a considerable margin exists.

For western PWR fuel, values between 1 and 8% are typically evaluated.

3.10.3. Conclusions

The limits as currently defined appear to be adequate; enough margin exists for actual fuel/core designs.

3.10.4. Recommendations

Verification experiments need to be performed to confirm (or possibly adjust) the limits for new requirements/new design elements. Pre-transient oxidation is important for Zircaloy materials and almost negligible for Zr-Nb materials. Hence, especially for Zircaloy materials, the effect of pre-transient oxidation on the level of (and on the compliance with) the limit needs to be resolved.

3.11. ITEM A-11 LOCA — HYDROGEN RELEASE

3.11.1. Description

A criterion is in effect to limit the total hydrogen production by oxidation of the cladding during a LOCA. This assures containment integrity (possible explosive gas mixture) rather than protecting against cladding embrittlement.

For western reactors, the LOCA limit on the amount of hydrogen generated from the chemical reaction between cladding and water/steam is generally 1% of the hypothetical amount that would be generated if all of the cladding were to react.

Also for WWERs the 1% criterion is defined.

3.11.2. Differences: Description and rationale

No difference.

3.11.3. Conclusions

See above.

3.11.4. Recommendations

None.

3.12. ITEM A-12 LOCA — LONG TERM COOLING

3.12.1. Description

For western reactors, in the event of a LOCA (see also Item A-9), emergency core cooling systems (ECCS) must operate to provide sufficient and long-term (post-transient) core cooling.

For WWERs, the regulatory framework includes the same criterion.

This criterion is not mentioned in the OECD/IAEA reports, as it pertains to ECCS-equipment performance capability.

3.12.2. Differences: Description and rationale

No differences.

3.12.3. Conclusions

See above.

3.12.4. Recommendations

Additional verification of long-term coolability is needed for high burnup fuel.

3.13. ITEM A-13 SEISMIC LOADS

3.13.1. Description

During a seismic event the fuel assemblies are subjected to dynamic, structural loads which could cause core component deformation that reduce coolant flow and/or fuel fragmentation, thereby endangering coolable geometry and degrading ECCS performance.

For western reactors, safety criteria require that core coolability and control rod insertion can be assured under the combined seismic and LOCA loads. These general criteria are usually quantified by design requirements for core components; see e.g. Item C-2 for requirements pertaining to the fuel rod cladding. Verification is performed both analytically and by experiments.

Identical criteria apply to WWER reactors.

3.13.2. Differences: Description and rationale

No differences in approach.

3.13.3. Conclusions

See above.

3.13.4. Recommendations

None.

3.14. ITEM A-14 HOLD-DOWN FORCE

3.14.1. Description

A safety criterion is defined to limit hydraulic vertical lift-off forces, in order to prevent a displacement (unseating) of the lower fuel assembly tieplate from the fuel support structure.

For western reactors, fuel assemblies are equipped with springs in the top piece that must provide sufficient hold-down force to prevent fuel assembly lift-off due to hydraulic loads during normal operation and anticipated operational occurrences (Condition I and II events). The required hold-down force is determined by the hydraulic force on the fuel assembly (which depends on the flow rate and the pressure loss coefficient), the buoyancy force and the fuel assembly weight. Verification is made analytically, using conservative numbers for the flow rate and the relevant tolerances/uncertainties, at beginning of life (BOL) and EOL.

Also for WWERs no lift-off is allowed for Condition I and II events.

3.14.2. Differences: Description and rationale

No differences.

3.14.3. Conclusions

See above.

3.14.4. Recommendations

None.

3.15. ITEM A-15 CRITICALITY

3.15.1. Description

This criterion was not included in the OECD / IAEA reports; it is included here since it is an important criterion, which, indirectly, pertains to the fuel and also connects to the enrichment criterion, see Item A-4.

For fuel manufacturing, transport and storage of fuel material the configuration of such material must be such that criticality does not occur. For western reactors, generally the IAEA criticality safety standard of $K_{eff} < 0.95$, i.e. a 5% margin to criticality, is in effect as safety criterion.

The verification of this safety criterion is performed analytically; usually analysis methods uncertainties and dimension tolerances are evaluated separately, and are applied in addition to the 5% margin.

For WWERs, the same criterion applies.

3.15.2. Differences: Description and rationale

No differences between western and eastern criteria. Some countries take credit for the lower reactivity of burnt fuel.

3.15.3. Conclusion and Recommendations

It is expected that more countries will want to take credit for fuel burnup for storage criticality; in these cases, a proper safety evaluation must take place.

3.16. ITEM B-1 DNB OPERATING LIMIT

3.16.1. Description

For western reactors, the DNB operating limit is derived from the DNB safety limit (see Item A-1 for the definition of DNB) by adding a margin based on the worst possible Condition II event (AOOs). Therefore, this limit that applies to normal operating conditions (NOC) automatically warrants adequate fuel performance during any Condition II event. The DNB operating limit is verified as part of the reload design; it may also be monitored during plant operation. Typical values are 1.30–1.70.

For WWERs, a DNB operating limit is not defined. Instead, a radial peaking factor (see Item C-9) is derived from the safety limit DNB (see Item A-1) for a bounding axial power distribution, which directly applies to reload design and normal operation (Condition I), in order to ensure that the safety limit DNB can be met.

3.16.2. Differences: Description and rationale

For WWERs, the radial peaking factor fulfils the function of the operating limit for PWRs.

3.16.3. Conclusions

Different licensing approach, however the basic requirement is the same. The western licensing approach offers more flexibility for the plant operator.

3.16.4. Recommendations

None.

3.17. ITEM B-2 LHGR LIMIT

3.17.1. Description

For western reactors, fuel specific thermal-mechanical operating limits are expressed as a burnup dependent LHGR (linear heat generation rate, W/cm or kW/ft) curve. Such a limit is defined to bound steady-state operation in a conservative manner, thus also protecting against class II transient thermal and mechanical overpower) for the following phenomena:

- Fuel melting (*Note*: sometimes not calculated explicitly, while considered to be covered by the 1% strain criterion, see Item C-2).
- Rod internal pressure (see Item A-5), fission gas release.
- Stress, strain, fatigue (see Item C-2).
- PCMI stress (see Item A-6).

Basically, such limits are analytically derived by the fuel vendor and validated against experimental data. Traditionally, the derivation includes conservative assumptions on the uncertainty in models, model parameters, manufacturing tolerances, and fuel/core management (i.e. power histories). Modern fuel design methodologies treat these uncertainties in a statistical manner: uncertainties are expressed as distributions of the corresponding parameters, which are varied in a Monte Carlo analysis to produce a ‘best estimate’ value for the limit (instead of an ‘upper bound’) from which the operating limit may then be derived by choosing the appropriate level of confidence.

Also for WWERs, LHGR must be less than an operational limit which is a function of burnup. This guards against all the above phenomena.

3.17.2. Differences: Description and rationale

Identical approach. Usually, the limiting phenomenon is FGR.

3.17.3. Conclusions

See above.

3.17.4. Recommendations

None.

3.18. ITEM B-3 PCI

3.18.1. Description

PCI (pellet–cladding interaction) fuel failures are due to stress corrosion cracking on the inside of the cladding material associated with local power ramping (e.g. reactor startup, manoeuvres or transients). Both the stress (from the power increase) and the corrosion (from e.g. aggressive fission product components) are necessary for bringing about PCI.

During the 1970s many PCI failures were observed in western reactors. First, operating rules to control the phenomenon were developed (so-called PCI operating management recommendations, or PCIOMRs); these restrict the power increase as a function of time and operation at reduced power, and furthermore condition fuel for fast power ramping.

Similar operating rules exist for WWERs. WWER vendor standards allow that the cladding could contain a defect of dimension not exceeding 35 microns. Evaluation of experiments set an incremental power increase, which is burnup dependent, to prevent extension of such a defect for burnup levels >25 MWd/kg. The limiting value at a burnup of 60 MWd/kg is 80 W/cm. This calculated burnup dependent incremental power increase limit corresponds to the maximal stress of 230 MPa in the cladding. These overpower levels are applied to a pre-conditioned power level which is defined as the (average power) sustained over 2 weeks prior to the power increment.

3.18.2. Differences: Description and rationale

No major differences in approach between east and west, however the application (rules and numbers used therein) may vary as this is based on experiments and is fuel design specific. Applications typically cover power increase limitations beyond a specified burnup level.

3.18.3. Conclusions

See above.

