Unclassified **NEA/CSNI/R(2003)10**

Organisation de Coopération et de Développement Economiques
Organisation for Economic Co-operation and Development **17-Jul-2003**

English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2003)10
Unclassified

FUEL SAFETY CRITERIA IN NEA MEMBER COUNTRIES

Compilation of responses received from member countries

March 2003

JT00147506

English text only

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

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

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

FUEL SAFETY CRITERIA IN NEA MEMBER COUNTRIES
March 2003

Compilation of responses received from member countries

EXECUTIVE SUMMARY

In 2001 the Committee on the Safety of Nuclear Installations (CSNI) issued a report on Fuel Safety Criteria Technical Review [1]. The objective was to review the present fuel safety criteria and judge to which extent they are affected by the “new” design elements, such as different cladding materials, higher burnup, the use of MOX fuels, etc. The report stated that the current framework of fuel safety criteria remains generally applicable, being largely unaffected by the “new” or modern design elements. The levels (numbers) in the individual safety criteria may, however, change in accordance with the particular fuel and core design features. Some of these levels have already been – or are continuously being – adjusted. The level adjustments of several other criteria (RIA, LOCA) also appears to be needed, on the basis of experimental data and the analysis thereof.

As a follow-up, among its first tasks, the CSNI Special Expert Group on Fuel Safety Margins (SEG FSM) initiated the collection of information on the present fuel safety criteria used in NEA member states with the objective to sollicit national practices in the use of fuel safety criteria, in particular to get information on their specific national levels/values, including their recent adjustments, and to identify the differences and commonalties between the different countries.

Two sources of information were used to produce this report: a compilation of responses to a questionnaire prepared for the June 2000 CNRA meeting, and individual responses from the SEGFSM members to the new revised questionnaire issued by the task Force preparing this report. In accordance with the latter, the fuel safety criteria discussed in this report were divided into three categories:

(A) safety criteria – criteria imposed by the regulator;
(B) operational criteria – specific to the fuel design and provided by the fuel vendor as part of the licensing basis;
(C) design criteria – limits employed by vendors and/or utilities for fuel&core design.

Based on the responses submitted, the report provides a brief overview of the current fuel safety criteria used in NEA member countries along with the identification of the effects of “new” elements, if any. The main body of the report consists of separate tables for each criterion listed, followed by the synthesis of responses to the two above-mentioned questionnaires.

The comparison of national fuel safety criteria in NEA member countries shows that the basic set of “safety” criteria is quite similar. Also the specific levels of many of these criteria are practically identical, e.g. for the LOCA criteria. However, in some cases the criteria levels differ from country to country, in particular with the specific fuel and core design features and also due to different progress in the understanding of the impact of the “new” elements, in particular the effect of high burnup. A number of tables are empty as no information was provided for those particular criteria. There might be two explanations for this: the criteria are not used or they are the subject of proprietary information by fuel vendors.

The differences in criteria levels indicate the areas where further international co-operation would be worthwhile in order to get a better understanding of the reasons for national differences and/or to contribute to better harmonization in criteria levels among the NEA member countries. These areas will be the focus of future SEFG FSM activities.

Table of Contents

FUEL SAFETY CRITERIA IN NEA MEMBER COUNTRIES
March 2003

Compiled from member countries' contributions

I. Introduction

In 2001 the Committee on the Safety of Nuclear Installations (CSNI) issued a report containing a technical review of nuclear fuel safety criteria [1]. This review was performed by an expert task force, with the objective to assess the possible consequences of "new" design elements such as different cladding materials, higher burnup, the use of MOX fuels, etc. on present fuel safety criteria. The report concluded that the current framework of fuel safety criteria remains generally applicable, being largely unaffected by the 'new' or modern design elements. The levels (numbers) of the individual safety criteria may, however, change in accordance with the particular fuel and core design features. Some of these levels have already been - or are continuously being – adjusted. Level adjustments of several other criteria (RIA, LOCA) also appear to be needed, on the basis of experimental data and the analysis thereof.

As a follow-up, among its first tasks, the CSNI Special Expert Group on Fuel Safety Margins (SEG FSM) initiated the collection of information on the present fuel safety criteria used in NEA member states with the objective to solicit national practices in the use of fuel safety criteria, in particular to get information on their specific national levels/values, including their recent adjustments, and to identify any differences and commonalities between the different countries. The present report contains the result of this activity.

Two sources of information were used to produce this report: a compilation of responses to a questionnaire prepared for the June 2000 CNRA meeting, and individual responses from the SEGFSM members to a further (more extensive) questionnaire issued by the SEGFSM Task Force responsible for this activity.

In accordance with the latter questionnaire, the fuel safety criteria discussed in this report are divided into three categories:

(A) safety criteria

(B) operational criteria

(C) design criteria

The following table lists all fuel safety criteria in each of the three categories:

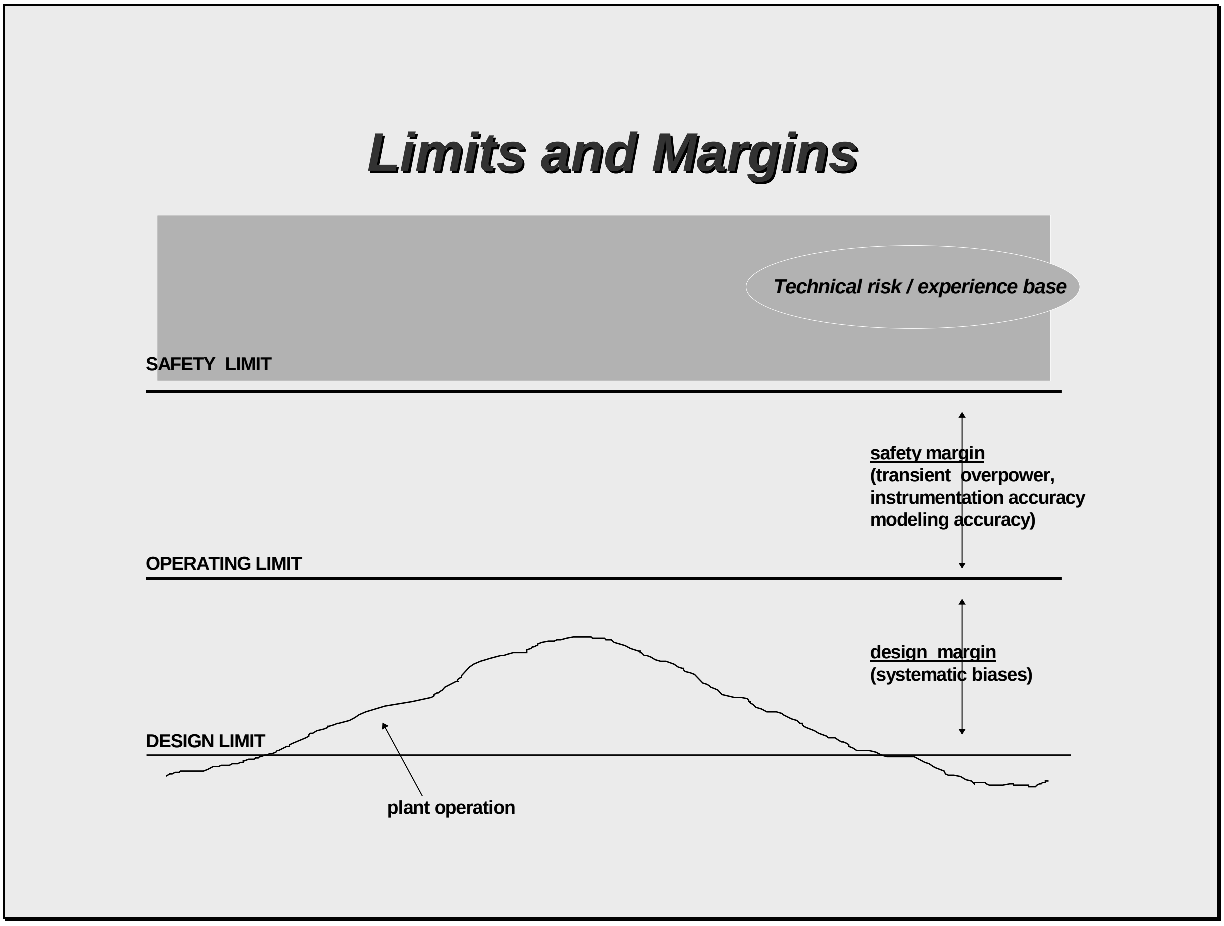
Safety criteria	***Operating criteria***	***Design criteria***
DNB safety limit	**DNB operating limit**	**Crud deposition**
Reactivity coefficients	**LHGR limit**	**Stress / strain / fatigue**
Shutdown margin	**PCI**	**Oxidation**
Enrichment	**Coolant activity**	**Hydride concentration**
Internal gas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

The first category includes all 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 second category includes operational criteria, some of which are derived from the category A criteria, covering 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.

The third category includes design limits, mostly not approved by the regulator, that are part of the design basis for the fuel with the aim to be able to meet the second or first category criteria.

The relationship between these three categories is symbolized in the following picture:



Based on the responses submitted, this report provides a brief overview of the current fuel safety criteria used in NEA member countries along with the identification of any effects of “new”elements to date. The main part of the report is constituted by separate tables for each individual criterion in all three categories; the content of these tables is based on a synthesis of responses to the two above-mentioned questionnaires.

2. Criteria tables

A. Safety Criteria

DNB/CPR SAFETY LIMIT

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements" so far	Type of methods and rational
Belgium	CHF/DNB	Depending on the correlation used	95/95, correlations: W3 (for low pressure), WRB-, HTP, ERB-, FC, ABBX-, ...depending on the fuel supplier	DNB operating limit	No change, verification required (new design, mixed cores)	Statistical
Canada	CHF		correlations (e.g. Balint-Cheng)			
Czech Rep.	DNB	1,125-1,274 VVER440 1,30 VVER1000	95/95, Russian and Czech correlations.for VVER440, Westinghouse correlation for VVER1000	3-D peaking at VVER440, DNB oper. limit at VVER1000,	Verification required	Statistical
Finland	DNB/CPR	1.33/1.06	95/95 / <0.1% of rods may experience DNB, correlations		No change (burnup limit 40 GWd/t)	Statistical
France	CHF	1.17, 1.30	95/95, correlations (WRB-, W-3 for low pressure)	operat. Limits (e.g. axial offset)	No change, verif. required (new design, mixed core)	Statistical
Germany	CHF		95/95-correlations (PWR), <1 rod experience dryout – THAM method (BWR), all corrlelations are FA specific	Addit. operat. crit.	Values change dep. on design	Statistical
Hungary	DNB	1.33	95/95, correlations (Bezrukov)	Pin power limit	Values change dep. on design	Statistical
Japan	DNB/CPR	1.17*/1.06*	95/95 / <0.1% of rods may experience DNB, correlations (e.g. MIRC-1, NFI-1)	DNB/CPR oper.lim.	No effect	Statistical
Korea	DNB	1.17-1.30	95/95, correlations (WRB-1, W-3)	DNB oper.limit	No change, verif. required	Statistical
Netherlands	DNB	1.30	95/95, correlations (W-3)	DNB oper.limit		Statistical
Spain	DNB/CPR	1.15/1.08	95/95, correlations (e.g. W-2, WRB-1)	DNB/CPR oper.lim.	Values change dep. on design	Statistical
Sweden	DNB/CPR	1.17/1.06	95/95, correlations (VRB-1)<0.1% of rods may experience dryout	DNB/CPR oper.lim.	Values change dep. on design	Statistical
Switzerland	DNB/CPR	1.10/1.40	95/95, correlations	DNB/CPR oper.lim.	Values change dep. on design	Statistical
UK	DNB		95/95, correlations	DNB oper.limit		Statistical
USA	DNB	various	95/95, correlations	DNB oper.limit		Statistical

*not criteria but typical value

REACTIVITY COEFFICIENT

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements" so far	Type of methods and rational
Belgium	Overall negative reactivity		Prompt inherent feedback	ITC must be negative, MTC must be lower than |DTC|		Conservative
Canada						
Czech Rep.	Overall negative reactivity		Prompt inherent feedback	MTC must be less than zero	Unaffected	Best-estimate
Finland	Overall negative reactivity		Prompt inherent feedback	Limit for isothermal coeff.		Best-estimate
France	Overall negative reactivity		Prompt inherent feedback	MTC must be negative		
Germany	Overall negative reactivity			MTC must be negative with exception for short term	unaffected	Best-estimate
Hungary	Overall negative reactivity		Prompt inherent feedback	Operational range for each coefficient in TechSpec	Unaffected	Conservative
Japan	Overall negative reactivity		Prompt inherent feedback		Unaffected	Best-estimate
Korea	Overall negative reactivity		Prompt inherent feedback	Individual coefficients can be positive	Unaffected	Statistical (?)
Netherlands						
Spain	Overall negative reactivity		Prompt inherent feedback	MTC can be positive at BOL, limit values in TechSpec	Unaffected	Conservative
Sweden	Overall negative reactivity		Prompt inherent feedback	ITC can be positive below operating temperature	New BWR fuel designs 10x10 has restrictions for nuclear heating of the reactor	Best-estimate
Switzerland	Overall negative reactivity		Prompt inherent feedback			Conservative
UK						
USA						

SHUTDOWN MARGIN

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements" so far	Type of methods and rational
Belgium	Boron, control rod worth	1,77%, 2%, 2.7%, 2.9% depending on the power plant	Attain subcriticality			Conservative
Canada	Boron, control rod worth		Attain subcriticality			
Czech Rep.	Boron, control rod worth	2% VVER440, 1,3%VVER1000	Attain subcriticality		Unaffected	Conservative
Finland	Control rod worth	1%	Attain subcriticality		Unaffected	Best-estimate
France	Boron, control rod worth	2500/21000ppm, 1.8%	Attain subcriticality		Fuel & core design dependent	Conservative
Germany	Boron, control rod worth	1%	Attain subcriticality		Unaffected	Conservative
Hungary	Boron, control rod worth	2%	Attain subcriticality			
Japan	Control rod worth		Attain subcriticality		Unaffected	Best-estimate
Korea	Control rod worth	2%	Attain subcriticality		Unaffected	Conservative
Netherlands	Control rod worth	1%	Attain subcriticality			
Spain	Boron, Control rod worth	1%	Attain subcriticality		Unaffected	Conservative
Sweden	Boron, control rod worth	2500ppm, 1%	Attain subcriticality		Unaffected	Best-estimate
Switzerland	Boron, control rod worth	2500ppm, 1%	Attain subcriticality		Fuel & core design dependent Nat. and enriched B in use	Conservative
UK						
USA	No criterion (??)					

ENRICHMENT

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new element“ so far	Type of methods and rational
Belgium	Explicit limit in some Royal Decrees	5%	Based on manufacturing	Licensed fuel up to 4.35%; (licensing up to 4.6%)		
Canada						
Czech Rep.	No explicit limit for VVER440 Fuel design limit for VVER1000	5%		LICENSED VVER440 FUEL UP TO 4,4%		
Finland	No explicit limit			Licensed fuel up to 4%		
France	Manufacturing	5%		Licensed fuel up to 4.2%		
Germany	No explicit limit			Licensed fuel upto 4.6%		
Hungary						
Japan	Manufacturing	5%		Licensed fuel up to 5%		
Korea	Fuel design limit	5%	Based on manufacturing			
Netherlands						
Spain	No explicit limit			Fuel licensed up to 5%		
Sweden	No explicit limit		Based on manufacturing	Licensed fuel up to 4,0% for PWR and 4,9% for BWR		
Switzerland	No explicit limit					
UK						
USA						

There is a general enrichment limit for fuel manufacturing of 5 wt%. This limit is linked to issues pertaining also to transportation and storage; an increase of this value does not appear feasible at this point in time.