3.18.4. Recommendations

In some western countries, PCI resistant fuel (with special cladding) is becoming rather important; such fuel is currently being developed and tested.

3.19. ITEM B-4 COOLANT ACTIVITY

3.19.1. Description

For western reactors, operating limits are defined (usually in the plant Technical Specifications) to limit the concentration of I-131, sometimes also of Cs-137, in the primary coolant to control plant operation after a loss of fuel integrity. This allows continued plant operation with a small, limited number of failed fuel assemblies, according to the plant (systems) design.

Limit values are typically around $1 - 2 * 10^9$ Bq/t; these values are also used for dose rate calculations.

For WWERs, two licensed criteria are defined for leaking fuel rods (leakers) relative to the total number of fuel rods in the core:

(a) 0.2% “gas leakers” or 0.02% leakers with direct contact between fuel and coolant
(b) 1.0% “gas leakers” or 0.1% leakers with direct contact between fuel and coolant

As these criteria cannot be measured directly, they are translated into primary coolant activity limits as follows:

(a) $1.0 * 10^{-3}$ Ci/kg ($3.7 * 10^{10}$ Bq/t) for the sum of Iodine isotopes
(b) $5.0 * 10^{-3}$ Ci/kg ($1.85 * 10^{11}$ Bq/t) for the sum of Iodine isotopes

If criterion (a) can no longer be met, plant operation is possible with the permission of the plant technical supervisor; if criterion (b) is reached, however, the plant must be shut down.

3.19.2. Differences: Description and rationale

Basically the same philosophy exists in west and east. The western I-131 limit of 2.10^9 Bq/t appears equivalent to the I-sum limit of 5.10^{-3} Ci/kg, as there is about a factor of 100 difference between the I-sum and the I-131activity only.

In western reactors, no second 'lower' level of coolant activity is defined for plant operation based on on-line assessment and subsequent technical approval; however negotiations about continued plant operation start well before the technical specifications limit is reached.

In some western countries, also Cs-137 is included.

3.19.3. Conclusions

Licensing approach and limits actually in use are similar. For WWERs, an additional (lower) limit is in place for deciding on further plant operation.

3.19.4. Recommendations

The intention of plant operators and fuel suppliers to have non-leaking cores is to be supported.

3.20. ITEM B-5 GAP ACTIVITY

3.20.1. Description

In western countries, safety analyses in support of source term evaluations (see Item B-6) assume a certain amount of release from the fuel pellet to the gap (e.g. 10% of the noble gas inventory, and up to 6% of halogens and alkali metals). These gap activities are then assumed to be released in case of failed fuel, for calculating off-site dose rates for postulated accidents. These assumptions can vary between different countries, representing various conservative approaches for safety evaluation; they may also be used for design purposes.

The Russian practice with respect to source term - evaluation is described below under Item B-6.

3.20.2. Differences: Description and rationale

For some western countries, the assumptions made are regulatory approved. Some WWER operating countries follow the approach described in Item B-6; others, such as Finland and the Czech Republic, follow the western approach.

3.20.3. Conclusions

See above.

3.20.4. Recommendations

None.

3.21. ITEM B-6 SOURCE TERM

3.21.1. Description

The part of the fission products inventory released into the containment, potentially available for release to the environment during and immediately following an accident, is called the source term. The source term is needed for estimating radiological releases to the public. Basically, there are three possibilities:

(a) evaluate source term analytically;
(b) define source term by law;

(c) inhibit public residence within a specified radius from the nuclear power station, eliminating the need for source term evaluation/definition.

In western countries the source term is usually defined analytically, to estimate radiological releases to the public for Condition III and IV events (Note: in most countries, a severe-accident source term is also defined related to beyond design-basis accidents viz. core melting.) Source terms are sometimes based on measured releases from irradiated fuel, tested under accident conditions; also, gap activity assumptions may be employed (see Item B-5). In addition, assumptions on the effects of retention or enhancement during the course of an accident sequence are made. These various assumptions can vary significantly between countries.

In the Russian Federation the inhibit zone is defined by law. Also other eastern countries followed this approach.

In some countries the source term is evaluated for the most severe DBA to define the emergency planning zone, in order to comply with country specific dose limits for public under accident conditions.

3.21.2. Differences: Description and rationale

In the west the source term definition is country dependent and under review with respect to high burnup and new fuel design, e.g., for MOX. The practice of some WWER operating countries is similar to western countries; some other countries however follow the different Russian approach (zone definition).

3.21.3. Conclusions

See above.

3.21.4. Recommendations

Revision for high burnup and new fuel designs is required.

3.22. ITEM B-7 ROD DROP TIME

3.22.1. Description

A general reactor design criterion is to have an appropriate system of control rods to control core reactivity and to shut down the reactor in a sufficiently fast manner.

For western reactors, the control rod drop time (or scram time) is limited to guarantee a fast reactivity reduction. Drop time operational limits (usually around 2–3 seconds from full-out to full-in, for each individual control rod) are specified in the technical specifications for the plants, and are subject to periodic verification; usually the drop time is verified at least at the time of plant startup after refuelling. Non-compliance entails immediate reactor shutdown.

The same requirement applies to WWER reactors. For WWER-1000 the maximum drop time is 4 sec and 12 sec for WWER-440; the difference is due to the different reactor and reactor scram system designs.

3.22.2. Differences: Description and rationale

Basically no difference in licensing approach; values differ due to the different reactor and reactor scram system designs.

3.22.3. Conclusions

Both western and eastern reactors have had difficulties in meeting this limit, due to excessive bow in the guide thimbles and/or the control rods. See the IAEA report as well as the OECD report, Section 5.5.

3.22.4. Recommendations

The problem of incomplete rod insertion is considered important; for WWERs this issue has largely been resolved, for PWRs a final resolution is still being pursued.

3.23. ITEM B-8 RIA FUEL FAILURE LIMIT

3.23.1. Description

For western reactors, in case of 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 defined as a maximum radially averaged fuel enthalpy increase of e.g. 170 cal/g for BWRs and as a DNB criterion for PWRs. However, based on some of the RIA experiments at the CABRI and NSRR test facilities during the 1990s, where PWR fuel rods at a burnup of approx. 50 MWd/kg or higher failed at rather low enthalpy values, it is not clear if these limits are still appropriate. Various alternative limits of fuel enthalpy as a function of burnup have been proposed, based either on direct experimental data renditions or on relevant parameters such as cladding oxide thickness.

For some WWER operating countries, no firm RIA fuel failure limit has been established. However, experiments (in stagnant water) have been performed which indicate that WWER fuel is unlikely to fail during a RIA with enthalpy values below 160 cal/g. The experiments described under Item A-7, which aimed at investigating RIA fragmentation phenomena, showed that no failure occurred up to 190 cal/g for fresh fuel and up to 160 cal/g for fuel with up to 50 MWd/kg and 140 cal/g up to 60 MWd/kg. Enthalpy values actually expected during this type of postulated accident are well below this level.

Other WWER operating countries follow the western approach; actual limit values for fuel enthalpy may differ between the various countries.

3.23.2. Differences: Description and rationale

For some WWER operating countries, no source term evaluations are assumed necessary (see discussion under item B-6) although this would of course be possible.

For countries that need to evaluate the source term, the WWER fuel vendor recommends a fuel failure limit of 160 cal/g as experimentally verified. Some WWER operating countries have independently established a RIA fuel failure limit similar to PWR operating countries.

3.23.3. Conclusions

See above.

3.23.4. Recommendations

For PWRs, more experiments are under preparation (e.g. CABRI-Water Loop) in order to better establish the RIA fuel failure limit at high burnup. For WWERs, more experimental data are needed in the range of 140–230 cal/g to define a firm failure limit.

3.24. ITEM C-1 CRUD DEPOSITION

3.24.1. Description

The amount of crud deposited and its composition can be significant to the corrosion performance and hydrogen uptake of the cladding (example: crud induced localized corrosion (CILC)). A strong dependence on water chemistry conditions has been observed.

Safety criteria or operational limits on crud deposition are not defined. However, crud deposition on the fuel is normally 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. crud scrape).

Larger crud deposits could also cause axial offset anomalies; in such cases, separate measures may become necessary (see e.g. the OECD report, Chapter 5.6).

No crud deposition criteria exist for WWERs.

3.24.2. Differences: Description and rationale

In WWERs, due to the different water chemistry, crud is mainly deposited in the primary circuit. Basically no crud is found on the fuel; for this reason, no criteria for crud deposition exist.