INTERNAL GAS PRESSURE

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Pressure	184 bar	P_{gas} can be > P_{sys}: basis is no gap reopening		Review of criterion value necessary for higher burnup	Conservative: partially statistical/ partially deterministic
Canada						
Czech Rep.	Pressure, Lift-off		Pgas<Psys for VVER440; lift-off for VVER1000		Review necessary in view of new designs, high burnup	
Finland	Pressure		Pgas<Psys			
France	Pressure	Approx.180 bar	Pgas can be > Psys: basis is no gap reopening		Yes, review (increase) of criterion value necessary (high burnup, new fuel & core designs)	
Germany	Lift-off		No lift-off during life-time			
Hungary	Pressure	Approx. 122 bar	Pgas<Psys for VVER440			
Japan	Lift-off		No pellet/clad gap increase			
Korea						
Netherlands						
Spain	Pressure, lift-off		Pgas<Psys		To be reviewed	
Sweden	Lift-off		No pellet/clad gap increase		Review needed	
Switzerland	Lift-off		No pellet/clad gap increase		Review needed	
UK						
USA						

PCMI

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Strain		1% elastic+plastic hoop strain or 2.5% equiv. plastic strain depending on fuel supplier		Verification only	Conservative
	Stress	Minimum failure threshold	Experimentally based (ramp tests)			
Canada						
Czech Rep.	Stress, strain		Hoop stress < yield strength with additional safety factor, 1% elastic+plastic for Zircaloy		Only values affected (materials)	Conservative
Finland	No explicit limit		Licensee must show low probability for PCMI failure			
France	Strain		1% elastic+plastic hoop strain		Verification only	Conservative
Germany	Limiting power change		experimentally based (ramp tests)		Design dependent	Conservative
Germany	Strain		2.5% equiv. plastic strain		No	Conservative
Hungary	No limit(?), strain		For BNFL fuel 1% strain limit			
Japan	Strain		1% elastic+plastic hoop strain		No	Conservative
Korea	Strain		1% elastic+plastic hoop strain		No	Conservative
Netherlands						
Spain	Strain		1% elastic+plastic hoop strain		No	Conservative
Sweden	No explicit limit		Licensee must show low probability for PCMI failure			Best estimate and conservative
Switzerland	Strain		2.5% equiv. plastic strain			Conservative
UK						
USA						

RIA FRAGMENTATION

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Radially ave. fuel enthalpy increase	225 (fresh), 200 (burnt)	Experimental (US, CABRI, NSRR)	RIA fuel failure		Conservative
Canada	No					
Czech Rep.	Radially ave. fuel enthalpy increase	230 VVER440, 200 VVER1000	Experimental (Russia, US)		Future verification needed	Conservative
Finland	Radially ave. fuel enthalpy increase	230	Experimental		No effect so far (burnup limited to 40 GWd/t)	Conservative
France	Radially ave. fuel enthalpy increase	225 (fresh), 200 (burnt)	Experimental (US, CABRI, NSRR)	RIA fuel failure	Criterion has been ‚replaced‘ by RIA fuel failure limit for high burnup (safety domain)	Conservative
Germany	Radially ave. Fuel enthalpy increase		Enthalpy rise less than experimentally determined failure limit			
Hungary	Radially ave. fuel enthalpy increase	230	Experimental (US, Russia)		No	
Japan	Radially ave. fuel enthalpy increase	230 – X	Experimental (US, Japan); X refers to decrease of melting point vs. Burnup	RIA fuel failure	Yes	Conservative
Korea	Radially ave. fuel enthalpy increase	280 (CE), 225/200 for W	See France		No	Conservative
Netherlands						
Spain	Radially ave. fuel enthalpy increase	280	Experimental	RIA fuel failure		Conservative
Sweden	Radially ave. fuel enthalpy increase	230 – 30 cal/g	Experimental, burnup dependent		No	Conservative
Switzerland	Radially ave. fuel enthalpy increase	no fuel melting	Experimental (fuel failure limit is governing)	RIA fuel failure		
UK						
USA	Radially ave. fuel enthalpy increase	280				

NON-LOCA RUNAWAY OXIDATION

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.	No		1200 C used for limiting cladding temp. in accident conditions			
Finland	No		1200 C used for limiting cladding temp. in accident conditions			
France	Temperature	2700 F	Experimental			
Germany	no		1200 C used for limiting cladding temp. in accident conditions			
Hungary	No		1200 C used for limiting cladding temp. in accident conditions			
Japan	No					
Korea	Temperature	2700 F	Experimental			
Netherlands						
Spain	Temperature	2700 F	Experimental			
Sweden	No					
Switzerland	Temperature	2700 F	Experimental			
UK						
USA						

LOCA PCT

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational (*)
Belgium	Temperature	1204 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	Conservative
Canada	Temperature	1760 C				
Czech Rep.	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Finland	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
France	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	Realistic, deterministic
Germany	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Hungary	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Japan	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Korea	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Netherlands	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Spain	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Sweden	Temperature	1204 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
Switzerland	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
UK	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	
USA	Temperature	1200 C	Experimental, embrittlement, high temperature oxidation	LOCA oxidation	To be verified!	

(*) Conservative models and assumptions (ref. Appendix K requirements) are still widely used for licensing calculations. However, more realistic (best-estimate) methods have become available for benchmarking and evaluating safety margins.

LOCA OXIDATION

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
Canada	No					
Czech Rep.	ECR	17% VVER1000 18% VVER440	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
Finland	Quantitative		Prevent excessive embrittlement	LOCA PCT		
France	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
Germany	ECR	17%	Experimental, embrittlement, transient oxidation	LOCA PCT	To be verified!	Conservative
Hungary	ECR	17%	Experimental, embrittlement	LOCA PCT	To be verified!	Conservative
Japan	ECR	15%	Experimental, embrittlement, related to no occurrence of significant oxidation (?)	LOCA PCT	To be verified!	Conservative
Korea	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
Netherlands	ECR	17%	Experimental, embrittlemen, transient oxidation	LOCA PCT	To be verified!	Conservative
Spain	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
Sweden	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
Switzerland	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
UK	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative
USA	ECR	17%	Experimental, embrittlement, total oxidation	LOCA PCT	To be verified!	Conservative

LOCA HYDROGEN RELEASE

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Canada	No					
Czech Rep.	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Finland	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
France	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Germany	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Hungary	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Japan	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Korea	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Netherlands	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Spain	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Sweden	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
Switzerland	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
UK	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative
USA	H release	1%	Containment integrity	LOCA oxidation	Unaffected	Conservative

LOCA LONG TERM COOLING

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
Canada						
Czech Rep.	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
Finland	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
France	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.	To be checked	
Germany	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
Hungary						
Japan	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
Korea	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
Netherlands						
Spain	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
Sweden	Qualitatively		Stable state after accident, decay heat removal		Not affected	Conservative
Switzerland	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		
UK						
USA	Qualitatively		Stable state after accident, decay heat removal	ECCS design req.		

SEISMIC LOADS

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Loads, forces	?	Control rod insertability, coolability			
Canada						
Czech Rep.	Loads, forces	Defined in SAR for NPP site	Control rod insertability, coolability		Verification required	Conservative
Finland	Not relevant		(Seismically stable)			
France	Loads, forces	?	Control rod insertability, coolability			
Germany	Strain, forces in FA skeleton		No residual deformation of FA skeleton		Limits do not apply to fuel cladding	
Hungary	Loads, forces	?	?			
Japan	Not applicable					
Korea	Loads, forces	?	Control rod insertability, coolability			
Netherlands						
Spain	Loads, forces	?	Control rod insertability, coolability			
Sweden	Not relevant		Seismically stable			
Switzerland	Loads, forces	depends on component design	Control rod insertability, coolability			
UK						
USA						

HOLD-DOWN FORCE

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events, with the exceptionof the pump overspeed transient, by holddown spring design		Unaffected	
Canada						
Czech Rep.	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events by holddown spring design		Unaffected	Conservative
Finland	Not relevant					
France	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events by holddown spring design		Unaffected	Overall conservative
Germany	Qualitative		Prevent hydraulic assembly lift-off under normal operation		Unaffected	Conservative
Hungary	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events by holddown spring design			
Japan	Not applicable					
Korea	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events by holddown spring design			Conservative or statistical
Netherlands						
Spain	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events by holddown spring design			Conservative
Sweden	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events by holddown spring design			Conservative
Switzerland	Qualitative		Prevent hydraulic assembly lift-off for condition I and II events by holddown spring design		Unaffected	Conservative
UK						
USA						

CRITICALITY

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Margin to criticality	2% or 5% with BU credit allowed for storage 5% for fabrication and transportation	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Canada						
Czech Rep.	Margin to criticality	5%, 2% optimum moderation	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Finland	Margin to criticality	5%	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
France	Margin to criticality	5%	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Germany	Margin to criticality	5%	Avoid criticality for fabrication, transportation and storage		Burn-up credit under evaluation	
Hungary	Margin to criticality	5%, 2% with single failure	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Japan	Margin to criticality		Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Korea	Margin to criticality	5%, 2% optimum moderation	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Netherlands						
Spain	Margin to criticality	5%, K<1 with boron credit	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Sweden	Margin to criticality	5% norm. op. 2% accid. cond.	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
Switzerland	Margin to criticality	5%	Avoid criticality for fabrication, transportation and storage		Unaffected	Conservative
UK						
USA						

BURNUP

Country	Criterion type	Value(s)*	Basis	Relation to other criteria	Effect of „new elements“	Type of methods and rational
Belgium	UO_2/MOX Ave.assy	55/50				
Canada	Ave.assy	20				
Czech Rep.	Ave. rod	60	VVER1000, Westinghouse fuel			
	Ave.assy/ ave.rod/ /peak pellet	53/58/64	VVER440, RUSSIAN FUEL			
Finland	Ave.assy	45				
France	Ave.assy	52	3 cycles limit for MOX			
Germany	No	Status: 55* (PWR/BWR) future 65* (PWR/BWR)	FA averaged values			
Hungary	Ave.rod	60/55	BNFL/Russian			
Japan	Ave.assy	48-55/40-45	UO_2/MOX			
Korea	Ave.rod	60/58	W/CE			
Netherlands	Ave.rod/ave.assy	60/55				
Spain	Peak pellet, rod average	Fuel dependent				
Sweden	Various					
Switzerland	Peak pellet/ave.rod/ave.assy	75/65/60				
UK	Peak pellet	55				
USA	Ave.rod	62				

* in GWd/t

B. Operational / licensing criteria

DNB/CPR OPERATING LIMIT

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	FDH	1.62-1.64 depending on the power plant		DNB safety limit		Conservative
Canada						
Czech Rep.	DNB	None VVER440 1,262VVER1000	SAFETY LIMIT + UNCERTAINTIES MODELS, MEASUREMENTS + MARGIN ACC. TO WORST TRANSIENT	DNB safety limit		
Finland	DNB/CPR	None/1.30	Safety limit + uncertainties models, measurements + margin acc. to worst transient	DNB/CPR safety limit, radial peaking factor	Unaffected	Conservative and statistical
France	DNB		Safety limit + uncertainties models, measurements + margin acc. to worst transient	CHF safety limit	Fuel & core design dependent	Conservative or statistical
Germany	Ave.assy	52-57				
Hungary	No		Covered by 3D radial peaking limit			
Japan	DNB/CPR	?/1.21*	Safety limit + uncertainties models, measurements + margin acc. to worst transient	DNB/CPR safety limit	Unaffected	Conservative and statistical
Korea	DNB	1.3-1.5	Safety limit + margin acc. to worst transient	DNB safety limit	Unaffected	Conservative and statistical
Netherlands	DNB					
Spain	DNB/CPR	Fuel and plant dependent	Safety limit + uncertainties models, measurements + margin acc. to worst transient	DNB/CPR safety limit	Unaffected	Conservative and statistical
Sweden	DNB/CPR	1.3 - 1.5	Safety limit + uncertainties models, measurements + margin acc. to worst transient	DNB/CPR safety limit		
Switzerland	DNB/CPR	1.2 - 1.8	Safety limit + uncertainties models, measurements + margin acc. to worst transient	DNB/CPR safety limit		Conservative
UK	DNB					
USA	DNB/CPR					

*not criteria but typical value

LHGR OPERATING LIMIT

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 & mechanical overpower) for the following phenomena:

- fuel melting (*Note*: sometimes not calculated explicitly, while considered to be covered by the 1% strain criterion)
- rod internal pressure, fission gas release
- stress, strain, fatigue
- PCMI stress

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.

PCI LIMIT

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Re – and de – conditioning	Max. 28 days at low power (< 85%Pn)	Stress limit in condition II transients			
Canada	Limiting local power change		Experimentally based correlations			
Czech Rep.	Limiting power change, re- and de- conditioning		Experimentally based (ramp tests), for condition I and II	LHGR	Design dependent	Conservative
Finland	Limiting power change		Experimentally based			
France	Limiting power change, re- and de- conditioning		Experimentally based (ramp tests), for condition I and II	LHGR, axial offset	Design dependent	Conservative
Germany	Limiting power change		Experimentally based (ramp tests)		Design dependent	
Hungary	No					
Japan	No					
Korea	No					
Netherlands	Conditioning rules		Experimentally based			
Spain	(Pre)conditioning rules		Experimentally based		Design dependent	Conservative
Sweden	Limiting power change		Experimentally based (ramp tests)		Design dependent	
Switzerland	Limiting power change		Experimentally based (ramp tests)		Design dependent	Conservative
UK	Limiting power change, re – and de – conditioning		Experimentally based (ramp tests), for condition I and II			
USA	No					

COOLANT ACTIVITY

Country	Criterion type	Value(s)*	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Activity I-131		Limit release after fuel failures.			
Canada						
Czech Rep.	Activity I-131	3,7	Limit release after fuel failures			
Finland						
France	Activity I-131		Limit release after fuel failures.			
Germany	Activity I-131	Plant specific	Limit release after fuel failures. Lower limit: intensified measuring, higher limit: plant shutdown			
Hungary	Activity I-131	3.7	Limit release after fuel failures			
Japan						
Korea						
Netherlands						
Spain	Activity I-131		Limit release after fuel failures			
Sweden						
Switzerland	Activity I-131	0.1	Limit release after fuel failures.			
UK						
USA						

In 10^{10} Bq/t

GAP ACTIVITY

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany	No					
Hungary						
Japan						
Korea						
Netherlands						
Spain	No limit					
Sweden						
Switzerland	No limit/criterion					
UK						
USA						

SOURCE TERM

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany	No					
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland	No criteria defined				considerable design dependence	
UK						
USA						

CONTROL ROD DROP TIME

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Rod drop time	1.35s – 3.2s depending on the core height				
Canada						
Czech Rep.	Rod drop time	8 -13s VVER440 3,5s VVER1000	Reactor design		None	
Finland						
France						
Germany	Drop time	< 4 s to reach end position, < 2 s for the average	Assure safe shut down		Minimum drop time determined by accident analysis	
Hungary						
Japan						
Korea						
Netherlands						
Spain	Drop time to dashpot (PWR)/ to 75% insertion (BWR)	Plant and fuel dependent	Assure safe shutdown		Minimum drop time determined by accident analysis	
Sweden	Drop time	1.5 – 4 sec.	Assure safe shut down (TechSpec)			
Switzerland	Time to dashpot entry (PWR) / to 75% insertion (BWR)	1.5-2.5 sec	Rod insertability (TechSpec)		Larger insertion times observed in newer designs	
Switzerland						
UK						
USA						