3.24.3. Conclusions

See above.

3.24.4. Recommendations

As most of the crud is deposited in the primary circuit, causing large amounts of low level waste and high dose rates, some WWER operators are thinking about changing to the western type of water chemistry. The effect on the fuel from such a change needs to be evaluated.

3.25. ITEM C-2 STRESS, STRAIN, FATIGUE

3.25.1. Description

Design criteria are defined to prevent cladding damage due to static and cyclic loads. For Western reactors, the following criteria exist:

- max. allowed *stress* (= load) shall not exceed the levels specified in e.g. ANSI/ANS–57.5 or KTA 3103 part B. These stress levels are usually a function of both the yield and the tensile strength at operating temperature;
- the elastic + plastic *strain* (= deformation) level shall not exceed a specified value, usually 1%, at BOL. This criterion was verified against RIA test (SPERT, TREAT, PBF) results, the 1% total strain being equivalent to about 140 cal/g enthalpy, on the basis that, if the cladding sustains the deformation in this fast transient, the criterion will be valid in slow (Condition II) transients. At EOL, typically a 2.5% limit is defined to limit cladding creep and fuel swelling;
- for *fatigue* (cyclic loads) usually the cumulative effect is limited, e.g. sum of all fatigue life usage ratios < 1.0, based on fatigue failure curves (failure stresses vs. cycle level).

These criteria are analytically verified by the fuel vendor; margins between the above limits and actual stress / strain levels generally depend on the specific material properties of fuel, cladding and on the burnup range.

For WWERs, criteria are defined in a similar way as follows:

- cladding stress always needs to be less than the standard yield strength (for Condition II events);
- no strain limit is defined, however, the failure level for fast transients experimentally verified for WWER reactors is 0.5% plastic deformation; similar RIA tests show no failure up to 160 cal/g;
- in the fatigue limit, creep is included; the summation of damage due to cycling and cumulative tensile stresses must be less than unity.

3.25.2. Differences: Description and rationale

The WWER stress criterion is more restrictive, as it only relates to standard yield strength. Due to this more restrictive limitation, there is almost no plastic strain; thus, no separate strain limit has been defined for WWERs. The creep, which is included in the strain limitation for PWRs, is included in the WWER fatigue limit (classical approach).

3.25.3. Conclusions

In spite of the differences mentioned above, the western and eastern criteria for stress/strain/fatigue are overall consistent.

3.25.4. Recommendations

None.

3.26. ITEMS C-3, C-4 OXIDATION AND HYDRIDE CONCENTRATION

3.26.1. Description

Oxidation and hydriding of Zircaloy materials are directly related to fuel performance for normal operation, transients and accidents and are leading parameters to limit the lifetime of nuclear fuel. Oxidation degrades material properties, most importantly the cladding thermal conductivity (with a consequential increase in fission gas release (and hence rod internal pressure) and 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 western reactors, oxide thickness and hydride concentration limits are often assumed for normal (steady-state) operating conditions for design purposes. Values are usually in the range of 100 microns and 500–600 ppm, respectively, at the end of fuel life; these values are 'empirical', 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 hydride distribution and hence mechanical properties. In some countries no explicit design limits are defined; in some other countries, design limits have been approved by the regulator. In all cases, however, oxidation and hydriding are considered when analysing cladding properties for performing stress and strain related design evaluations.

For WWERs the uniform corrosion limit is 60 microns for ZrNb cladding, on the basis that corrosion is linear up to this value. This value is well above current operating experience, typically 20–25 microns for 70–75 MWd/kg. At this burnup internal oxidation is 2–17 microns; the internal oxidation could become significant at higher burnups. No limit for internal oxidation has been defined so far. Nodular corrosion is not considered to be a problem for WWERs.

Hydrogen levels observed for burnups of 70–75 MWd/kg are in the range of 100–150 ppm hydrogen; because this is so low, no limit was set for hydrogen uptake previously, however at present a design limit has been implemented of 400–450 ppm to match western design criteria.

3.26.2. Differences: Description and rationale

Design limits are defined in a similar manner, duly accounting for the differences in fuel design. Corrosion is less of an issue with WWERs due to the higher corrosion resistance of ZrNb during NOC; in addition, the different water chemistry contributes to the good corrosion performance.

3.26.3. Conclusions

See above.

3.26.4. Recommendations

Internal oxidation is likely to become more significant at higher burnups; as no limits are set at this time, a common position is desirable for eastern and western plants, if possible. If rod internal oxidation is of concern, this might be a suitable topic for further investigation.

3.27. ITEM C-5 TRANSPORT AND HANDLING OF LOADS

3.27.1. Description

The fuel design shall be such that transport and handling loads do not mechanically damage fuel components.

For western reactors, maximum design loads are usually between 2 and 4 g. Also, a maximum allowable tension stress (for Zr bar material) is sometimes used as a design limit. The design is evaluated analytically and experimentally against these limits by the fuel vendor.

WWERs have a vendor design load limit of 4 g; for this load, tests are performed to verify the design.

The above western design criteria are not mentioned in the OECD report.

3.27.2. Differences: Description and rationale

No difference in approach between east and west; numbers vary according to the fuel design and/or transport and manipulation requirements.

3.27.3. Conclusions

See above.

3.27.4. Recommendations

None.

3.28. ITEM C-6 FRETTING WEAR, FRETTING CORROSION

3.28.1. Description

The fuel assembly design shall be such that fuel rod failure due to fretting does not occur or does not exceed limiting values that could lead to a reduction of fuel assembly structure stability and fuel rod life time.

For western reactors, no explicit design limits are in place; other design limits (such as stress and strain limits) are considered to preclude fretting wear. Verification is however performed by the vendor for each fuel design, both analytically and via mock-up (endurance) tests.

A minimum spring force is sometimes defined to guarantee the contact with the cladding until EOL, accounting for clad creepdown and spring relaxation. It is found, however, that such a design requirement will not completely prevent fretting.

For WWERs, a first fretting wear design criterion requires that no fretting (due to rapid movement such as vibration e.g. in lower or upper tie plates) shall occur after min. 3000 hrs endurance testing. A second design criterion for avoiding fretting wear limits the cladding reduction (due to creepdown) to 0.10 mm; this criterion is in place due to the different spacer grid design of WWER fuel, which does not include any springs — the contact between grid and fuel rods is controlled only by the grid construction and must be warranted also after cladding creepdown.

A separate design criterion exists to limit corrosion after continuous (slow) fretting in the contact points of the spacer grid. Such fretting corrosion is not to not exceed 10–15 microns. This criterion is confirmed by a large database from post-irradiation examinations up to discharge burnup levels of ~50 MWd/kg; values observed are 5–10 microns. For PWRs, similar values have been observed.

3.28.2. Differences: Description and rationale

The overall concept for limiting fretting wear is similar between east and west. The difference in the fuel (spacer) design leads to an additional design criterion for fretting corrosion and cladding creepdown for WWERs.

3.28.3. Conclusions

See above.

3.28.4. Recommendations

It is desirable that the fuel design aim at avoiding fretting. Currently some fuel vendors are changing the design criteria in order to achieve this goal.

3.29. ITEM C-7 CLADDING DIAMETER INCREASE

3.29.1. Description

For WWERs, it was observed experimentally that single event PCI criteria (stress below 230 MPa, see Item B-3) no longer protect against stress corrosion cracking beyond a creep and cyclic accumulation of plastic deformation of 0.4%. Thus, a design (strain) criterion limiting cladding diameter increase of 0.4% was put in place, covering creep and cyclic accumulation of plastic deformation. For practical purposes, this design criterion is transformed into an operational recommendation to limit the number of significant power transients (including scram, startups etc.).

(Note: the text in the IAEA report (criterion 1.7 / DC3) does not adequately describe this issue.)

For western reactors, no such limit is defined; the requirement is considered to be covered by existing PCI criteria (see Item B-3).

3.29.2. Differences: Description and rationale

Approach similar between east and west, with the exception of the definition of an additional design criterion prohibiting creep and cyclic accumulation of plastic deformation above 0.4% instead of extending the PCI rules. In practice, plant operation is not much affected by this extra criterion or the derived operating recommendation (as an example, load following is limited).

3.29.3. Conclusions

See above.

3.29.4. Recommendations

Research needs to be continued for a better understanding of the different behaviour of Zircaloy4 and Nb-containing Zr-alloys for cyclic damage accumulation.