RIA FUEL FAILURE LIMIT

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Radially ave. Fuel enthalpy increase, DNB	225 (fresh), 200 (burnt)	Experimental (US, CABRI, NSRR)	RIA fuel fragmentation , DNB		Conservative
Canada	No					
Czech Rep.	Enthalpy rise No - VVER1000	140cal/g VVER440	Experimental	DNB, RIA FUEL FRAGMENTATIO N	New fuel and core design (cladding material, increased burn up,..). Limit under development	Conservative (3D kinetic)
Finland	Enthalpy rise	140cal/g VVER440				
France	No fuel failure > 47 GWd/t ave.assy. (safety domain)		Experimental (CABRI, NSRR) Safety domain (peak fuel enthalpy, enthalpy rise, corrosion limit, no spallation...)	RIA fuel fragmentation	New fuel and core design (cladding material, increased burn up,..)	Conservative (3D kinetic)
Germany	No			RIA fuel fragmentation		Conservative (3D kinetic)
Hungary	Enthalpy rise	140 cal/g (average)	Experimental	DNB, RIA fuel fragmentation		
Japan	Enthalpy rise	Burnup dep.	2 Limits: PCMI & rod burst Experimental	RIA fuel fragmentation	Unaffected	Conservative
Korea	No					
Netherlands	Enthalpy rise	Burnup dep.	Experimental	RIA fuel fragmentation		
Spain	Enthalpy increase, DNB (PWR)	170 cal/g (BWR)	Experimental	RIA fuel fragmentation		Conservative
Sweden	Enthalpy increase	Burnup dep.	Experimental	RIA fuel fragmentation		
Switzerland	Enthalpy increase	Burnup dep. („Swiss curve“)	Experimental, burnup dependent			Conservative
UK						
USA	Enthalpy increase, DNB (PWR)	170 cal/g (BWR)		RIA fuel fragmentation		

CLADDING STABILITY

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

C. Design Criteria

CRUD DEPOSITION

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany	No					
Hungary						
Japan						
Korea						
Netherlands						
Spain	No criterion					
Sweden						
Switzerland	No limit/criterion					
UK						
USA						

STRESS / STRAIN / FATIGUE

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Stress / Strain / Fatigue		Stress < yield stress / 1% hoop strain / No failure			
Canada						
Czech Rep.	Stress		Stress < standard yield strength Cladding OD change < 0.45%			
Finland						
France	Stress / strain / fatigue		Below loss of integrity / 1% / 0.8%			
Germany	Stress / strain / fatigue		< yield stress / < 2.5% equivalent creep strain / < 1 degree of fatigue			
Hungary						
Japan	Stress, strain		Stress < yield stress; 1% hoop strain			
Korea						
Netherlands	Strain		1% total clad strain			
Spain	Stress, strain		1% total clad strain			
Sweden						
Switzerland	Strain		1% total clad strain			
UK						
USA	Stress, strain, fatigue		Ref. ASME Section III			

OXIDATION, HYDRIDING

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium	Oxide thickness, hydride conc.	100microns, 600ppm	Prevent steep increase in spalling and excessive hydriding			
Canada						
Czech Rep.	Oxide thickness	60microns	Prevent steep increase in spalling and excessive hydriding			
Finland						
France	Oxide thickness, hydride conc.	100microns, 600ppm	Prevent steep increase in spalling and excessive hydriding			
Germany	Oxide thickness, hydride conc.	Defect probability integral < 1 rod/cycle, 500ppm H	No rod failure due to corrosion			
Hungary						
Japan	No criteria					
Korea						
Netherlands	Oxide thickness, hydride conc.	100microns, 500ppm	Prevent steep increase in spalling and excessive hydriding			
Spain	Oxide thickness, hydride conc.	100microns, 600ppm				
Sweden						
Switzerland	Oxide thickness, hydride conc.	100microns, 500ppm	Prevent steep increase in spalling and excessive hydriding			
UK	Oxide thickness, hydride conc.		Should be limited			
USA						

HYDRIDE CONCENTRATION

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

TRANSPORT LOADS

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

FA FRETTING CORROSION/WEAR

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

CLAD DIAMETER INCREASE

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

CLADDING ELONGATION

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements" so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

RADIAL PEAKING FACTOR

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

3D PEAKING FACTOR

Country	Criterion type	Value(s)	Basis	Relation to other criteria	Effect of „new elements“ so far	Type of methods and rational
Belgium						
Canada						
Czech Rep.						
Finland						
France						
Germany						
Hungary						
Japan						
Korea						
Netherlands						
Spain						
Sweden						
Switzerland						
UK						
USA						

.

3. Summary and Conclusions

The data, obtained from various NEA member countries as a response to the questionnaires as mentioned in the introduction has been condensed in the above tables, which to a large extent are self-explanatory. The different safety criteria are listed together with their respective values / levels and any effect of 'new' design elements in the past; also, the basis for the criterion definition is listed in brief. In some instances, no information is included; this is partly due to the fact that not each individual member country provided a complete response, and therefore does not necessarily imply that no such criteria (or values / effects from new design elements) exist. It is assumed that the proprietary nature of some of the information has, to a degree, contributed to such incomplete responses.

When examining the tables and comparing the data therein, it is easily observed that many countries indeed observe most of the fuel safety criteria. Thus, it may be concluded that, generally, fuel safety is well covered in NEA countries by applying an adequate set of criteria. Also the specific values or levels of many of these criteria are often quite similar if not practically identical; this is, for instance, the case for the LOCA cladding embrittlement/oxidation criteria. However, in some cases criteria levels differ from country to country, in concord with the different specific fuel types and core design features and, in addition, due to different progress in dealing with the impact of new design elements – in particular the effect of high burnup. Furthermore, differences in regulatory perception (e.g. amount of conservatism applied for limits definition) may play an important role here.

The differences in criteria as applied in the member countries indicate areas where further international co-operation would be worthwhile in order to get a better understanding of the reasons for these differences and, possibly, make the criteria set more uniform. Likewise, the variation in criteria levels could be reduced via harmonization among the NEA member countries. These issues will be the focus of future SEFG FSM activities.

References.

[1] OECD / NEA report 'Nuclear Fuel Safety Criteria Technical Review', ISBN 92-64-19687-0.

Appendix I

COMPILATION OF THE NATIONAL RESPONSES TO THE CNRA QUESTIONNAIRE ON FUEL SAFETY CRITERIA

Responses from: Belgium, Canada, Czech Republic, Finland, France, Germany, Hungary, Japan, Korea, Netherlands, Spain, Sweden, Switzerland, UK, USA

December 2000

FUEL SAFETY CRITERIA

1. Normal Operation

1.1 Critical heat flux (CHF)

Please state the limit for critical heat flux if this is used in safety evaluations in your country. Please specify to which event category the limit is applicable.

Canada: Various CHF correlations are being used (Balint-Cheng, etc.). The correlations are derived from sliding thermocouples in fuel bundle simulators that have prototypic non-uniform radial and axial power shapes. New correlations are being developed for crept pressure tubes. The CHF limit is used for loss of regulation, small loss of coolant, loss of normal electrical power, and pump seizure as the criterion for avoiding failure analysis of the fuel and fuel channel.

Czech Republic: No

Finland: No

France: The design bases of the RCC-C (Design and Construction Rules for Fuel Assemblies of PWR Nuclear Power Plants) and safety reports require that cladding temperature shall not significantly exceed coolant temperature, thus ensuring that rod integrity is not lost through overheating or accelerated corrosion.

To meet this requirement, the minimum CHF ratio has to be evaluated for conservative reactor operating conditions. The calculated critical heat flux value shall not exceed the value of the criterion, which depends on the correlation used. There must be at least a 95% probability at a 95% confidence level that critical heat flux will not be reached in limiting core rods.

Germany: The safety requirements of the KTA rule 3101 requires to avoid CHF occurrence or to limit fuel cladding temperatures to acceptable values. This takes into account that the CHF-ratio is an early indicator of insufficient fuel rod cooling which will not imply directly fuel rod failure. Fuel rod failure would occur if the fuel cladding temperatures reach high values.

Therefore the CHF-correlation applied for the determination of the CHF-limits should be based on experimental data for the specific fuel assembly design. The statistical evaluation of these data should define the 95/95 – (fraction/confidence) – limit value of the CHF correlation.

The necessary margin for transients is determined by the thermal design. Usually statistical thermal design methods are applied to consider technological and operational uncertainties. These methods differ in the number or type of parameters which are still treated in a deterministic manner. This approach applies to the subsequent items of DNBR for PWRs and CPR for BWRs.

Therefore no single value can be specified as reference value for normal operation.

Hungary: No

Japan: No

Korea: Margin should be sufficient to prevent fuel rod from having CHF during normal operation and anticipated operational transients.

Netherlands: No

Spain: No

Sweden: No

Switzerland: No

UK: The requirement is to have at least two lines of protection against CHF. Various CHF correlations are being used. These are derived from CHF experiments over a range of typical reactor and accident conditions and axial power shapes. For very low pressures, OEF lookup tables based on tube data are used. See also section 1.2.

USA: No. Items 1.2 and 1.3 cover this.

1.2 Departure from nucleate boiling ratio (DNBR), for PWRs.

Please specify the limit for DNBR, if this is a safety criterion in your country. Please specify to which event category the limit is applicable. Please specify how the DNBR safety limit is established, and the underlying basis (correlations).

Belgium: There must be at least a 95% probability at the 95% confidence level that DNB will not occur on the limiting fuel rods during normal operation and incidents of moderate frequency (condition I & II events). For condition III events, the objective is to avoid more than 5 % of fuel rods entering in DNB and for condition IV events, this objective becomes 10 %.

DNB correlations are fuel design-dependent. Specific tests are performed to establish these correlations. Typical limit value of minimum DNBR is 1.18 with HTP CHF correlation.

Canada: Canada has no PWRs.

Czech Republic: The limit for DNBR is a safety criterion in Czech republic. The limit is applicable to event categories:

Condition I – Normal operational and operational transients
Condition II – Faults of moderate frequency

DNBR safety limits are higher than the design limit DNBR values (the difference results in available DNBR margin. The design limits DNBR are based on used DNBR correlations and the used thermal design procedures.). Numerical values are different depending on the correlation, database and statistical evaluation. The design limits DNBR values are equal or higher (depends on the used thermal design procedures) than DNBR correlation limits.

Finland: The adequate cooling of the cladding shall be ensured. It will be reached, if there is a 95 % probability, at the 95 % confidence level that the hottest fuel rod does not reach the heat transfer crisis or transition boiling condition. The number of rods reaching the heat transfer crisis shall not exceed 0.1% of the total number of fuel rods in the reactor.

France: There must be at least a 95 % probability at the 95 % confidence level that DNB will not occur on the limiting fuel rods during normal operation and incidents of moderate frequency (condition I & II events).

DNB correlations are fuel design-dependent. Specific tests are performed to establish these correlations. Typical limit value of minimum DNBR is 1.17 with WRB1 CHF correlation.

Germany: There exists no single value for the DNBR for normal operation. The statistical approach described under 1.1 is applied. Experimental data gained from the specific fuel rod bundle tests including relevant design features (spacer grids with or without vanes) are used for the determination of the 95/95 – limit value.

Sufficient margin to this DNB limit is provided for normal operating condition when the probabilistic thermal hydraulic analysis for a Loss of Flow Event (LOFE) shows that the rods will not experience DNB related to the specific 95/95 – limit-value. The resulting minimum DNB-ratio is monitored during operation by the power density limitation system (RELEB).

Hungary: For AOOs DNBR is at least 1.33 (Bezrukov correlation)

Japan:

(1) Limit of minimum DNBR
Allowable limit of minimum DNBR is defined as the value at which DNB should not occur even in the severest fuel rod within the confidence level of 95 % reliability and 95 % probability (95x95 criterion).

As the allowable limit of minimum DNBR, 1.17 is commonly used in all PWR plants. The deviations of plant parameters from nominal values, prediction uncertainties of correlation equations and etc. are considered to determine the allowable limit.

(2) Event category
Minimum DNBR should be less than allowable limit for normal operation and abnormal transient.

(3) DNB correlations and subchannel codes

The combinations of DNB correlation and subchannel code used in the evaluation are shown in the following table. The DNB correlations are the empirical equations developed from the test data which cover existing core conditions.

	DNB Correlation	Subchannel Code
Type-A Fuel	MIRC-1	THINC
Type-B Fuel	NFI-1	COBRA-3C

Korea: See Section 1.1

Netherlands: Only a related criterion is applied (by the calculation of the DNBR via the Westinghouse – 3 correlation are the hot-channel fuel rods conform the 95%/95% criterion of the KTA – rule-3101.1 for nucleate boiling protected when for normal operation a minimum allowable DNBR = 1.3.

For anticipated occurrences only limited DNBR violation is allowed; if DNB occurs cladding temperature lower than 600 0C

As a derived criterion for accident conditions (MSLB accident is limiting condition Postulated Initiating Event for this) shall be demonstrated that the hot channel fuel rods are protected against film boiling if DNBR >1.45 (100 bar < p < 140 bar)*).

**)note: the Westinghouse – 3 correlation is not applicable in this domain. The CHF–- table for KWU – fuel elements is used.*

Spain: There is a general criterion applicable for PWR reactors: fuel rod damage due to overheating of the clad will be avoided for C-I & II events. For C-III and C-IV fuel rods predicted to reach DNBR limits will be assumed to fail and should be included in the radiological dose calculations.
DNB-correlations are fuel design-dependent, for example:

- W – 3 correlation is still used for HIPAR & LOLOPAR fuel for an old one-loop W – design plant in operation in Spain;
- WRB – 1 correlation is used for the fuel in all other W-design plants (for fuel with Intermediate Flow-Mixers WRB-2 correlation is used instead);
- KWU/CHF correlation is used for the Siemens fuel.

DNBR-correlation limits are established to ensure there will be at least a 95 percent probability at a 95 % confidence level that boiling transition will not occur on the limiting fuel rods during condition I & II events. Limits values are considered to be proprietary information, but typically range from 1.15 to 1.2 and higher in case of very old correlations. However, it is important to notice that actual DNBR limit values for safety analyses differ from the limit values of the DNB correlations. This is so because analytical and surveillance uncertainties must be accounted for in the safety analyses. There are several methods to handle these uncertainties. Depending on the methodology employed to do that (i.e. the W-Standard Thermal-Design Process or the W-Revised Thermal-Design Process) the final safety limit applicable for the evaluation of a particular transient will be different. In addition to that, it is quite common that licensing analyses include extra-margins to cover anticipate or unanticipated issues that could have an impact in terms of DNB performance. These extra-margin play an important role, because the method-dependent DNBR safety limits are not recalculated every reload, and therefore an “extra-cushion” is needed to avoid unexpected shortcomings during plant operation.

Sweden: For departure from nucleate boiling ratio (DNBR) correlation's there should be a 95% probability at the 95% confidence level that the hot rod in the core does not experience a departure from nucleate boiling transition condition during normal operation or anticipated operational occurrences.

DNB-correlations are fuel design-dependent. Fuel design specific tests are performed to establish the correlation between critical power and parameters of influence. Limits values of the DNB correlations are typically in the range of 1.15 to 1,17. However, because analytical and surveillance uncertainties must be accounted for in the safety analyses, the actual DNBR limit values for safety analyses are higher. Usually in the range of 1,3 to 1,5.

For accident conditions fuel rods that are predicted to reach DNBR limits will be assumed to fail and should be included in the radiological dose calculations.

Switzerland: Fuel dependent safety limit (values from approx. 1.20. to 1.40), applies to Anticipated Operational Occurrences (AOO, cat. 2 events). Basis: DNB shall not occur (95% probability, 95% confidence) for limiting fuel rods. Fuel design specific test are performed to establish the correlation between DNB and parameters of influence.

UK: The requirement is to ensure adequate cooling such that DNB is avoided in normal operation and frequent faults (more frequent than 10-3/year) with 95% probability at a 95% confidence level. To this end, a safety DNBR limit is calculated via a statistical design procedure. The limit consists of the 95/95 correlation uncertainty and the 95/95 uncertainty for plant and fuel parameters, and - if required - a mixed core allowance. The limit is fuel dependent and hence no single limit is available**.**

Compliance with the limit is demonstrated in two ways (for frequent faults):

- The DNBR of the lead pin must not be less than the limit;
- The statistically expected number of rods in DNB in the core must not exceed 1.

The limit applies to all faults. (Note that faults less frequent than 10^{-3}/year are allowed to enter DNB (which is then assumed to commence at the DNBR limit), but must comply with fuel degradation limits, such as clad oxidation.)

USA: Different vendors use different numerical values depending on their DNB correlations, codes, and their data bases.

1.3 Critical power ratio (CPR), for BWRs

Please specify the limit for CPR, if this is a safety criterion in your country. Please specify to which event category the limit is applicable. Please specify how the CPR safety limit is established, and the underlying basis (correlations).

Belgium: Belgium has no BWRs

Canada: Canada has no BWRs.

Czech Republic: No

Finland: -

France : France has no BWRs.

Germany: As already explained for PWRs, the 95/95 – limit value of the CPR – correlation is determined experimentally. The experimental data are based on full scale fuel assembly tests.

The statistical thermal hydraulic analysis is used to determine the limit value for normal operation. It has to be demonstrated that 99.9% of all fuel rods of the core will not experience CHF during abnormal operational transients of moderate frequency (MASL 99.9). Such transients are e.g. maximum pump speed increase up to 100% or loss of main heat sink. The analysis methodology leads to a specific margin between the CPR limit and the minimal allowable CPR during normal operation which is monitored by the power density limitation system.

Hungary: –

Japan:

(1) Limit of minimum CPR

Allowable limit of minimum CPR is defined as the value at which boiling transition should not occur in the more than 99.9% fuel rods in a reactor. Typical allowable limit of minimum CPR is between 1.05 and 1.07. The uncertainties of CPR correlation, operating conditions, fuel fabrication specification, core monitoring system and etc. are considered to determine the allowable limit;

(2) Event category

Minimum CPR should be more than allowable limit for normal operation and abnormal transient;

(3) CPR correlations

CPR correlations (e.g. GEXL correlation) were developed based on the large database of thermal hydraulic tests.

General formula of CPR correlations is as follows:

$Xc = f(L_B, D_Q, G, L, P, R)$

Where, Xc is average critical quality at cross section
L_B is boiling length
D_Q is thermal equivalent diameter
G is coolant mass flow
L is heat length
P is coolant pressure
R is coefficient with respect to local peaking pattern

Korea: There is no BWRs in Korea

Netherlands: The only BWR in the Netherlands is a BWR which is in a so-called Post-operational state awaiting decommission

Spain: Similar to the PWR case, fuel rod damage due to overheating of the clad will be avoided in case of Anticipated Operational Occurrences (AOO's). In Unexpected Operational Occurrences (UOO's) and Design Basis Accidents (DBA's) those fuel rods predicted to reach MCPR safety limits will be assumed to fail and should be included in the radiological dose calculations.
CPR correlations are fuel-design dependent, for example:

- GEXL07 for GE-11 fuel design;
- GEXL10 for GE-12 fuel design;
- ABBD 2.00 for SVEA – 96+ fuel design.

MCPR safety limit (SLMCPR) is established to ensure that at least 99.9% of the fuel rods in the core will not experience boiling transition, and it is established taken in to consideration all analytical and surveillance uncertainties. MCPR operating limit (OLMCPR) is established to ensure that SLMCPR will not be reached during the assumed worst AOO.

Sometime ago, SLMCPR limit value used to be a fixed fuel design-dependent number (i.e. 1.08). Nowadays, SLMCPR is specifically calculated for each reload core and all type of fuel designs loaded in the core. There are several approaches to do this. In addition, a set of pre-selected AOO's (the most MCPR challenging events for a particular type of plant) are re-evaluated for each reload core in order to determine the cycle and fuel design-dependent OLMCPR. Finally, when needed, the operating cycle can be broken in several sub-intervals in terms of cycle-exposure with different OLMCPR values.

Sweden: For CPR correlation's, the limiting (minimum) value of CPR is to be established such that at least 99.9% of the fuel rods in the core could not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

Typical allowable limit of minimum CPR is between 1.05 and 1.07. The limit is determined of the uncertainties of CPR correlation, operating conditions, fuel fabrication specification and core monitoring system. Fuel design specific tests are performed to establish the correlation between critical power and parameters of influence.

Fuel and cycle dependent MCPR are used, with values from 1,28 to 1,70. Those are applied to normal operation and anticipated operational occurrences.
In Sweden a stationary dryout limit is applied, which is set to 1+4σ. That will assure that stationary dryout is avoided by at least 99,99 percent. Where σ is the uncertainty (see above).

Switzerland: Fuel / cycle dependent safety limit (values from approx. 1.15 to 1.30), applies to Anticipated Operational Occurrences (AOO, cat. 2 events).Basis: dryout (critical power, i.e. point on onset of transition boiling) to be avoided (95% confidence) for more than 99.9% of fuel rods. Fuel design specific tests are performed to establish the correlation between critical power and parameters of influence.

UK: –

USA: Different vendors use different numerical values depending on their CPR correlations, codes, and their data bases.

1.4 Thermal mechanical limitations

Please state the basis for the current thermal mechanical limitation(s) in your country, if applicable. Please specify what the basis for these limitations is (underlying criteria such as strain level, oxidation, internal gas pressure etc.) and to what extent these limits are exposure dependent.

Belgium: There is a general criterion requiring no fuel system damage (including any fuel assembly component) during normal operation and condition II events. To meet this criterion, each fuel vendor has to specify acceptable fuel design limits to cope with all known fuel system damage mechanisms. A corresponding acceptable value/criterion must be defined for the following:

- Stress;
- Strain;
- Fatigue;
- Dimensional changes (growth and bowing);
- Internal pressure;
- Fretting wear;
- Oxidation/Hydriding;
- Fuel pellet overheating;
- Fuel assembly hold-down force;
- Control rod integrity.
- ....

Canada: A licence limit is put on maximum channel power and on maximum bundle power. The limit is currently used for all events except slow overpower, which uses a probabilistic approach.

CANDU operators have limits on maximum wear and minimum bundle to channel clearance. Other CANDU fuel mechanical limits are being developed.

Czech Republic: Requirement: (Regulation 195/1997Sb.) §14 (1) The fuel system shall withstands the design irradiation in the reactor core, without its damage under conditions of normal and abnormal operation despite all considered processes of deterioration of material properties and of environs conditions that can occur during the operation.

This is a general criterion requiring no fuel system damage under conditions of normal and abnormal operation (Condition I and II) Fuel vendors have to identify a set of specified acceptable fuel design criteria to meet this requirement. All known fuel system damage mechanisms has to be covered.
See Attachment.

Finland: The probability of fuel damage caused by the mechanical interaction between fuel and cladding shall be extremely low.

France : Fuel assembly and fuel rod design bases shall ensure that :

- there is no unacceptable mechanical deformation;
- the fuel rod integrity is maintained
  during normal operation and incidents of moderate frequency (condition I & II events).

To meet these general requirements, the following criteria have to be fulfilled for condition I & II events:

- the maximum fuel temperature shall be less than the melting temperature fuel (usually checked through the limitation of the linear power density - 590 W/cm);
- the internal pressure of the lead rod shall be limited to a value below that which could cause the diametral gap to increase due to outward cladding creep;
- the cladding stress must remain less than the value at which loss of rod integrity is predicted. Particularly, cladding stress shall be less than the PCI failure threshold as determined by power ramp tests;
- the cladding fatigue damage shall be less than 0.8;
- the total tensile creep strain is limited to 1% from the unirradiated condition for steady-state operation;
- the total strain during a transient is limited to 1%;
- the fretting wear shall be less than 10% of the cladding thickness;
- the fuel rod growth shall preclude rod to nozzle interference contact;
- the circumferential average oxide thickness shall not be greater than 100µm (Cf. also 2.6) and the hydrogen pickup in the cladding shall not exceed 600 ppm;
- the metal oxide interface temperature shall not be greater than 400°C for conditions I event and 425°C for conditions II event.

  The above limits shall take in account irradiation, tolerances and uncertainties.

Germany: Thermal mechanical limits are defined for gas pressures, oxidation, hydrogen content, plastic strain and others. The criterion for maximum fission gas results from the provision that lift-off of the cladding during lifetime of the fuel rod does not occur. The criterion for acceptable oxidation results from a probabilistic analysis which is based on a corrosion data bank developed by SIEMENS. The provision is that less than 1 rod from all rods would fail during lifetime if the predicted corrosion levels of all rods are compared with the 100% failure threshold derived from this data bank. Concerning hydrogen content a value of 2000 ppm is accepted. This value is derived from an embrittelment investigation showing that the neutron embrittelment is the leading embrittelment process. The cladding strain for a follow up operation is limited up to 2.5% of the equivalent plastic strain of the cladding.

Hungary: Both for Russian and BNFL fuels several thermal mechanical limits are given. These cover the requirements of NSS Guide No. 3.6 (*Requirements for the Fuel Design of NPPs*), which have regard for collapse of cladding, fission gas release, PCI, deformations, stresses, creeping, fatigue of materials, density and volume changes of fuel pellets, spring inside the fuel element and fuel assembly handling. Details are confidential.

Japan:

(1) General criteria :
Fuel assembly should be designed to maintain its integrity considering the various practical effects through the life time in reactor.
Fuel assembly should be design not to be deformed remarkably during transportation and handing;

(2) Criteria for abnormal transient is that fuel cladding does not fail mechanically, and fuel enthalpy should be less than allowable limit.
The phrase, 'Fuel cladding does not fail mechanically', means as follows:
Circumferential plastic strain of cladding is less than 1%;

(3) The above limits are independent of exposure.

Korea: To preclude systematic fuel rod damage(including rod failure) during normal operation and anticipated operational occurrences, thermal and mechanical limitations for all known damage mechanism(stress, strain, fatigue, oxidation, fretting, rod internal pressure, internal hydriding, cladding collapse, overheating of cladding and pellet, PCI, mechanical fracturing, etc.) should be given and obeyed up to approved burnup limit.

Netherlands: There is a general criterion requiring no fuel system damage (including any fuel system damage) during normal operation and anticipated operational occurrences. To meet this criterion, the fuel vendors have to identify specified acceptable fuel design limits to cope with all known fuel system damage mechanisms. (stress, strain, fatigue, oxidation, hydriding, etc.). To mention are:

- Total clad strain < 1%;
- No centre-line fuel melting;
- Maximum oxidation level (< 100μ);
- Hydride concentration: < 500 ppm.

During the evaluation of the generic fuel vendor report for a particular fuel design, the criteria/limits proposed are compared with widely used industry standards (i.e., those included in ASME III) and well established references.

Spain: There is a general criterion requiring no fuel system damage (including any fuel assembly component) during normal operation and AOO's (condition I & II events). To meet this criterion, fuel vendors have to identify specified acceptable fuel design limits to cope with all known fuel system damage mechanisms. A corresponding acceptable value/criterion must be defined for the following:

- Stress;
- Strain;
- Fatigue;
- Dimensional changes (growth and bowing);
- Internal pressure;
- Fretting wear;
- Oxidation/Hydriding;
- Fuel pellet overheating;
- Fuel assembly hold-down force;
- Control rod integrity.

(Note: Allowable oxidation/hydriding and crud levels as well as fretting wear must be taken into consideration when assessing the thermal-mechanical criteria).

During the evaluation of the generic fuel vendor report for a particular fuel design, the criteria/limits proposed are compare with widely used industry standard (i.e. those included in the section III of the ASME code) and well established references (i.e. no fuel centerline melting for the fuel pellet overheating criterion), and eventually are accepted.

Frequently, the fuel vendors established a burnup-dependent limit curve establishing the allowed local power in the hot rod for a particular fuel design during steady-state operation. This curve is the result of the so-called fuel rod thermal-mechanical performance limits and it is included in the plant technical specifications (or in the Core Operating Limits Report) for each fuel design. Demonstration of individual bundle and core designs to these performance limits ensures compliance with the fuel rod thermal-mechanical licensing limits, provided that the assumed transient-dependent overpowers bound the core specific overpower (this must be verified during the reload safety evaluation reports). For off-rated operating conditions, correction factors can be needed to ensure that the event initiated from off-rated conditions shall be no more severe than if the event was initiated from the rated conditions.

Sweden: There shall be thermal and mechanical limits to ensure that the fuel rod is not damaged during normal operation and anticipated operational occurrences. An acceptance valu/criteria must be defined for stress, strain, fatigue, fretting, collapse, dimensional changes and internal pressure. Oxid, hydrogen, crud and irradiation induced effects must be included when assessing the acceptance valu/criteria.

A thermal-mechanical operating limit curve for each fuel design is used for the Swedish BWR.

Switzerland: (a) Total clad strain < 1%; no centerline fuel melting (PCMI)(b) Max. oxidation level: fuel design specific values, from approx. 70 to 100 microns at EOL(c) Hydride conc: 500 ppm, however only defined for some fuel designs(d) Internal gas pressure: 'no lift-off' criterion for most fuel designs
Therm.-mech. limits normally established as a function of exposure.

UK: The rods are limited thermo-mechanically in terms of clad stress, hoop mean clad strain, localised clad strain, clad collapse, clad fatigue, clad oxidation and hydriding, rod internal pressure and fuel temperature (melt).

USA: (See separate criteria under 1.8 Others)

1.5 Discharge burnup

Please state the value(s) of the current maximum discharge burnup in your country, if applicable. Please specify what the value(s) refer to (i.e. average or peak rod, average or peak assembly, average batch etc.)

Belgium : UO2 fuel is presently licensed for a maximum fuel assembly average burnup of 55 GWd/t and MOX fuel for 50 GWd/t.

Canada: CANDU operators have limits on maximum bundle-average discharge burnup such as 20 MW.d/kg U. There is no licence limit for natural uranium CANDU fuel, which has a bundle-averaged discharge burnup between 6 and 9 MW.d/kg U depending on the reactor and the region of the core. (Note that CANDU on-power fuelling is done by pushing only a few fresh bundles into the channel in one direction so the burnup peaks at the low-power discharge end of the channel).

Czech Republic: Requirement: (Regulation 195/1999Sb.) §14 (1) The fuel system shall withstands the design irradiation in the reactor core, without its damage under conditions of normal and abnormal operation despite all considered processes of deterioration of material properties and of environs conditions that can occur during the operation.
This is a general requirement, no limitation is specified. But for individual fuel designs the maximum burnup has to be be specified.

Finland: –

France : UO2/Zy-4 fuel is presently licensed for a maximum fuel assembly average burnup of 52 GWd/t.
There is no formal limit for MOX fuel but the license is applied to 3 annual cycles and an average Pu content of 7.08% per assembly, i.e. a maximum fuel assembly average burnup of 42 GWd/t.