3.30. ITEM C-8 CLADDING ELONGATION

3.30.1. Description

Following a general fuel design requirement, the fundamental mechanical and hydraulic functions of the assembly shall not be impaired due to irradiation growth of fuel rods and channel; in particular, the fuel assembly shall give sufficient space for differential rod growth to occur without it becoming restrictive.

For western reactors, no explicit elongation (axial growth) design limits are defined. The vendor design process includes verification of the general design requirement for conditions I and II against values obtained from experimental data (in-pile and out-of-pile).

For WWERs, an ultimate cladding elongation criterion is defined, see the IAEA report 9316, for conditions I-IV.

3.30.2. Differences: Description and rationale

For WWERs, the design criterion applies to all conditions including accidents (i.e. Conditions I–IV), for western reactors the criterion only applies to Condition I and II events. For WWERs it is considered that the load on the cladding has to be minimized during accidents, especially during a LOCA (see Item A-10).

3.30.3. Conclusions

See above.

3.30.4. Recommendations

The PWR criterion needs to be reviewed in view of the effect of rod elongation during Condition III and IV events.

3.31. ITEM C-9 RADIAL PEAKING FACTOR

3.31.1. Description

The radial peaking (F_r for WWERs or enthalpy rise hot-channel factor $F_{\Delta h}$ for PWRs) is sometimes used as a limit to prevent DNB and for WWERs also to prevent reaching saturation temperature of the coolant on the assembly outlet under normal operating conditions and AOOs (Condition I and II events).

A radial peaking factor (K_r or F_{xy}) is derived by including the uncertainties in measurements, design methods and fabrication tolerances; this becomes one of the limits for reload design purposes. The limit is also verified during operation with the use of core monitoring programs.

For most western reactors, the radial peaking is employed to indirectly verify the DNBR criterion (see Items A-1 and B-1) not only for core design but also during plant operation; for this reason, it is sometimes specified in the technical specifications of the plants.

For WWERs with Russian legislation, this is a licensed limit as no operating limit DNB is defined (see Item B-1).

3.31.2. Differences: Description and rationale

For WWERs, the definition of K_r additionally covers the effect of assembly bowing with an additional margin for conservatism. Also, in WWERs the in-core instrumentation for monitoring this and other limits is more sophisticated than in PWRs. A summary of currently used core monitoring systems (CMS) is presented in Appendix II.

3.31.3. Conclusions

Similar approach but different licensing procedures.

3.31.4. Recommendations

None.

3.32. ITEM C-10 3D PEAKING FACTOR

3.32.1. Description

A total peaking or ‘hot spot’ factor (K_0 or F_Q) is defined for design purposes to limit local power peaking during normal operation. The limit is also verified during operation with the use of core monitoring programs.

For western reactors and for WWERs, the 3-D peaking factor is employed to indirectly verify LHGR (see item B-2) as well as the DNBR operating limit (see item B-1) not only for core design but also during plant operation; for this reason, it is sometimes specified in the technical specifications of the plants.

3.32.2. Differences: Description and rationale

No difference. For core instrumentation and core monitoring, see the previous item C-9.

3.32.3. Conclusions

See above.

3.32.4. Recommendations

Special attention must be paid to WWER440 fuel due to potentially large local power peaking (up to 70%) in the fuel surrounding the connecting part between the absorber and the fuel follower of the control rod (see Appendix III for more information). A special IAEA meeting has recently been dedicated to this issue (IAEA Technical Meeting on WWER local peaking induced by control rods.

3.33. ITEM C-11 CLADDING STABILITY

3.33.1. Description

Cladding stability limits are defined to prevent clad collapse due to ovalization. For western reactors these are normally design limits, constraining elastic and plastic deformation, which are verified analytically.

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

3.33.2. Differences: Description and rationale

Basically the approach is identical, analytically verifying ovality for the fuel lifetime. For WWERs, the primary circuit integrity test posed an additional design criterion.

3.33.3. Conclusions

See above.

3.33.4. Recommendations

None.

4. SUMMARY AND CONCLUSIONS

The comparison of PWR and WWER fuel safety criteria, as described in detail in Section 3, shows that there are many common features between the western and eastern criteria. There appears to be no fundamental difference in the basic approach to defining and classifying safety criteria. Differences observed between the criteria and their numerical values are mostly due to differences in fuel design and materials; also, differences in reactor characteristics and country specific licensing requirements sometimes lead to differences between criteria definitions. All these contrasts need to be understood for a correct explanation of the divergence in safety criteria that has been observed.

In the various workshops which involved the various western and eastern representatives, such an understanding was achieved. The following three tables summarize the results of this comparison, one table for each of the three categories defined in Section 1. It should be noted that the text is very brief, and that the detailed explanations in Section 3 need to be consulted for a proper understanding of these summarized results.

In most cases where differences were observed, the WWER criteria (or their numerical values) proved to be conservative; partly this is a result of the changed general safety requirements during the late-1980s and early 1990s.

This report not only identifies common features and differences, but also serves as a basis to outline the general safety case (= the complete set of safety evaluations needed for evaluation by the regulator) for PWR and WWER fuel. Therefore, it will be very beneficial for PWR and WWER licensing activities, as it will help to focus on the issues of importance for individual fuel safety case reviews. The report generally makes the safety level of fuel design and operation more visible and transparent for both PWR and WWER operating countries.

As part of the fuel safety assessment, Core Monitoring Systems were also reviewed. For WWERs, there is a clear trend to implement and license "state-of-the-art" systems which include a 3-dimensional core simulator for calculating the power distribution and associated operating limit margins, with appropriate coupling to the measured detector readings. For PWRs, this trend is not yet visible; in a few cases such state-of-the-art systems are available, however the traditional (vendor made) process computer is still the licensed tool for verifying compliance with the cycle data that are available from the core design process.

To conclude, this report highlights the basic safety principles and their bases, and broadens the insight in failure mechanisms; this will be beneficial for future FSC review, particularly with reference to new fuel or core designs. As an example, the report is likely to be appreciated by those plant operators who intend to operate with mixed cores. The need to review the FSC periodically, and on an international level, was generally recognized in order to reflect the many challenges and innovations in the fuel area. Especially in view of high burnup and new materials, there is a need to further develop the safety criteria and their numerical values; this might include the analysis of relevant experimental data. Even without such innovations, there is a clear potential for refining and improving some of the PWR and WWER criteria and/or their numerical values. Thus, this report can serve as a basis for discussions on a more in-depth co-operation for future R&D activities that are needed to verify the existing safety criteria or to support improvements. Generally, further discussion is necessary to harmonize safety approaches within the constraints of design differences. Therefore, a closer collaboration in the review of the safety criteria for PWR and WWER plants and in reviewing the respective safety cases is recommended.

TABLE IV. SAFETY CRITERIA

	Criterion	Summary of comparison
1	DNB safety limit	Difference only in CHF- correlations used
2	Reactivity coefficients	In some countries that operate WWERs, each reactivity coefficient must be negative (instead of the sum of all reactivity coefficients)
3	Shutdown margin	Additional requirement in the Russian Federation for new generation NPPs: no recriticality down to 100 deg C coolant temp.
4	Enrichment	No difference
5	Internal gas pressure	In some countries that operate WWERs, the more restrictive of the two PWR criteria is used
6	PCMI	Same approach, however different basis for defining criteria
7	RIA fragmentation	Different limit values, approach identical
8	Non-LOCA runaway oxidation	Criterion A-9 applies to all DBAs; different value for some PWRs, safety approach identical
9	LOCA-PCT	Same limit values, but different basis
10	LOCA-Oxidation	Almost same limit value, but different basis
11	LOCA-H release	No difference
12	LOCA-long term cooling	No difference in approach
13	Seismic loads	No difference in approach
14	Hold-down force	No difference
15	Criticality	No difference

TABLE V. OPERATING CRITERIA

	Criterion	Summary of comparison
1	DNB operating limit	Same basic requirement, but different licensing approach (see Item C-9)
2	LHGR limit	Same approach
3	PCI	No difference in approach, rules/values are design dependent
4	Coolant activity	Same approach, WWERs have extra (lower) limit for decision on further operation
5	Gap activity	For WWERs, no separate criterion, covered by Item B-6
6	Source term	Different approaches, country dependent
7	Control rod drop time	No difference
8	RIA fuel failure limit	In some countries that operate WWER plants, the number of failures is not calculated. However, a failure limit is recommended by the fuel vendor (see text)

TABLE VI. DESIGN CRITERIA

Criterion		Summary of comparison
1	Crud deposition	For WWERs, no limit defined due to different water chemistry (see text)
2	Stress / strain / fatigue	Differences due to more restrictive stress criterion for WWERs, overall approach identical
3	Oxidation	Same approach, differences are due to different fuel designs
4	Hydride concentration	See Item C-3
5	Transport loads	Same approach
6	Fretting wear	Same approach, two additional design criteria for WWERs due to different spacer designs
7	Clad diameter increase	Additional strain criterion for WWERs
8	Cladding elongation	Same criterion, applies to Conditions I and II for PWRs and to Conditions I to IV for WWERs
9	Radial peaking factor	Same criterion, different licensing approach
10	3D peaking factor	Same criterion, different licensing approach
11	Cladding stability	Same approach, additional design criterion for WWERs

REFERENCES

[1] NUCLEAR ENERGY AGENCY OF THE OECD, Fuel Safety Criteria Technical Review, CSNI/R (99) 25, OECD, Paris (2001).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment and Verification for Nuclear Power Plants, Safety Standard Series No. NS-G-1.2, IAEA, Vienna (2001).