Germany:	The reactor safety commission (RSK) accepts a stepwise increase of the discharge burnup provided the evaluation of both the operational experience and the recent research results in the high burnup regime would confirm the acceptability of an increase. The presently approved fuel assemblies with 4% resp. 4.4% enrichment will achieve discharge burnups of 52 GWd/t resp. 57 GWd/t.

Hungary: Discharge burnup will be limited for both type of fuels. Determining of the actual values of the limits is under way.
The design limits of discharge burnup are:

BNFL	60 GWd/tU for average burnup of fuel rod
Russian profiled	49 GWd/tU for average burnup of assembly 55 GWd/tU for average burnup of fuel rod 64 GWd/tU for average burnup of fuel pellet

Japan: Maximum fuel assembly average burnups:

BWR : 55GWd/t (UO_2), 40 GWd/t (MOX)
PWR : 48GWd/t (UO_2), 45 GWd/t (MOX)

Korea: The approved maximum rod average burnup in Korea is different with fuel type.

- Westinghouse type fuel (14X14, 16X16, 17X17): 60 GWD/kgU;
- CE type fuel (16X16): 58 GWD/kgU (Tentative).

Netherlands: 60 GWd/tU for fuel rod (License limit)
55 GWd/tU fuel for element assembly as a design limit (KWU)

Spain: There is no a general safety criterion applicable to all Spanish reactors associated to fuel discharge burnup. However, for all fuel designs there must be specified the maximum burnup reachable during reactor operation.
The origin of the applicable burnup limit can be different depending on the safety analysis methodology of the fuel supplier. There are to basic approaches currently in use in Spain depending on the fuel supplier:

- An explicit maximum burnup value, associated to the analytical tools used for fuel performance assessment qualification range (in terms of exposure), can be established; or
- An indirect maximum burnup value associated to the maximum burnup assumed in the fuel rod thermal-mechanical performance limits assessment (also consistent with the range of qualification of the analytical tools).

It is worthwhile to notice that burnup limits are strongly influenced by the assumed operating conditions in the thermal-mechanical performance limits assessment. Therefore, the established burnup limits could no longer be applicable if the actual reactor operating conditions significantly differ from those assumed in the safety evaluation. The same is also true for the prediction capability of the analytical tools, as most of the fuel performance models are semi-empirical and have been developed from a fixed fuel data-base that should be representative of the actual reactor operating conditions. As a consequence, current regulatory practice in Spain is to focus more on the fuel operating history than on the burnup limit values when reviewing plant-specific reload safety evaluation reports.

Sweden: No limit is specified. However, for each fuel design the maximum burnup has to be specified and justified.

Switzerland: Fuel design dependent limits, up to 60 MWd/kgU (PWR) and 50 MWd/kgU (BWR) assembly max. (or peak) exposure.

UK: Limits are determined by fuel design, validation and fuel handling issues. Currently the PWR pellet limit is 55 GWd/te.

USA: 62 GWd/t average for the peak rod. Some fuel designs have slightly lower limits because the vendors did not request an increase for all fuel types.

1.6 Shutdown margin (SDM)

Please state the value of the current SDM limit(s) in your country, if applicable. Please specify the basis for this limit (methods / manufacturing uncertainties or other).

Belgium: The shutdown margin criterion requires that the core shall be subcritical under both hot and cold conditions even if the most effective control is stuck out of the core. In these conditions and taking into account the uncertainties, shutdown margins, are between 1.77 and 2.9%, according to the type of reactor.

Canada: CANDU, being a natural uranium reactor with on-power refuelling, normally has reactivity changes that are dominated by the effect of power changes. The shutdown system in CANDU is designed to be independent of the process systems and the other safety systems. Shutdown depth is designed for moderator poison dilution from a fuel channel rupture during start up without credit for the control absorbers, the other shutdown system, nor the emergency core cooling system. Margins for stuck shut off rods and analytical uncertainties are included.

Czech Republic: Requirement: (Regulation 195/1999Sb.) § 21:

(1) The reactor shall be provided with the systems which are capable to shut down it under normal and abnormal operations and under the accident conditions. They shall ensure that the shut down can be maintained even at the most reactive core situation. The effectiveness, speed of action and shutdown margin shall assure that the specified design limits are not exceeded;

(2) … been capable to perform their functions even in the case of a single failure;

(5) …. the special attention shall be devoted to the failures originating anywhere at nuclear installation that might put out of operation a part of these components;
The subcriticality margin is a core design dependent value. Minimum is specified in the plant technical specifications.
Shutdown margins are determined by results of analysis of control requirements, which produce the most limiting conditions.
In practise, the highest-worth control rod is assumed to be stuck and analytical model uncertainties are accounted for by subtraction of 10%.

Finland: –

France: The shutdown margin criterion requires that the core shall be subcritical under both hot and cold conditions even if the most effective control is stuck out of the core. In these conditions and taking into account the uncertainties, shutdown margins, are between 1,6 and 2%, according to the type of reactor.

Germany: The minimum shutdown margin for normal operation is defined in the KTA rule 3101.2. In case of an analytical demonstration with an accepted design code, the net shut down reactivity shall not be lower than 1% until the poisoning system takes upon the long term subcriticality. If the shutdown reactivity is explicitly measured a minimal value of 0.3% net shutdown reactivity is allowed. For hot zero power and power condition automatic control means keep the control rod groups within the insertion limits.

Hungary:
2% at the beginning of cycles
0 at the end of boron cycles

Japan: Reactor should be able to be shutdown under both hot and cold conditions, even if the most effective control rod was withdrawn and could not be inserted.

Korea: Reactivity control system should have capability of controlling reactivity changes under normal operation including anticipated operational occurrences and accident conditions, with appropriate margin for stuck rods.

Netherlands: (NVR 2.1.14 = amended version of IAEA SS50-D14)
“One of the shutdown systems shall be capable of quickly rendering the nuclear reactor subcritical by an adequate margin from operating and accident conditions. An adequate margin is considered to be achieved when the k-effective is calculated to be less than 0.99 in the hot shutdown condition and the highest worth control rod not inserted, which value includes tolerances and uncertainties of calculation”.

Spain: All Spanish reactor must verify for each reload that the core will be subcritical at its most reactive condition throughout reactor life by an amount equal to the minimum shutdown margin specified in the pant technical specifications. In all analysis, the single highest-worth control rod will be assumed to be in its full-out position and analytical model uncertainties will be accounted for.

The subcriticality margin is a reactor-dependent value specified in the plant technical specifications. There are several shutdown margin values in the plant technical specifications depending on the allowable reactor operating conditions (e.g. hot power, hot standby, hot shutdown, and cold shutdown). Typical values for shutdown margin at power are between 0.3 – 0.5 %ΔK/K.

Sweden: The shutdown margin shall be at least 0,5 %. It consist of a physical margin is 0,25 % and a measurement uncertainty of 0,25 %. When calculating the shutdown margin it shall be at least 1,0 %, where 0,5 % is the uncertainty in the calculation. When the shutdown margin is calculated the most reactive control rod is supposed to be stuck out of the core. Operating values is normally 1,4 % for BWR.

The Swedish BWR reactors are required to have an independent system from the control rod system to shut down the reactor. The shutdown margin should be at least 2,5%, including margins for uncertainties, and verified by calculation each new core design.

Switzerland: Limits vary from 0.3 to 1% deltaK/K (subcriticality margin, with strongest rod withdrawn).

UK: Shutdown margin is assessed at levels dependent on the assumed core state, eg 1.3% for a PWR immediately post--trip with its most reactive rod stuck out. Analysis is performed taking due account of calculational uncertainty.

USA: No. Shutdown margins in PWRs and BWRs are determined by specific accidents, which produce the most limiting conditions.

1.7 PCI

Please state the basis for any limitations on PCI in your country, if applicable.

Belgium: The general design bases require that fuel integrity shall be maintained during normal operation and incidents of moderate frequency (condition I & II events). So, it shall be demonstrated that PCI will not lead to fuel rod failure even under condition II event.

PCI limits shall be set for each fuel design, based on experimental power ramp tests at different burnup levels. If needed, operating limits shall be specified to include the effect of:

- extended operation at low power,
- rates of the power changes.

Canada: Various PCI correlations are being used to prevent fuel defects during on-power fuelling and during control rod withdrawals. They are based on fuel performance in CANDU reactors and in research reactor loops.

Czech Republic: No current criterion is applied

1. Related criteria;

a) uniform strain of cladding less then 1%
b) *no fuel melting*
2. Operational experience for the same or similar design is considered as supporting evidence;
3. Analysis of crack (depending of size) propagation, rate of power changes, sugested preconditioning etc is encouraged.

Finland: The damage caused by the mechanical interaction between fuel pellet and cladding shall be prevented. Because of that, operating limits concerning changes in power and the rate of the changes shall be determined for the fuel types used. The stress corrosion of the cladding, among other things, shall be taken into account in determining the limits.

France: The general design bases require that fuel integrity shall be maintained during normal operation and incidents of moderate frequency (condition I & II events). So, it shall be demonstrated that PCI will not lead to fuel rod failure even under condition II event.

PCI limits shall be set for each fuel design, based on experimental power ramp tests at different burnup levels. If needed, operating limits shall be specified to include the effect of:
- load following;
- extended operation at low power;
- rates of the power changes.

Germany: According to KTA rule 3101.2 the power density change is to be limited such that the local failure thresholds in terms of power density are not exceeded. These thresholds are identified in specific power ramp tests. They depend on both local thermal hydraulic condition and burnup.

Hungary: See 1.4

Japan: No

Korea: There is no definte criterion to preclude PCI failure, but the following 2 criteria are applied as a minimum requirement.
- uniform strain of cladding; < 1%
- no fuel melting.

Netherlands: A related criterion is:
uniform strain of cladding < 1%;
no fuel melting;
fuel conditioning rules in Technical Specifications.

Spain: No specific criterion for PCI is currently applied. However, it is assumed that several fuel thermal-mechanical related criteria (i.e. stress/strain and fuel pellet-overheating limits) together with specific fuel design features (cladding-barrier) and fuel preconditioning practices (PCIOMR's) will help to avoid this type of fuel failures.

Sweden: The Swedish requirement is that each fuel design shall have a PCI limit. The PCI limit shall be based on experimental investigations.

Switzerland: PCI limits / limitations, which are fuel vendor specific, not formally part of the licensing basis. However, compliance is strongly promoted by the regulator, and thus nuclear operators generally obey these limits.

UK: In addition to the thermo-mechanical constraints noted above the core is designed to comply with the PCI local power limit which must be fulfilled in normal operation and frequent faults.

USA: No. Some people think that the 1% strain limit is a PCI limit, but the 1% strain limit preceeds consideration of PCI by NRC. NRC tried, but was never successful at developing a PCI criterion.

1.8 Others

Please add new items to the list if they are missing in the above.

Belgium: In condition II events, the LHGR is limited to 590 – 656 W/cm depending on the power plant.

Canada: There are other requirements imposed by the fuel on its interfacing systems: coolant, fuel channel, fuel handling, fuel management, and reactor instrumentation.

Czech Republic: See Attachment

Finland: The damage of fuel and control rods has to be prevented during operation, handling and transport. To ensure this, at least the following phenomena and facts shall be taken into account:

- stresses and strains of the various parts of fuel and control rods;
- fatigue damages caused by cycling loads during operation;
- oxidation of various parts and hydriding of the cladding;
  chemical and physical properties of the coolant;
- densification and swelling of fuel pellets;
- spring force of the spring inside the fuel rod to prevent fuel pellets from moving during the transport and handling of fresh fuel;
- stresses caused by handling and transport, which can affect the behaviour of fuel and control rods during operation.

France: –

Germany: –

Hungary: At present outlet temperature of assemblies, inlet temperature of the core and linear power density of fuel elements (not burnup dependent) and different kind of power peaking inside the core are limited. These limitations are as follows:

- Linear power density: 32.5 KW/m;
- Outlet temperature of assemblies: 312 ºC;

- Core inlet temperature: 267 ºC;
- Maximum assembly power peaking: 1.35;
- Maximum fuel element power peaking: 1.55;
- Maximum assembly node power peaking: 1.9;.
- Maximum fuel element node power peaking: 2.14

Limits on power peaking change by the reactor power. The above values are valid at nominal power.

A licensing process is under way, which will change the set of above limited parameters to the set of sub-channel outlet temperature of assemblies, core inlet temperature and burnup dependent linear power density of fuel elements. These new limitations will be valid for the core containing Russian profiled assemblies. The aimed values are as follows:

- Linear power density: 32.5 KW/m for fresh fuel and decreasing by burnup with different slope along subsequent section;
- Sub-channel outlet temperature: 325º C;
- Core inlet temperature: 267 ºC.

Several additional limitations affecting the fuel conditions in different operational states are in force, which cover limitations on reactivity worthness of all absorbers without the most effective, that of the control group and the most effective absorber, differential worthness of the absorber groups, reactivity coefficients of moderator temperature, boron acid concentration and fuel temperature, critical values of boron acid concentration, coolant flow rate of assemblies etc.

Japan: No

Korea: –

Netherlands: –

Spain: There are two additional criteria:

- The fuel assembly design must ensure that the occurrence of thermal-hydraulic instabilities in the core will be unlikely. This criterion is especially relevant for BWR;
- The fuel assembly design must ensure control rods can be inserted in the core when required. For the evaluation of this criterion, the effect of combined Safe Shutdown Earthquake and Loss-of Coolant Accident loads need to be evaluated to assure that component deformation is not severe enough to prevent control rod insertion.

Sweden: –

Switzerland: Reactivity Initiated Accident (RIA): no formal fragmentation limit, however fuel failure limit (which should also warrant against fragmentation) is currently defined from about 130 cal/g (radially averaged fuel enthalpy increase) at 0 exposure to 0 cal/g at 80 MWd/kgU exposure (the so – called 'Swiss Curve')
LOCA criteria: basically identical to those defined in 10CFR50 par 50.46.

UK: –

USA:

Design Stress	ASME Section III
Design Strain	ASME Section III
Strain Fatigue	<2 on Stress; <20 on Cycles
Fretting Wear	Should be limited (Include in Stress/Strain/Fatigue)
Oxidation	Should be limited (Include in Stress/Strain/Fatigue)
Hydriding	Should be limited (Include in Stress/Strain/Fatigue)
Crud	Should be limited
Rod Bow	Include in Design Analysis
Irradiation Growth	Include in Design Analysis
Internal Gas Pressure	< System Pressure or Justified
Hydraulic Lift Loads	<Hold down Force
Cladding Collapse	No collapse
Overheating of Fuel Pellets	No Centerline melting

2. Accident Conditions (LOCA, RIA)

2.1 LOCA, cladding corrosion

Please state the maximum cladding oxidation specified in the LOCA related safety criteria in your country. Clarify whether this is the total oxidation (i.e., the one incurred during base irradiation + the one occurring during the LOCA transient), or the oxidation during the LOCA transient only.

Belgium: Maximum cladding oxidation shall not exceed 17% of the actual cladding tube wall thickness at any point. Currently, this criterion includes corrosion before and during the LOCA transient.

Canada: No

Czech Republic: Cladding material properties dependent
The total local oxidation of the cladding does not exceed 17% (Zircaloy-4) or 18% (alloy Zr1%Nb) of the initial thickness before oxidation.