[3] American Nuclear Society, LWR fuel design and evaluation, American National Standard ANSI/ANS-57.5 Appendix A, (1996).

[4] TONG, L.S., WEISMAN, J., Thermal Analysis of Pressurized Water Reactors, American Nuclear Society (1978).

[5] MCDONALD, P.E., et al., "Assessment of light-water-reactor fuel damage during a reactivity-initiated accident", Nuclear Safety 21, 5 (1980).

Appendix I
REFERENCE TABLES OF FUEL SAFETY CRITERIA

TABLE I.1. REFERENCE TABLE FOR WESTERN FUEL SAFETY CRITERIA (FROM REF. [1])

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	*Categories:*
(o) cladding embrittlement / oxidation	C	3, 4, 7, 8	*A – normal operation*
(p) blowdown / seismic loads	C	3, 7	*B – anticipated transients*
(q) assembly hold-down force	A, B, C	1, 11	*C – postulated accidents*
(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	

TABLE I.2. REFERENCE TABLE FOR WWER FUEL (FROM REF. [2])

No.	Fuel safety criteria	Code	Type	Category	Potential impact by 'new' requirements
1.1.	Fuel stress corrosion cracking of cladding	SC1	FR	A, B	1.1*, 1.2*, 1.3*, 1.4*, 2.2*, 3.1*, 4.1*, 5.1*, 7*, 8.1, 8.2, 9*, 10*, 11.1*, 11.2*
1.2.	Ultimate stress of cladding	SC2	FR	A, B	no impact (?)
1.3.	Ultimate pressure of coolant	SC3	FR	A, B	1.4, 3.1, 4.1, 5, 5.1, 6
1.4.	Ultimate value of cladding damageability	SC4	FR	A, B	no impact (?)
1.5.	Ultimate value of cladding diameter reduction	DC1	FR	A, B	1*, 1.4*, 2*, 2.1*, 3.1*, 3.2*
1.6.	Limiting cladding elongation (axial growth)	DC2	FR	A, B	1*, 4, 4.1*
1.7.	Limiting cladding diameter growth	DC3	FR	A, B	3.1*, 4.1*, 7*, 8.1, 8.2, 9, 11.1*, 11.2*
1.8.	Limiting Fuel Temperature	TC1	FR	A, B	no impact (?)
1.9.	Limiting rod internal gas pressure	TC2	FR	A, B	1*, 6*, 8*
1.10.	Ultimate fuel rod linear power	TC3	FR/FA/ CD	A, B	1.1*, 1.2*, 1.3*, 1.4*, 2.2*, 3.1*, 3.2*, 7*, 8.2*, 9*, 10*, 11.1*, 11.2*
1.11.	Limiting LHGR ramp during transients	TC4	FR	B, C	1, 1.1*, 1.2*, 1.3*, 1.4*, 2.2*, 3.1*, 4.1, 5.1*, 7*, 8.1, 8.2, 9, 11.1*, 11.2*
1.12	Cladding outer surface oxidation	CC1	FR	A, B	1.4*, 3.1*, 5*, 8.1, 8.2, 10, 11.1*, 11.2*
1.13.	Cladding fretting corrosion	CC2	FR	A, B	1.4*
1.14.	Peak cladding temperature (PCT)	AC1	FR	C	3, 4*,
1.15.	Maximum cladding local oxidation depth	AC2	FR	C	4.1*
1.16.	Maximum fraction of Zr in the core reacted with steam	AC3	FR	C	criterion is not changed and its general

'New' requirements
(for more details see Section 2)

1. New FR design
1.1. IFBA Gd_2O_3
1.2. Pellet microstructure optimization
1.3. Central hole diameter increase
1.4. FR diameter decrease (WWER-440/230)
2. New FA design
2.1. Modification to WWER-1000 FA
2.2. Modification of the fuel/absorber connection(WWER-440)
3. New core design
3.1. Load follow methodology
3.2. Low leakage pattern
4. New materials
4.1. E 635 alloy
4.2. Hf content decrease
5. New manufacturing procedures
5.1. Tube manufacturing
6. Long fuel cycle
7. Uprated power (WWER-440/213 only)
8. Higher burnup
8.1. Peak fuel rod average 60 MWd/kgU
8.2. Peak fuel assembly average 52 MWd/kgU
9. MOX
10. Mixed core
11. New operating practices
11.1. 5-6 years of residence time at base load
11.2. 7-8 years of residence time at load follow

1.17.	Pellet cross section averaged maximum enthalpy	AC4	FR	D	1.1*, 1.2*, 1.3*, 4.1*, 8.1, 8.2, 9*, 11.1*, 11.2*
1.18.	No local fuel melting	AC5	FR	C, D	
2.1.	DNBR	DN	FA	A, B	1.1, 1.4, 2.1, 3.2, 6, 7,8, 9,10,11
2.2.	Rod drop time	RD	FA	A, B, C	2.1, 3.1, 3.2, 4.1, 5.1, 8.1, 8.2, 10,11.1, 11.2
2.3.	Transport loads	TL	FA	–	2.1, 4.1
2.4.	Assembly hold down force	AHF	FA	A, B, C	numerical values of 1.4, 2.1, 5.1, 4.1, 8.2, 11may need to be revised
2.5.	FA fretting wear		FA	A, B	1.4, 2.1, 3.1, 4.1, 5.1, 6, 8, 10, 11
3.1.	Coolant activity	CA	(FA/CD)/PS	A, B	3.1, 4, 5.1, 6, 7, 8,9,11
3.2.	Moderator temperature coefficient	RK	CD	A, B, C	1.1, 1.3, 1.4, 2, 3.2, 4, 6, 7, 9, 10, 11 can adversely affect the value of MTC
3.3.	Shutdown margin	SM	CD	A, B, C	1.1, 1.3, 1.4, 2.1, 3.2, 4, 6, 7, 8, 9, 10, 11.
3.4.	FR radial peaking factor - K_r		CD	A, B	1.1, 1.3, 1.4, 2, 3, 4, 6, 7, 8, 9, 10,11
3.5.	FR 3D peaking factor - K_0		CD	A, B	1.1, 1.3, 1.4, 2, 3, 4, 6, 7, 8, 9, 10,11
3.6	Seismic loads	SL	FA	C	2.1, 4.1

.