Finland: An excessive embrittlement of the cladding shall be prevented. To ensure this, it shall be shown that:

- the cladding is not oxidised during an accident to the degree that it cannot withstand the loads caused by the accident, for example stresses due to thermoshock during quenching at the late phase of a loss of coolant accident. In estimating the total thickness of the required ductile part of the cladding, attention has to be paid to the external and possible internal oxidation of the cladding during the accident and to the preceding oxidation during normal operation. Additionally, the chemical interactions between uranium dioxide and cladding material in connection with cladding collapse have to be taken into consideration. Also the loads caused by the handling, removal transport and storage of the fuel rod bundle after the accident have to be included in this assessment;

- the oxygen adsorbed during normal operation and during an accident does not excessively embrittle the cladding. The effect of the adsorbed oxygen on the cladding shall be experimentally determined;
- the temperature rise of the cladding has been limited to the level, where the oxidation of the cladding as a consequence of metal-water reaction is still controllable. Because of this, the highest temperature of the cladding may not exceed 1200 ºC in case of an accident.

France: Maximum cladding oxidation shall not exceed 17% of the actual cladding tube wall thickness at any point. Currently, this criterion includes corrosion before and during the LOCA transient.

Germany: According to RSK guidelines the maximum oxidation depth of the cladding calculated will not exceed 17% of the actual cladding tube wall thickness at any point. This provision ignores the corrosion layer which might have build up during normal operation because it is assumed that this layer does not contribute to both the strength of the cladding and the protection against oxidation.

Hungary: Cladding oxidation shall not exceed 17% including LOCA and base irradiation periods.

Japan: The criteria requires following:

- The calculated stoichiometric oxidation shall not exceed 15% of the cladding thickness before significant oxidation;
- 15% is the oxidation during the LOCA transient only.

Korea: Total oxidation (normal oxidation + transient oxidation) should be less than 17% of cladding thickness before oxidation.

Netherlands: 17% of original wall thicknes during LOCA (as per 10CFR50 par 50.46)

Spain: In order to provide adequate core cooling during a LOCA, a local clad oxidation of less 17 percent of the total clad thickness prior oxidation must be assured. The rate of cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just correlation.

While it is acknowledge that could be overly conservative, currently, the 17% criterion must be interpreted as total oxidation, including pre-transient and transient oxidation. Investigation is under way to relax this interpretation or even to change the criterion.

Sweden: The total cladding oxidation shall not exceed 17 %.

Switzerland: 17% of original wall thickness, during LOCA (as per 10CFR50 par 50.46)

UK: There are two limits that have to be complied with for LBLOCA. These are:

- 17% local clad oxidation, taken as ratio of the original clad thickness before oxidation. This pessimistically includes the pre-fault oxidation, although it is believed that any pre-fault oxidation would inhibit oxidation;

- 1% core-wide clad oxidation – this is to limit the amount of Hydrogen build-up.

USA: 17%. This is corrosion before the accident plus oxidation during the transient.

2.2 LOCA, hydrogen content in cladding

Please state the maximum hydrogen content in the cladding, if this is specified in the LOCA related safety criteria in your country. If not, state "No". If you don't know, do not write anything here. Clarify whether this is total H content (base irradiation + LOCA transient) or the H picked-up during the LOCA transient only.

Belgium: No

Canada: No

Czech Republic: No

Finland: –

France: No

Germany: The maximum hydrogen content in the cladding due to the LOCA transient is not limited.

Hungary: No

Japan: No

Korea: No

Netherlands: No

Spain: There is no specific requirement except that the rate of hydrogen generation from the metal/water reaction shall be calculated using the Baker-Just correlation.

Sweden: No

Switzerland: No

UK: No

USA: No

2.3 LOCA, peak cladding temperature

Please state the value of the maximum cladding temperature specified in the safety criteria in your country.

Belgium: 1204 ^{0}C

Canada: 1760 ºC

Czech Republic: The fuel rod cladding temperature does not exceed 1200^{0}C

Finland: The highest temperature of the cladding may not exceed 1200 ºC in case of an accident.

France : The maximum cladding temperature shall not exceed 1204 ºC

Germany: The maximum fuel rod cladding temperature is not allowed to exceed 1200°C.

Hungary: Cladding temperature shall not exceed 1200 ºC

Japan: The maximum cladding temperature shall not exceed 1200 ºC.

Korea: Peak cladding temperature should be less than 1204^{o}C

Netherlands: 1204 ^{0}C

Spain: In order to provide adequate core cooling during a LOCA, a peak clad temperature of less than 2200 ºF must be assured. This limit, together with the 17% oxidation limit will assure no excessive clad embrittlement will happen and that the fuel rod will be able to withstand the expected loads during the transient.

Sweden: The peak cladding temperature shall not exceed 1204° C.

Switzerland: 1200 ºC (as per 10CFR50 par 50.46)

UK: The peak clad ternperature is limited to 1204°C. This, together with the 17% clad oxidation limit, prevents excessive embrittlement of the cladding.

USA: 2200 ºF

2.4 LOCA, rod pressure

Please state the value of the maximum rod inner pressure if this is specified in the safety criteria in your country.

Belgium: No

Canada: No

Czech Republic: No

Finland: –

France: No

Germany: The internal pressure of the fuel rod during LOCA transient is not limited. Instead of specifying such a limit the RSK Guideline requires the demonstration that not more than 10% of all fuel rods would fail during LOCA transient. This limitation makes it necessary to assess the effect of rod internal pressure on the number of failed rods.

Hungary: No

Japan: No

Korea: No

Netherlands: No, there is not a specific internal rod pressure criterion. However, steady-state internal pressure criterion will limit the prior maximum internal rod pressure. In addition, clad swelling and burst models used for LOCA analysis must be accurate enough to predict rupture of the rod clad and not to underestimate the assembly flow blockage, including the effect of differential pressure, local temperature and heat up rate.

Spain: No, there is no a specific internal rod pressure criterion. However, steady-state internal pressure criterion will limit the prior maximum internal rod pressure. In addition, clad swelling and burst models used for LOCA analysis must be accurate enough to predict rupture of the rod clad and not to underestimate the assembly flow blockage, including the effect of differential pressure, local temperature and heat up rate.

Sweden: No, but often the rod failure criteria is based on a temperature-rod pressure curve.

Switzerland: No

UK: The rod intemal pressure during normal operation is limited to the coolant pressure (155 bar). This prevents outward creep of the cladding during normal operation and is also an assumption used in the LBLOCA analysis. Clad ballooning during the LBLOCA is then explicitly modelled, using 155 bar for rod internal pressure.

USA: This criterion is usually listed as being applicable to normal operation, but it is intended primarily for LOCA. Rod pressure is limited such that the outward cladding creep rate is not less than the pellet swelling rate; therefore, the gap does not open.

2.5 LOCA, hydrogen generation

Please state the maximum value for hydrogen generation, if this is a safety criterion in your country.

Belgium:	Hydrogen generation from the reaction of cladding and steam shall not exceed 1% of the amount which would be produced by the reaction of the whole active cladding
Canada:	No. The limit is on hydrogen concentrations in containment.
Czech Republic:	The total amount of hydrogen generated from the chemical reaction of cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the cladding in the active core were to react.
Finland:	The amount of hydrogen caused by the chemical interaction between coolant and cladding shall not exceed 1% of the amount, which could be developed, if the whole active cladding surrounding fuel pellets would react with the coolant.
France:	The total calculated amount of hydrogen produced by the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount which would be produced if all the cladding material enclosing the fuel (excluding the cladding in the plenum region) reacted.
Germany:	According RSK guideline it has to be demonstrated that the hydrogen generation due to zirconium-water reaction affects maximal 1% of the entire zirconium in the cladding tubes.
Hungary:	Hydrogen generation from the reaction of cladding and steam shall not exceed 1% of the amount which would come from the total reaction of cladding.
Japan:	To maintain integrity of the containment vessel, the criteria requires that the amount of hydrogen generated by cladding-water and core component- water reactions shall be sufficiently low. Since the contribution of metal-water reactions in the whole hydrogen generation depends on the design of plants, the limit cannot be determined unequivocally. Practically, capability of ECCS is accepted if the amount of generated hydrogen is estimated to be smaller than that generated by oxidation of 1% of the whole zircaloy cladding with a relevant margin.
Korea:	The calculated total hydrogen generation should be less than 0.01 times the hypothetical amount generated by the chemical reaction of all cladding with water or steam.
Netherlands:	<1% of amount from oxidation of all cladding material (as per 10 CFR 50 par 50.46).
Spain:	The LOCA rate of hydrogen generation shall be calculated with the Baker-Just correlation, and total amount of hydrogen generated from the reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the active fuel column were to react.
Sweden:	The hydrogen generation shall not exceed 1% of the amount which would come from the total reaction of the cladding.

Switzerland: <1% of amount from oxidation of all cladding material (as per 10CFR50par50.46).

UK: The hydrogen build-up in the core is limited by assessing the core-wide clad oxidation against a limit of 1% (see point 2.1).

USA: 1% of that from complete reaction of the fuel rod cladding, excluding the cladding in the plenum region.

2.6 RIA, energy deposition

Please state the value or formula for the maximum energy deposition specified in the safety criteria in your country. Please specify fragmentation or fuel failure limits, as applicable. Please clarify if a dependency on burnup or on a burn-up-related parameter exists (or is under consideration) for these limits in your country.

Belgium: The RIA must not cause a fuel radial average enthalpy rise of more than 225 cal/g for fresh fuel and 200 cal/g for irradiated fuel at any point along a fuel rod.

Canada: No. The limit is fuel temperature below melting. For LOCA, fuel element-averaged enthalpy limits of 200 and 230 cal/g used to be used.

Czech Republic: The radially averaged fuel pellet enthalpy shall not exceed 963 J/kg (230 cal/g) at any axial location in rod.

Finland: The fragmentation and melting of the fuel rod shall be prevented. The fuel enthalpy of any fuel rod may not exceed the value of radial average enthalpy, i.e. 963 J/g UO_2 (230 cal/g).

France: Reactivity excursions must not cause a fuel radial average enthalpy rise of more than 225 cal/g for fresh fuel and 200 cal/g for irradiated fuel at any point along a fuel rod.

This criterion was found to be non-conservative for highly irradiated fuel in light of test data performed in Japan and France. Based on the analysis of the CABRI experimental database, the following set of physical parameters that define a safety domain for RIAs on high burnup fuel (UO2/Zy-4) has been proposed:

- cladding waterside corrosion limited to 100 μm (in order to prevent in-reactor cladding spallation and localized hydriding);
- transient enthalpy increase, maximum enthalpy and pulse width consistent with CABRI database (no rod failure);
- maximum cladding temperature less than 700°C.

This safety domain is not considered as a fuel rod failure criterion but as a conservative way to guarantee no fuel failure. This approach has been temporarily accepted by the French Safety Authority in the expectation of adequate safety criteria for high burnup fuel.

Germany: There is no maximum energy deposition value defined in the regulation. The German licensing practice requires that no fuel rod would fail during a rod ejection accident (PWR) and during rod drop accident (BWR), respectively.

Hungary: The maximum energy deposition during RIA is 963 J/gUO2 (230 cal/g) which is an averaged value in radial direction and it is independent from burnup and power level. For AOOs the limit is 586 J/gUO2 (140 cal/g)

Japan:

(1) Criteria for fuel failure due to PCMI
(enthalpy limits for abnormal transients and failure criteria in accidents)
Fuel enthalpy increase due to abnormal transients in RIEs(reactivity initiated events) should not exceed these limits. In cases of accidents, fuel rods which had enthalpy increase equal to or higher than these limits should be considered to fail. Even if fuel fragmentation and mechanical energy generation due to the failure would occur, the shutdown capability and integrity of the reactor vessel should not be threatened. Coolable geometry of the core should also be maintained.

Fuel burnup(BU)<25GWd/tU: fuel enthalpy increase =<110cal/g
25GWd/tU=<BU<40GWd/tU : =<85cal/g
40GWd/tU=<BU<65GWd/tU : =<50cal/g
65GWd/tU=<BU<75GWd/tU : =<40cal/g (provisional)

(2) Criteria for fuel failure due to rod burst
In addition to the PCMI criteria above, allowable design limit which takes account of the burst failure of the rod are defined. The fuel enthalpy limits are,
Rod internal-external pressure difference < 6 kg/cm2 : 170cal/g
6 kg/cm2 =< Rod internal-external pressure difference < 44.4 kg/cm2 :
Linear function to the pressure difference 137 (at 6kg/cm2)-
65cal/g (at 44.4kg/cm2)
44.4kg/cm2 < Rod internal-external pressure difference : 65cal/g

In the abnormal transients, fuel enthalpy should not exceed the limits above.

(3) Enthalpy limit for the accident
Fuel enthalpy should not exceed the limit below in any RIAs. 230cal/g minus corresponding amount of enthalpy caused by various effects such as decrease in fuel melting temperature due to burnup, Gd addition, Pu addition etc.

Korea: A radially averaged fuel enthalpy of 280 cal/g is applied as a criterion to preclude fuel fragmentation.

Netherlands: Threshold line (Borssele specific)

burn up (MWd/kgU)	enthalpy rise (cal/g)
0	170
50	60
80	60

Spain: A maximum energy deposition equivalent to a radially averaged fuel enthalpy of less than 280 cal/g must be assured for any reactivity initiated accident. However, most fuel vendors use more restrictive criteria (i.e. 240 cal/g).
The adequacy of this is criterion for high burnup fuel is under investigation, and most likely the criterion will be redefined in near future.

Sweden: The fragmentation of fuel shall be prevented and for fuel rod failure a dose calculation shall be performed. The limits is based on experimental results.

	Limit for fragmentation	Limit for rod failure
Fuel pellet burnup	Fuel pellet average enthalpy	
0 MWd/kgUO	230 cal./gram UO_2	140
33	230	140
40	100	100
50	60	60
60	30	30

Switzerland: No formal fragmentation limit, however fuel failure limit (which should also warrant against fragmentation) is currently defined from about 130 cal/g (radially averaged fuel enthalpy increase) at 0 exposure to 0 cal/g at 80 MWd/kgU exposure (the so-called 'Swiss Curve')

UK: The energy deposition (assessed as radially averaged peak fuel enthalpy, RAPFE) during a rod ejection accident is limited to 752 J/g (180 cal/g), which prevents both loss of coolable geometry during the fault (for fresh and irradiated fuel) and during post-fault fuel handling.

The RAPFE limit on clad failure during fast accidents is 586.2 J/g (140 cal/g). However, in practice this limit does not need to be employed because cladding failure is only an issue for frequent faults (more frequent than 10–3 per year), which are all slow compared to the time constant of the fuel, and cladding failure for these faults is assessed using the PCI and DNB criteria.

Fuel that is failed prior to a rod ejection fault is assumed to shatter if it exceeds a RAPFE of 251 J/g (60 cal/g). This limit is based on tests with waterlogged fuel rods. Loss of coolable geometry due to rod failure propagation is not expected under RIA conditions below the 752 J/g limit.

USA: 280 cal/g radially averaged peak fuel enthalpy limit. This is to prevent fuel fragmentation and its consequences. 170 cal/g is used for low-power and zero-power accidents in BWRs as an indicator of cladding failure for dose calculations (it is not a limit).

2.7 RIA, departure from nucleate boiling (DNB)

Please state if DNB is considered in the RIA related safety criteria in your country and if/how this is applied.

Belgium: The objective is to avoid more than 10% of fuel rods entering in DNB.