Type of fuel design
FR - Fuel rod

FA - Fuel assembly
CD - Core design

PS - Plant systems

Category of operational modes
A. Normal operation
B. Anticipated operational occurrences
C. DBA (LOCA, transients incl. reactivity transients
D. RIA (fuel fragmentation)

*numerical value of the criterion might be changed

TABLE I.3. CROSS-REFERENCE TABLE

No.		OECD/NEA Report		IAEA Report
A			Safety Criteria	
1.	3.1.	(a) DNBR - safety criteria	2.1.	DNBR - safety criteria
2.	3.2.	(b) reactivity coefficient	3.2.	Moderator temperature coefficient
3.	3.3.	(c) shutdown margin	3.3.	Shutdown margin
4.	3.4.	(d) enrichment		
5.	3.8.	(i) internal gas pressure	1.9.	Limiting rod internal gas pressure
6.	3.9.	(j) PCMI		
7.	3.11.	(l) fuel fragmentation (RIA)	1.17.	Pellet cross section averaged maximum enthalpy
			1.18.	No local fuel melting
8.	3.13.	(n) cladding embrittlement / PCT (non-LOCA run away oxidation)	1.14.	Peak cladding temperature (PCT)
9.	3.14.	(o) LOCA PCT	1.14.	Peak cladding temperature (PCT)
10.	3.14.	(o) LOCA oxidation	1.15.	Maximum cladding local oxidation depth
11.	3.14.	LOCA H-release	1.16.	Maximum fraction of Zr in the core reacted with steam
12.	3.14.	LOCA long term cooling		
13.	3.15.	(p) blowdown / seismic loads	3.6.	Seismic loads
14.	3.16.	(q) assembly hold-down force	2.4.	Assembly hold down force
15.		Criticality safety		Criticality safety
B			Operational Criteria	
1.	3.1.	(a) DNBR – operational limit	3.4, 3.5.	
2.	3.9.	LHGR limit	1.4, 1.8, 1.10.	Ultimate fuel rod linear power
3.	3.10.	(k) PCI	1.11.	Limiting LHGR ramp during transients
			1.1.	Fuel stress corrosion cracking of cladding
4.	3.17.	(r) coolant activity	3.1.	Coolant activity
5.	3.18.	(s) gap activity		
6.	3.19.	(t) source term		
7.		Control rod drop time	2.2.	Rod drop time
8.	3.12.	(m) RIA fuel failure		
C			Design criteria	
1.	3.5.	(e) crud deposition		
2.	3.6.	(f) stress/strain/fatigue	1.2, 1.4	Ultimate stress of cladding, ultimate value of cladding damageability
3.	3.7.	(g) oxidation	1.12., 1.13	Cladding outer surface & fretting oxidation
4.	3.7.	(h) hydride concentration	1.12	
5.		Transport loads	2.3.	Transport loads

6.	Fretting wear, fretting corrosion	2.5.	FA fretting wear
		1.5.	Ultimate value of cladding diameter reduction
		1.13.	Cladding fretting corrosion
7.	Clad diameter increase	1.7.	Limiting cladding diameter growth
8.	Cladding elongation	1.6.	Ultimate elongation of cladding
9.	Radial peaking factor	3.4.	FR radial peaking factor - Kr
10.	3-D peaking factor	3.5.	FR 3D peaking factor - K0
11.	Cladding stability	1.3	Cladding stability

Appendix II
CORE MONITORING SYSTEMS

Fuel safety criteria (FSC) are normally exhibited in the following publications:

- Safety Analysis Report,
- Technical Specifications, and
- Fuel/cycle specific nuclear design reports.

During core operation, compliance — either directly or indirectly — with FSC and with assumptions made in the above mentioned documents and analyses is demonstrated with the use of the core and reactor instrumentation and the on-line evaluation thereof, commonly called *core surveillance* or *core monitoring*.

Core monitoring tools have a long development history. Traditionally, the main focus has been on in-core power and coolant temperature distribution, and corresponding core monitoring systems were developed and maintained by reactor/fuel suppliers; such systems reflect the state of the art in technology of the 1970s.

Two basic approaches for core monitoring can be identified for PWR and 'old' (i.e. commissioned before 1990) WWER reactors:

- one approach, represented by e.g. Westinghouse and Siemens-KWU (now Framatome ANP), relies on pre-calculated data from core/cycle design that are used during cycle operation to monitor FSC; from time to time these data are verified against flux map measurements using movable in-core detectors;
- the other approach, used in all WWER and in Combustion Engineering reactors, employs on-line measurements from self powered neutron detectors (SPND) and fuel outlet temperature sensors which process, however, cannot provide a direct absolute power evaluation.

Thus, in western PWRs the demonstration of compliance with FSC generally relies on pre-operational analyses, with periodic in-core measurements during operation to confirm the appropriateness of these analyses. Any discrepancies require time consuming analyses and/or costly special solutions. In WWER reactors the Technical Specifications were based on non-measurable parameters such as assembly temperature rise and on core outlet temperature measurements; consequently, core monitoring models used for power distribution reconstruction depended on measurement interpolation. In summary, simple core monitoring tools were traditionally developed for WWER and PWR reactors. The simplicity was also essential due to the very limited real-time computing capacity of the plant I&C hardware. The methodology related to these tools was usually licensed (regulatory approved) for monitoring of FSC and FSC related operating margins.

A significant change in philosophy and approach to core monitoring can nowadays be observed with more advanced and accurate core modelling as well as much increased computing power made available through new technology, allowing more and more detailed evaluations to be performed within a very short amount of time. Today, core monitoring systems (CMS) are available including a 3-dimensional core simulator, allowing an accurate evaluation of the distribution of local power (fuel bundle or fuel rod), with the following characteristics:

- provide a helpful tool for reactor operators as well as core physicists to optimize core control and core operation;
- easy surveillance of Technical Specification requirements and of key parameters representing the basic assumptions and initial conditions for safety analysis transients and accident scenarios; and
- direct monitoring of margin to limits: CMS are capable of predicting and evaluating real-time (on-line) the margin to applicable fuel safety limits (e.g. calculate actual minimum DNBR instead of

static enthalpy rise hot channel factor, or fuel centreline temperature instead of maximum local linear heat rate).

At present CMS combine very sophisticated modelling, based on a large amount of verification and validation, as well as user-friendly man-machine interface programs (MMI, usually containing advanced graphics modules) with high computing power (most of the CMS are now running on standard PCs). Due to low uncertainties in prediction of key parameters CMS can provide utilities with significantly higher operating and manoeuvring flexibility, while at the same time licensing authorities can be assured that the fuel in safe in use by properly and permanently monitoring the core. Also, CMS may be beneficial for optimizing the requirements of the technical specifications.

State of the art core CMS (containing a 3-D core simulator) allow evaluation of margins for the following fuel safety criteria:

- DNB(R)
- 3D power distribution including axial and radial power peaking factors
- LHGR
- PCI
- coolant activity; and also provide an improved validation of plant measurements (e.g. drifting or failed sensor identification).

In several — notably Eastern European — countries, core monitoring systems with a 3-D core simulator have been adopted and licensed. As an example, the “SCORPIO” CMS has been implemented at WWERs in Dukovany, Bohunice, Temelín (Czech Republic) as well as in Loviisa (Finland) and in the McGuire, Catawba and Oconee (USA) and Ringhals (Sweden) PWRs.

Figure II.1 shows one of the screens available for the operator from this CMS [II.1]:

FIG. II.1. Example of screen available for the operator from the CMS.

To date, in only a few Western European PWRs core monitoring systems have been installed to perform core surveillance in parallel to the existing core monitoring tools of the first generation. In the following figure one of screens available for the operator from the "GARDEL" CMS [II.2] is shown:

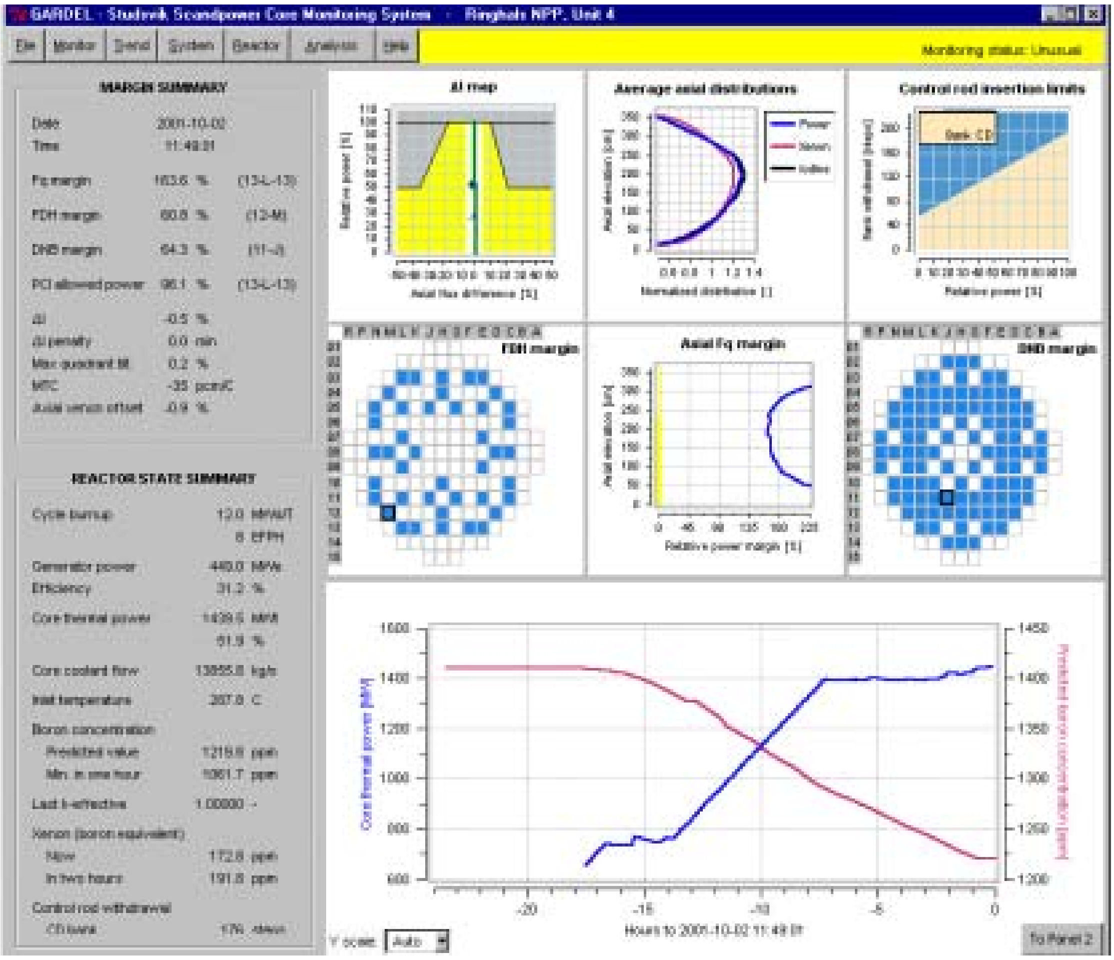

FIG. II.2. Example of screen available for the operator from the "GARDEL" CMS.

Clearly, also in western Europe the new generation of CMS have the potential to replace the existing core monitoring tools. In spite of this potential, the progress of verifying or validating these CMS and licensing their models is rather slow; this may at least partly be contributed to the fact that western PWRs are mostly operating in "baseload", i.e. without appreciable deviation between core/cycle design and operation (see [II.3] for more detailed information on the implementation status of CMS).

The new generation CMS with state of the art models are designed to cover more than just plant operational support. Basically, they offer three types of function or 'modes':

- core follow mode
- predictive mode
- reload design mode

In the *core follow* mode, a core state-point calculation is periodically performed (e.g. on a hourly basis); the theoretical calculation of the core power distribution (based on the heat-balance from plant instrument signals) is then combined with any available in- or ex-core detector readings. An automatic check against fuel operating limits (i.e. operating margins evaluation) is performed for this state-point. The operator obtains relevant information on core status and operating margins through the MMI in the form of trend curves, core map pictures and diagrams displaying operating margins. This is the plant operational support function, as already described in more detail earlier.

In the *predictive* mode, the operator can forecast the reactor behaviour during the coming period of time (e.g. for the next hours during a plant manoeuvre). Obviously, in this case no plant instrument signals are available, and the accuracy of the predicted core state depends strongly on the quality of the physics modelling in the 3-D core simulator. Again, the projected state-points are checked against fuel operating limits, and the predicted behaviour of the core and the fuel may be analysed by the operator through a number of dedicated displays from the MMI. Such predictive capabilities may be of importance for strategy planning, offering the possibility to check the consequences of operational manoeuvres in advance by a prediction of critical parameters, or in case of unusual operational events.

The *reload design* mode enables straightforward core configuration and fuel operating limits (margins) calculations for the following cycle through the MMI (or a separate, specially designed MMI). In this mode the CMS is linked with the off-line core design code system for core loading pattern design, preparing configuration files, archiving of core follow data, calculating neutron fluence data at the reactor vessel wall, etc.

Summary and conclusions

Many NPPs continue to monitor FSC during plant operation indirectly, using first generation core monitoring tools that rely on data which are pre-calculated as part of the core design process. However, a new generation of core monitoring systems with state of the art (3-dimensional) core simulation methods has become available, and has been implemented successfully in various — notably Eastern European — NPPs. This type of core monitoring system allows a direct and continuous on-line assessment of key FSC and subsequent evaluation of fuel safety margins, and thereby simultaneously offers:

- a higher operating and manoeuvring flexibility (to the operator);
- the necessary guarantee that the fuel is safe in use (to the regulator);
- synergy with core design processes, and thus potential fuel cycle savings (to the utility).

REFERENCES TO APPENDIX II

[II.1] ØIVIND, B., et al, “User Interface design and system integration aspects of core monitoring systems”, paper presented at the OECD/NEA Workshop on Core Monitoring for Commercial Reactors, Stockholm (1999) (Ref. [II.3]).

[II.2] NOEL, A, BECKER, A., “State-of-the-art Core Monitoring in BWR and PWR”, paper presented at Annual Meeting on Nuclear Technology 2000, Dresden (2001).

[II.3] OECD/NEA Workshop, Core Monitoring for Commercial Reactors: Improvements in Systems and Methods, (ISBN 92-64-17659-4) Proceedings, Stockholm (1999).

Appendix III
WWER-440 LOCAL PEAKING PROBLEMS

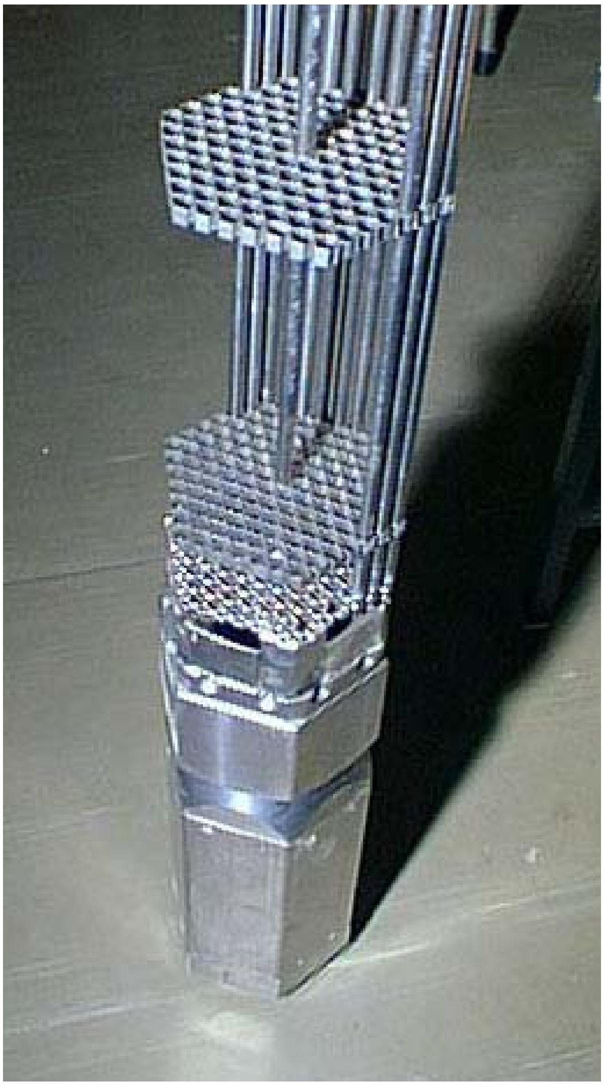

FIG.III.1. Model of the fuel follower — absorber junction of a control assembly.

Differences of WWER-440 reactors, compared with western PWR and WWER-1000, are especially found in the fuel assembly as well as in the absorbing control components design. As a consequence, there are specific aspects related to the local power distribution.

For design purposes and to limit local peaking during normal operation a total peaking or 'hot spot' factor (K_Q or F_Q) is defined, similarly as for any other PWR reactors. This limit is always confirmed in the reloads designs and verified during operation with core monitoring system. For western reactors and for WWERs, the 3-D peaking factor is employed to indirectly verify LHGR (see Item B-2) as well as the DNBR operating limit (see Item B-1) not only for core design but also during plant operation; for this reason, it is sometimes included in the technical specifications of the plant.

Additionally, some specific operating modes are sometimes applied to take into account the PCI operating limit (see Item B-3) to avoid fuel failures during local power transients (e.g. reactor start-up, manoeuvres or transients).

The major differences between the WWER-440, the PWR and WWER-1000 are:

- lower average and local LHGR of fuel rods,
- smaller core size and power,
- smaller hexagonal fuel assemblies s wrapper tube, and
- different control rods.