Canada: No

Czech Republic: In the Czech republic the DNB is considered in:
Condition II – Faults of moderate frequency (as a safety criterion) Condition III – Infrequent faults and
Condition IV – Limiting faults
(All fuel rods experiencing DNB are assumed to fail)

Finland: –

France: DNB is not a RIA related safety criterion. However, as the safety domain defined in 2.6 assumes no fuel rod failure, DNB is considered as a potential cause of failure during RIA and is included in the accident analysis.

Germany: The heat transfer from the fuel rods to the fluid is considered under 2.6 thus departure from nucleate boiling (DNB) is considered if this might occur.

Hungary: –

Japan: The burst failure criteria stays constant at 65 cal/g above 44.4kg/cm2 of rod internal-external pressure difference, because no DNB is assumed below this enthalpy. The enthalpy corresponding with DNB was estimated 88cal/g from a result of the NSRR experiments.

Korea: DNBR should be used as fuel failure criterion to meet the dose requirements after RIA event.

Netherlands: (Borssele specific)
For RIA accidents shall be demonstrated that the hot channel fuel rods are protected against film boiling if DNBR >1.45 (100 bar < p < 140 bar). See further answer 1.2.

Only limited DNBR violation; if DNB occurs cladding temp. < 600 ^{0}C

Spain: To assess the radiological consequences of RIA-type of accidents, in PWR the DNB criterion is used to determine the number of fuel rods expected to fail during the transient. For BWR rod drop events initiated from cold conditions, an equivalent enthalpy level of 170 cal/g is used as a fuel failure indicator.

Sweden: No

Switzerland: No

UK: Not a design criterion but the current working assumption is that fuel enters DNB during a rod ejection accident.

USA: DNB is used as an indicator of cladding failure for dose calculations (it is not a limit).

2.8 Others

Please add new items to the list if they are missing in the above.

Belgium: –

Canada: Five percent equivalent uniform strain is used as the limit for all events except large loss of coolant as a fuel failure criterion. The fuel limits for the large loss of coolant accident are no fuel melting and axial fuel expansion must be less than the fuel channel gap.

Czech Republic: See Attachment

Finland: The higher the initial event frequency of a Class 2 postulated accident, the smaller the number of damaged fuel rods shall be. The number of damaged fuel rods may not exceed 10% of the total number of fuel rods in the reactor. The consequences of the postulated accident may not endanger the coolability of the fuel, either.

France: –

Germany: –

Hungary: Coolant flow shall not be blocked by the deformation of structural elements
Absorbers shall not melt even partially. Movement of absorbers shall not be blocked by the deformation of fuel elements and structural elements of the reactor.

Japan: No

Korea: –

Netherlands: –

Spain: Clad embrittlement during locked rotor break accident: a higher than the LOCA peak clad temperature limit can be established for this type of accident (i.e. 2700 ºF). The rational is that the time interval with a temperature of the clad greater than 2200 ºF clad is very limited, and can be accommodated by the cladding without substantial damage.

Sweden: –

Switzerland: –

UK: –

USA: Structural Deformation limited during LOCA blowdown and earthquakes such that control rods can be inserted and coolability is not lost (SRP 4.2 Appendix A).

Appendix II

COMPILATION OF THE RESPONSES TO THE SEGFSM REVISED QUESTIONNAIRE ON FUEL SAFETY CRITERIA.

Responses from: Finland, Germany, Hungary, Japan, Korea, Spain, Sweden, Switzerland,

PART I

FUEL SAFETY CRITERIA:

The following information was required for each safety criterion:

Q1. If this criterion applies in your country, how is it defined and what the rational behind this definition? If this criterion is not defined, why not? Please provide a numerical example (only in case of a limit.)

Q2. Is this criterion associated with regulatory approved criteria of Cat. II ? If yes, what are they (again, please include numerical examples.)

Q3. Is this criterion associated with design criteria of Cat. III ? If yes, what are they (again, please include numerical examples.)

Q4. How have new fuel&core design elements (e.g. high burnup) affected this safety criterion so far?

Q5. What type of methods are used to evaluate and maintain (monitor) this safety criterion and its safety margin (e.g. 'conservative', best-estimate, probabilistic/statistical) ?

Q6. How have new fuel&core design elements (e.g. high burnup) affected these methods so far?

PART II

What (regulatory approved) fuel designs are in use ? What fuel designs are being considered for the future ?

A. SAFETY CRITERIA

DNB/CPR safety limits

Finland

Q1. ***Definition:*** DNBR/CPR ensures that there shall be at least a 95% probability at a 95 % confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling or boiling transition condition or the limiting value of DNBR/CPR is to be established such that only 0.1% of the fuel rods in the core would be experienced a departure from boiling or transition condition during anticipated operational conditions.
Rationale: provides basis for cladding integrity by avoiding overheating and subsequent considerable oxidation and failure.
Limit defined: For WWER-440 minimum DNBR is 1.33. For BWR minimum CPR is different for different fuel design. Typical value is around 1.06.

Q2. Yes, operating limit has been defined in TechSpec, taking into account the worst anticipated transient. Typical value for BWR is 1.30. In WWER-440 DNBR is not a limiting thermal margin and it has not been defined explicitly in TechSpec. Instead of that the outlet temperature of the subchannel in the hottest fuel bundle has limited (no bulk boiling in the hottest subchannel), which take care also DNBR margin.

Q3. No.

Q4. DNBR/CPR correlations are fuel design dependent and correlations are based on full scale experiments. Fuel burnup has not affected the criterion so far (licensed fuel bundle maximum burnup 40 MWd/kgU)

Q5. Conservative and statistical methods.

Q6. Unaffected due to the low fuel burnup limit (licensed fuel bundle maximum burnup 40 MWd/kgU)

Germany

Q1. The safety requirements of the KTA rule 3101 requires to avoid CHF occurrence or to limit fuel cladding temperatures to acceptable values. This takes into account that the CHF-ratio is an early indicator of insufficient fuel rod cooling which will not imply directly fuel rod failure. Fuel rod failure would occur if the fuel cladding temperatures reach high values.
Therefore the CHF-correlation applied for the determination of the CHF-limits should be based on experimental data for the specific fuel assembly design. The statistical evaluation of these data should define the 95/95 – (fraction/confidence) – limit value of the CHF correlation.
The necessary margin for transients is determined by the thermal design. Usually statistical thermal design methods are applied to consider technological and operational uncertainties. These methods (e.g. THAM-method for BWR) differ in the number or type of parameters which are still treated in a deterministic manner. This approach applies to the subsequent items of DNBR for PWRs and CPR for BWRs. Therefore no single value can be specified as reference value for normal operation.

Q2. Yes. For PWR/BWR operating limits are defined. E.G. BWR: MASL100 for normal operation condition (minimum margin which assures that less than 1 rod will experience the transition to dry-out)

Q3. No.

Q4. Criteria unaffected so far, value however changes with fuel & core design, also the correlation used are specific to fuel design.

Q5. For BWR the THAM method is used to determine the margin to dry-out. For PWR the hot channel analysis is used to determine the margin to DNB. Sufficient margin is provided for normal operating condition when the analysis of a Loss-Of-Flow-Event (LOFE) shows that no rod experience DNB related to a specific 95/95-limit value. The resulting minimum DNB-ratio is monitored during operation by the power density limitation system (RELEB)

Q6. Methods are unaffected by new fuel & core design elements.

Hungary

Q1. The limit ensures that the boiling crisis is avoided with 95% probability at a confidence level of 95%. This is a protection against overheating of the cladding.
For Russian fuel using the Bezrukov correlation the 95/95 rule results in a DNBRmin limit of 1.33 (required for the worst AOO condition).

Q2. There is no DNB related operational limit, however, the pin power is limited by the requirement that the hot subchannel outlet enthalpy must not exceed the saturation value (limit for VVER-440).

Q3. No.

Q4. New fuel design affects only the DNBRmin value which is specific for any special design, however, the 95/95 rule is not influenced.

Q5. Evaluation of the criterion is made in a conservative way.

Q6. Up to the present, the maximum burnup licensed in Hungary does not affect the criterion or the evaluation procedure.

Japan

CPR safety limit

Q1. ***Definition***: CPR safety limit is defined as the value at which boiling transition should not occur in the more than 99.9% fuel rods in a reactor.
Rational: provides basis for cladding integrity by avoiding overheating and subsequent rod failure
Limit defined: Typical value is 1.06

Q2. Yes: Operating limit has been defined, taking into account the worst anticipated transient. Typical value is 1.21.

Q3. No.

Q4. Unaffected.

Q5. Statistical method.

Q6. Unaffected.

DNB safety limit

Q1. ***Definition:*** DNB safety limit ensures that 95% of fuel rods will not experience DNB condition with 95% confidence level.
Rational: provides basis for cladding integrity by avoiding overheating and subsequent rod failure
Limit defined : Typical value is 1.17 □ 0.75 x DNBRnominal.

Q2. Yes: Operating limit has been defined, taking into account the worst anticipated transient.

Q3. No.

Q4. Unaffected.

Q5. Statistical method.

Q6. Unaffected.

Korea

Q1. ***Definition:*** DNB safety limit provides assurance that there be at least a 95% probability at a 95% confidence level that the hottest fuel rod in the core does not experience a DNB condition during normal operation or anticipated transients.
Rationale: protect the clad overheating and subsequent failure, and fission product release to coolant.
Numerical example: Three definitions are introduced for evaluating this criterion.
DNBR correlation limit: based on CHF correlation uncertainty and dependent on the correlation in use; typically 1.17 for WRB-1, 1.19 for CE-1, 1.30 for W-3 correlations.
DNBR design limit: based on design parameter uncertainty and dependent on the design method/NSSS system; typically 1.3 ~ 1.5 for plants using statistical method (combining design parameter uncertainty with CHF correlation uncertainty) and equal to DNBR correlation limit for plants using deterministic method (treating design parameter uncertainty conservatively).
DNBR safety analysis limit: based on the margin from DNBR design limit to compensate for possible cycle specific effects; typically 1.5~ 1.8.

Q2. Yes : but system operating conditions such as pressure, temperature, power peaking factor, etc., instead of DNBR itself, are defined as Limiting Conditions for Operation of Tech. Spec. Plant operation under conditions defined in Tech. Spec. prevents a DNBR design limit from being reached during the worst anticipated transients or condition II events.

Q3. No : design margin between DNBR design and safety analysis limits employed by vendor or utility can be utilized for design/operation flexibility.

Q4. Criteria unaffected so far. values however change with fuel/core design, correlations used- derived from CHF testings specific to fuel design.

Q5. Statistical methods or Deterministic methods.

Q6. Unaffected. (the method itself is not directly related to high burnup fuel).

Spain

Q1. There is a general criterion applicable for PWR reactors: fuel rod damage due to overheating of the clad will be avoided for C-I & II events. For C-III and C-IV fuel rods predicted to reach DNBR limits will be assumed to fail and should be included in the radiological dose calculations.

DNB-correlations are fuel design-dependent, for example:
W-3 correlation is still used for HIPAR & LOLOPAR fuel for an old one-loop W-design plant in operation in Spain.
WRB-1 correlation is used for the fuel in all other W-design plants (for fuel with Intermediate Flow-Mixers WRB-2 correlation is used instead).
KWU/CHF correlation is used for the Siemens fuel.
DNBR-correlation limits are established to ensure there will be at least a 95 percent probability at a 95 % confidence level that boiling transition will not occur on the limiting fuel rods during condition I & II events. Limits values are considered to be proprietary information, but typically range from 1.15 to 1.2, and higher in case of very old correlations. However, it is important to notice that actual DNBR limit values for safety analyses differ from the limit values of the DNB correlations. This is so because analytical and surveillance uncertainties must be accounted for in the safety analyses. There are several methods to handle these uncertainties. Depending on the methodology employed to do that (i.e. the W-Standard Thermal-Design Process or the W-Revised Thermal-Design Process) the final safety limit applicable for the evaluation of a particular transient will be different. In addition to that, it is quite common that licensing analyses include extra-margins to cover anticipate or unanticipated issues that could have an impact in terms of DNB performance. These extra-margin play an important role, because the method-dependent DNBR safety limits are not recalculated every reload, and therefore an “extra-cushion” is needed to avoid unexpected shortcomings during plant operation.
Similar to the PWR case, fuel rod damage due to overheating of the clad will be avoided in case of Anticipated Operational Occurrences (AOO’s). In Unexpected Operational Occurrences (UOO’s) and Design Basis Accidents (DBA’s) those fuel rods predicted to reach MCPR safety limits will be assumed to fail and should be included in the radiological dose calculations.
CPR correlations are fuel-design dependent, for example:

- GEXL07 for GE-11 fuel design;
- GEXL10 for GE-12 fuel design;
- ABBD 2.00 for SVEA-96+ fuel design.

MCPR safety limit (SLMCPR) is established to ensure that at least 99.9% of the fuel rods in the core will not experience boiling transition, and it is established taken in to consideration all analytical and surveillance uncertainties. MCPR operating limit (OLMCPR) is established to ensure that SLMCPR will not be reached during the assumed worst AOO.
Sometime ago, SLMCPR limit value used to be a fixed fuel design-dependent number (i.e. 1.08). Nowadays, SLMCPR is specifically calculated for each reload core and all type of fuel designs loaded in the core. There are several approaches to do this. In addition, a set of pre-selected AOO’s (the most MCPR challenging events for a particular type of plant) are re-evaluated for each reload core in order to determine the cycle and fuel design-dependent OLMCPR. Finally, when needed, the operating cycle can be broken in several sub-intervals in terms of cycle-exposure with different OLMCPR values.

Q2. Yes. For PWRs, the system operating conditions defining the DNB value such as pressure, temperature, coolant flow, power peaking factor, are defined as Limiting Conditions for Operation in the Technical Specifications of the plant, instead of the specific DNBR limit value.
In the case of BWRs, the specific value of the OLMCPR for the cycle is included in the Tech Specs, usually in the Core Operating Limit Report.

Q3. No.

Q4. No. Values have been changing with new fuel and designs, generally showing an improvement in the available margin.

Q5. Conservative and statistical methods are used.

Q6. They are unaffected up to now. The methods do not take into account high burnup effects.

Sweden

Q1. ***Definition:*** For PWR DNBR/CPR ensures that there shall be at least a 95% probability at a 95 % confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling or boiling transition condition and for BWR the limiting value of DNBR/CPR is to be established such that only 0.1% of the fuel rods in the core would be experienced a departure from boiling or transition condition during anticipated operational conditions.
Rationale: provides basis for cladding integrity by avoiding overheating and subsequent considerable oxidation and failure.
Limit defined: For PWR minimum DNBR is around 1.17. For BWR minimum CPR is different for different fuel design. Typical value is 1.05-1,07.

Q2. Yes, operating limit has been defined in Tech. Spec., taking into account the worst anticipated transient. For PWR DNBR is 1.33 for Ringhals 3 and 4 and 1,407 for Ringhals 2. Typical MCPR value for BWR is from 1.30 to 1,60, depending on power and flow. For BWR MCPR is calculated cycle specific.

Q3. No.

Q4. DNBR/CPR correlations are fuel design dependent and correlations are based on full scale experiments. Fuel burnup has not affected the criterion.

Q5. For BWR best-estimate methods are used and for PWR conservative and statistical methods.

Q6. Unaffected (but for BWR new model to account for part length rods)

Reactivity coefficients

Finland

Q1. ***Definition:*** The prompt inherent nuclear feedback characteristics shall mitigate the increase in reactivity.
Rational: This is a general safety criterion for the fuel and reactor core design. It provides stable and controllable behavior of reactor power during normal operational conditions, anticipated transients and accident conditions.
Limit defined*:* Isothermal reactivity coefficient (Doppler and moderator temperature coefficient) must be negative.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion (licensed fuel bundle maximum burnup is 40 MWd/kg U).