These differences lead to higher margins in the fuel safety criteria, resulting in higher safety and a negligible fuel failure rate in comparison with other LWR reactors.

In WWER-440 plants, reactivity is controlled by absorbing assemblies working in tandem with fuel assemblies, as shown in Fig. 1. This arrangement leads to large power disturbances in the core power distribution which, however, are always properly calculated within the safety design process and monitored during plant operation.

Besides power disturbances in the fuel assembly scale, there are local disturbances due to the specific design of absorbing assembly. Water in the junction of the fuel follower and the absorber of a control assembly causes a considerable local peak of the power distribution in its vicinity. The power peaking (and the resulting power change when control rods are moved) is greatest in the peripheral rods and drops down rather quickly with increasing distance from the control rod. These local disturbances were usually included into a 3-D peaking factor limit via so called engineering factors and therefore not considered in the power distribution calculations and not explicitly analysed and monitored during operation. This approach reflected the available hardware and software, as well as the available core instrumentation.

Such a practice was sufficient for the past WWER-440 core and reloads design and steady state operational mode of the plant. It should be emphasized that the overall WWER-440 fuel failure rate was and is very low, but statistically more fuel rod failures were registered in assemblies in the surroundings of a control assembly. These very small statistical numbers may nevertheless indicate a possible impact of the local peaks accompanying movements of the control assemblies.

State-of-the art computer modelling of core designs allows now to evaluate the local power distribution in the rods close to the fuel follower – absorbing assembly junction. Such evaluations were performed and published by Russian, Czech and other core designers in countries which operate WWER-440s. The results of these evaluations have shown that this local power peaking may in some cases be as high as 70% (Figures III.2 [III.1] and III.3 [III.2]). However, the uncertainty of such computations is quite high as a consequence of a complex geometry of the junction.

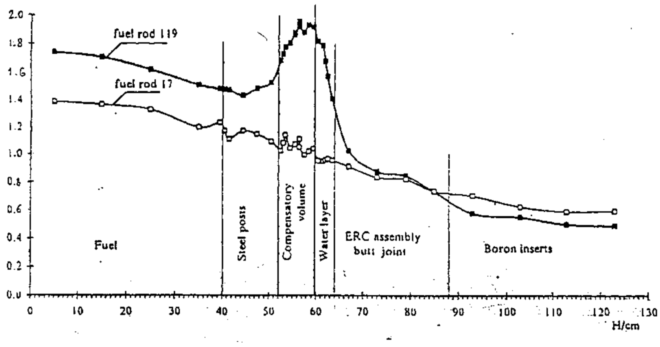

FIG. III.2. Distribution of relative heat rate over fuel rod height (Qz) of working assembly adjacent to CPS working group [III.1].

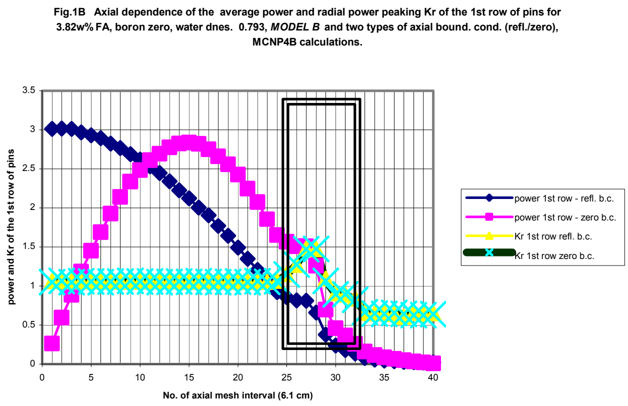

FIG. III.3. Axial dependence of the average power and radial power peaking Kr of the 1st row of pins for 3.82w% FA, boron zero [III.2].

The incentive to reduce electricity generation costs and to comply with the grid demands for load-following operation mode brings about a number of improvements in fuel design and utilization:

- higher burnups,
- reactor power increase,
- mixed cores, and
- load-following operation.

These improvements reduce the safety margins in the individual safety criteria and they have a non-negligible effect on the local power distribution. Preliminary evaluation of the influence of burnup extension on the LHGR due to the local peaking led to the conclusion that the LHGR limit may be reached (Fig. III.4).

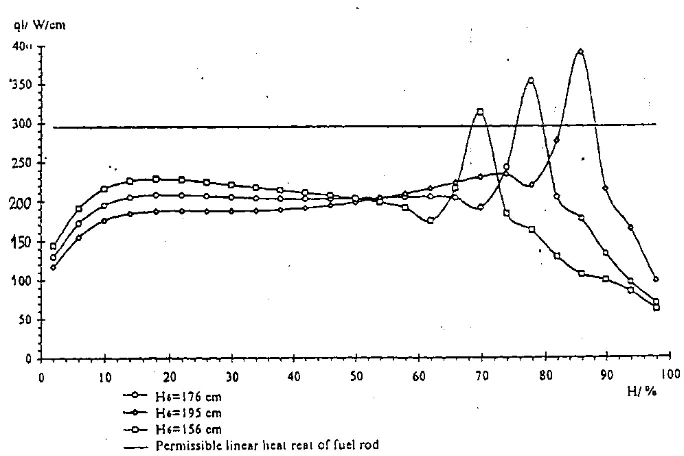

FIG. III.4. Distribution of linear heat rate over the height of fuel rod [1].

The local power peaking may also require that more operating restrictions must be imposed to comply with the PCI criteria.

Compliance with the fuel safety criteria, for the intended mentioned above improvements can be achieved by employing more accurate design methods which use a part of safety margins previously included into the engineering factors, and by design modifications such as hafnium plates in the junction which substantially reduce the local power peaking [III.3].

Present core and reload design methods, as well as core monitoring systems [III.4] for WWER-440 enable to take into account the local power peaking in the fuel follower — absorber junction surroundings. A certain complication follows from the uncertainty in the computer codes, their adjustment for the complex junction geometry and their validation.

Existing experimental data are insufficient to remove these uncertainties and to validate the computer codes, more experiments with the real junction geometry and material composition are needed to generate data for these purposes.

It may be concluded that:

- WWER-440 cores are extremely safe, and fuel failure resistant
- a generic feature of WWER-440 core is a large local power peaking due to the control assembly design specifics
- it is desirable to utilize the large safety margins which will allow to improve the power plant economy, and for this purpose:
- are available advanced core design methods and core surveillance systems that are capable to cover the local power peaking,
- however, more experimental data are needed to achieve sufficient accuracy and validation of the computer codes.

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ABBREVIATIONS

AOO	Anticipated operational occurrence (‘normal’ transient)
BOC	Beginning of cycle (after refuelling)
BOL	Beginning of life
BWR	Boiling water reactor
CHF	Critical heat flux (at which DNB occurs)
CILC	Crud induced localized corrosion
CMS	Core monitoring system
CSNI	Committee on the Safety of Nuclear Installations
DBA	Design basis accident
DNB(R)	Departure from nucleate boiling (ratio)
ECCS	Emergency core cooling system
ECR	Equivalent cladding reacted (LOCA oxidation)
EOC	End of cycle (before refuelling)
EOL	End of life
FA	Fuel assembly
FGR	Fission gas release
FSC	Fuel safety criteria
HTP	High thermal performance
IFMS	Intermediate flow mixers
LHGR	Linear heat generation rate
LOCA	Loss of coolant accident
LOCA-H	Loss of coolant accident-hydrogen
LWR	Light water reactor
MOX	Mixed oxide fuel (i.e. containing both U and Pu)
MMI	man-machine interface (also called ‘user interface’)
NOC	Normal operating condition(s)
NPP/NPS	Nuclear power plant/station
PCI	Pellet–cladding interaction (=stress corrosion cracking)
PCIOMR	Preconditioning interim operating management recommendation (to avoid PCI)
pcm	1/1000 of a per cent (0.00001 fraction)
PCMI	Pellet–cladding mechanical interaction
PCT	Peak cladding temperature
PWG2	Principle Working Group 2
PWR	Pressurized water reactor
ppm	Parts per million (concentration)
R&D	Research and development
RCS	Reactor coolant system
REA	Rod ejection accident
RIA	Reactivity initiated accident (=REA in case of a PWR/WWER)
SCC	Stress corrosion cracking
SDM	Shutdown margin
SPND	Self powered neutron detectors
TFFSC	Task force on fuel safety criteria
WA	Working assembly
WWER	Water moderated, water cooled reactor

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