Q5. Best estimate methods. Reactivity coefficients are calculated for each cycle and it is measured after each reloading in the start up tests.

Q6. Unaffected due to the low fuel burnup limit (licensed fuel bundle maximum burnup 40 MWd/kgU).

Germany

Q1. Under normal operating conditions the core design must provide sufficient negative reactivity coefficients for both temperature & moderator coefficient. Exceptions from that are allowed for short term transients.

Q2. Limit values for the temperature & moderator coefficients are included in the Tech Specs.

Q3. No.

Q4. Unaffected.

Q5. Realistic

Q6. Methods are unaffected by new fuel & core design elements. However the neutronic data basis has to be kept updated with respect to the extended use of MOX and Gadolinium rods.

Hungary

Q1. According to the requirement the reactivity coefficients shall ensure the stable operation of the reactor in any operational condition.

Q2. The operational range of the reactivity coefficients is given in the Tech. Spec. and is approved by the authority. The accident analysis considers the permitted value of the reactivity coefficients worst from the point of view of the investigated transient or accident.

Q3. No

Q4. The approved reactivity coefficient range is design specific.

Q5. Evaluation is conservative.

Q6. No.

Japan

Q1. ***Definition:*** The prompt inherent nuclear feedback characteristics shall mitigate the increase in reactivity.
Rational: This is a general safety criterion for the fuel and reactor core design. It provides stable and controllable behavior of reactor power during normal operational conditions, anticipated transients and accident conditions.
Limit defined: Isothermal reactivity coefficient (Doppler and moderator temperature coefficient) must be negative.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. Best estimate methods. Reactivity coefficients are calculated for each cycle or measured after each reloading in the start up tests.

Q6. Unaffected.

Korea

Q1. ***Definition:*** safety requirement of reactivity coefficients ensures that the reactor core have prompt inherent nuclear feedback characteristics enable to compensate for a rapid increase in reactivity.
Rational and limit: negative Doppler coefficient or negative power coefficient (there are no criteria that explicitly establish acceptable ranges of coefficient values)

Q2. No.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative, statistical method.

Q6. Unaffected so far.

Spain

Q1. ***Definition:*** the reactor core should have prompt inherent nuclear feedback characteristics able to compensate for a rapid increase in reactivity.
Rationale and Limit: negative reactivity feedback provides for stable and controlled core behaviour. The Moderator Temperature Coefficient (MTC) most positive and most negative values are limited.

Q2. Limit values for the MTC are included in the Tech Specs.

Q3. No.

Q4. Unaffected.

Q5. Realistic.

Q6. Unaffected.

Sweden

Q1. ***Definition:*** The prompt inherent nuclear feedback characteristics shall mitigate the increase in reactivity.
Rational: This is a general safety criterion for the fuel and reactor core design. It provides stable and controllable behaviour of reactor power during normal operational conditions, anticipated transients and accident conditions.
Limit defined: All reactivity coefficients must be negative at operating temperature.

Q2. No.

Q3. No.

Q4. New fuel designs with 10x10 fuel and part length rods have been affected the criterion. The isothermal reactivity coefficient (Doppler and moderator temperature coefficient) becomes more positive and it affects nuclear warming of the reactor. The isothermal reactivity coefficient could be positive up to about 210 °C.

Q5. Best estimate methods. Reactivity coefficients are calculated for each cycle and ITK is measured after each reloading in the start up tests.

Q6. Yes, new methods to calculate ITK more precisely for the core design.

Shutdown margins

Finland

Q1. ***Definition:*** The reactor shall be provided with two independent reactivity controls systems with diverse operating principles. Each of them shall be separately capable of shutting down the reactor during normal operational conditions. At least one of these systems alone must be capable of maintaining the reactor in the shutdown state at any reactor temperature. Both systems are capable of accomplishing their safety function even in the event of single failure.
Rational: The reactor shall be able to shutdown with defined shutdown margin at all reactor temperature.
Limit defined: Calculational SDM-limit is 1 % deltk/k. Verification of SDM is carried out after each reloading. The TechSpec limit for this measurement is 0.5 % deltk/k. In WWER 440 the reactor can not become recritical after stemline break accident.

Q2. same as above.

Q3. No.

Q4 New fuel/core designs including fuel burnup have not been affected the criterion(licensed fuel bundle maximum burnup is 40 MWd/kg U)

Q5. Best estimate method.

Q6. Unaffected.

Germany

Q1. The minimum shutdown margin for normal operation is defined in the KTA rule 3101.

Q2. In case of an analytical demonstration with an accepted design code, the net shut down reactivity shall not be lower than 1% until the poisoning system takes upon the long term subcriticality. If the shutdown reactivity is explicitly measured a minimal value of 0.3% net shutdown reactivity is allowed. For hot zero power and power condition automatic control means keep the control rod groups within the insertion limits.

Q3. No.

Q4. Unaffected.

Q5. Conservative methods.

Q6. Unaffected.

Hungary

Q1. According to the requirement the reactivity worth of the control rods and boron concentration should be sufficient to ensure that it is impossible to reach criticality in an uncontrolled manner. The approval of the fulfilment should consider also single failure.
The value of the required subcriticality is -2% and it is given in the Tech. Spec.

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Japan

Q1. ***Definition:*** Reactor should be able to be shutdown under both hot and cold conditions, even if the most effective control rod was withdrawn and could not be inserted.
Rational: The reactor shall be able to shutdown with defined shutdown margin at all reactor temperature.
Limit defined: same as above

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. Best estimate method.

Q6. Unaffected.

Korea

Q1. ***Definition:*** safety requirements of shutdown margin ensures that the reactivity control systems have capability of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.
Rational:

Q2. Yes : different operating limits that are plant specific are stated for each reactor operation mode. The limits are directly associated with accident(or transient) analysis. (ex, 2%, 6.5% delta k).

Q3. No.

Q4. Unaffected so far.

Q5. Conservative method.

Q6. Unaffected.

Spain

Q1. All reactors must verify for each reload that the core will be subcritical at its most reactive condition throughout reactor life by an amount equal to the minimum shutdown margin specified in the plant technical specifications. In all analysis, the single highest-worth control rod shall be assumed to be in its full-out position and analytical model uncertainties should be accounted for.

Q2. The subcriticality margin is a reactor-dependent value specified in the plant technical specifications. There are several shutdown margin limit values depending on the allowable reactor operating conditions (e.g. hot power, hot standby, hot shutdown, and cold shutdown). Typical values for shutdown margin at power are between 0.3-0.5 %ΔK/K.

Q3. No.

Q4. Unaffected.

Q5. Conservative methods.

Q6. Unaffected.

Sweden

Q1. ***Definition:*** The reactor shall be provided with two independent reactivity controls systems with diverse operating principles. Each of them shall be separately capable of shutting down the reactor during normal operational conditions. Both of these systems alone must be capable of maintaining the reactor in the shutdown state at any reactor temperature. Both systems are capable of accomplishing their safety function even in the event of single failure.
Rational: The reactor shall be able to shutdown with defined shutdown margin at all reactor temperature.
Limit defined: The shutdown margin shall be at least 0,5 %. It consist of a physical margin is 0,25 % and a measurement uncertainty of 0,25 %. When calculating the shutdown margin it shall be at least 1,0 %, where 0,5 % is the uncertainty in the calculation. When the shutdown margin is calculated the most reactive control rod is supposed to be stuck out of the core. Operating value is normally 1,4 % or more for BWR.
The Swedish BWR reactors are required to have an independent system from the control rod system to shut down the reactor. This system is based on water with Bor and the shutdown margin should be at least 2,5 %, including margins for uncertainties, and verified by calculation each new core design.

Q2. Yes, same as above.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. Best estimate method.

Q6. Unaffected (New 10x10 fuel designs with part length rods have increased the shutdown margin).

Enrichment

Finland

Q1. ***Definition:*** Explicit safety criterion for maximum enrichment has not been defined.
Rational: Due to the low fuel burnup limit maximum fresh fuel enrichment is below 4% and present nuclear data base covered this enrichment range.
Limit defined: Licensed fuel maximum enrichment is 4 % .

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup has not been affected the criterion.

Q5. –

Q6. Unaffected.

Germany

Q1. No explicit limit has been set.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. –

Q6. Unaffected.

Hungary

Q1. –

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Japan

Q1. ***Definition:*** Explicit safety criterion for maximum enrichment has not been defined.
Rational: Due to the fuel fabrication facility licensing maximum fresh fuel enrichment is below 5% and present nuclear data base covered this enrichment range.
Limit defined: Licensed fuel maximum enrichment is 5 %.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. –

Q6. Unaffected.

Korea

Q1. Though a definite limit is not defined in regulatory guide, it is implicitly required that maximum enrichment of the PWR fuel be less than 5 wt% to maintain the basis for validation of the physics and safety of criticality.

Q2. Yes : maximum limit of 5 wt% is defined in the licensed documents for fuel design.

Q3. No.

Q4. Unaffected so far, though the specific value was changed below 5 wt% limit.

Q5. Conservative (measured enrichment during fuel fabrication process should be less than 5 wt%)

Q6. Unaffected so far.

Spain

Q1. No explicit limit has been set. A 5% limit is implicitly observed.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. –

Q6. Unaffected.

Sweden

Q1. ***Definition:*** Explicit safety criterion for maximum enrichment has not been defined.
Rational: Due to the limitation in fuel fabrication factory the maximum fresh fuel enrichment is below 5 % and present nuclear data base covered this enrichment range.
Limit define: Licensed fuel maximum enrichment is 4,9 % for BWR fuel and for PWR 4,0 % .

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup has not been affected the criterion.

Q5. Not applicable.

Q6. Not applicable.

Internal gas pressure

Finland

Q1. ***Definition:*** The internal pressure of the fuel rod caused by the release of fission gases and pre-pressurization must remain so low that the internal pressure of the rod will not exceed the normal pressure of the coolant.
Rational: This criterion limits outward creep of the cladding and loads of the cladding during normal operation transients and accidents, which may cause premature and extensive fuel failure.
Limit define: Rod internal pressure lower that coolant pressure during normal operational conditions.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. Conservative and statistical methods.

Q6 High burnup effects, which are related to fission gas releases, have been taken into account in fuel performance codes (as far as they are known).

Germany

Q1. Thermal mechanical limits are defined for gas pressures, oxidation, hydrogen content, plastic strain and others. The criterion for maximum internal gas pressure results from the provision that lift-off of the cladding during lifetime of the fuel rod does not occur.
Rationale: prevent the loss of dimensional stability and cladding integrity from the force exerted by internal gas pressure.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. Conservative methods.

Q6. Unaffected.

Hungary

Q1. The criterion says that the internal gas pressure must not exceed the coolant pressure at nominal conditions in the reactor coolant system. The aim is to avoid lift-off of the cladding due to creep. The nominal pressure in the VVER-440 type is 12.26 MPa.

Q2. No.

Q3. In a broader sense the decrease of the permitted linear heat rate level with increasing burnup helps the fulfilment of the criterion.

Q4. Until now there is no effect.

Q5. The methodology is conservative.

Q6. Up to now no change.

Japan

PWR

Q1. ***Definition:*** The internal pressure of the fuel rod caused by the release of fission gases and pre-pressurization must remain so low that the internal pressure of the rod will not exceed the pressure which causes to increase the pellet/cladding gap by creeping out the cladding in normal operation.
Rational: avoid thermal feedback by lift off.
Limit defined: depend upon the design.

Q2. No.

Q3. No.

Q4. Unaffected

Q5. Conservative

Q6. Unaffected

BWR

Q1. –

Korea

Q1. ***Definition:*** fuel rod internal gas pressure should remain below the nominal coolant pressure during normal operation unless other criterion justified.

Rational: prevent the loss of dimensional stability and cladding integrity from the force exerted by internal gas pressure.

Q2. Yes : ‘non lift-off’ criterion is defined.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative

Q6. Unaffected so far.

Spain

Q1. ***Definition:*** fuel rod internal pressure should remain below the nominal coolant pressure during normal operation unless other criterion justified.
Rationale: prevent the loss of dimensional stability and cladding integrity from the force exerted by internal gas pressure.

Q2. Yes: the gap should not reopen.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Sweden

Q1. ***Definition:*** The internal pressure of the fuel rod caused by the release of fission gases and pre-pressurisation must remain so low that the internal pressure of the rod will not cause the gap between the pellet and the cladding to increase.
Rational: This criterion limits outward creep of the cladding and loads of the cladding during normal operation transients and accidents, which may cause premature and extensive fuel failure.
Limit defined: Rod internal pressure shall not cause the gap between the pellet and the cladding to increase.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. Conservative and statistical methods.

Q6. High burnup effects, which are related to fission gas releases, have been taken into account in fuel performance codes (as far as they are known).

PCMI

Finland

Q1. ***Definition:*** The failure probability of the fuel rods caused by the PCMI during anticipated transients and accidents shall be low.
Rational: provides basis for cladding integrity by avoiding PCMI type failures.
Limit defined: Explicit safety limit have not been defined. Licensee must show that fuel failure probability during anticipated transients and accidents due to PCMI is small.

Q2. Linear heat generation rate (LHGR) shall not exceed the thermal mechanical limit specified as a function of fuel burnup. This curve has defined for each fuel design separately. RIA fuel failure limit is based on PCMI failure. RIA fuel failure limit is 140 cal/gU.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion (licensed fuel bundle maximum burnup is 40 MWd/kg U).

Q5. Best estimate and conservative methods.

Q6. Fuel design codes have updated to cover models for phenomena, which are relevant for fuel burnup up to maximum licensed fuel burnup (licensed maximum fuel bundle burnup 40 MWd/kgU)

Germany

Q1. According to KTA rule 3101.2 the power density change is to be limited such that the ocal failure thresholds in terms of power density are not exceeded. These thresholds are identified in specific power ramp tests. They depend on both local thermal hydraulic condition and burn-up.
The cladding strain for a follow up operation is limited up to 2.5% of the equivalent plastic strain of the cladding.
Rationale: prevent the failure of cladding due to PCMI.

Q2. No.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. The experimental data base (ramp tests) has to be kept on a relevant burn-up level.

Hungary

Q1. For the limitation of the pellet clad mechanical interaction there is no criteria in Hungary at present. The reason is that the degradation of the mechanical properties of the Zr1%Nb cladding material is rather small up to the burnups presently licensed in Hungary and the experience with this material proved that the DNBR criterion is appropriate to avoid fuel failure in an AOO.

Q2. No.

Q3. No.

Q4. For the BNFL VVER-440 fuel design it has been proved that the 1% strain limit is not exceeded.

Q5. –

Q6. –

Japan

Normal and anticipated transients

Q1. ***Definition:*** Fuel assembly should be designed to maintain its integrity considering the various practical effects through the lifetime in reactor.
Rational: provides basis for cladding integrity by avoiding fuel cladding mechanical failures.
Limit defined: Circumferential plastic strain of cladding is less than 1%.

Q2. Linear heat generation rate (LHGR) shall not exceed the thermal mechanical limit.

Q3. No.

Q4. Unaffected

Q5. Conservative methods

Q6. Unaffected

In RIE (Reactivity Initiated Events) conditions

Q1. ***Definition:***. In cases of abnormal transients and accidents, even if fuel fragmentation and mechanical energy generation due to the failure would occur, the shutdown capability and integrity of the reactor pressure vessel should not be threatened.
Rational: Fuel fragmentation and small pressure pulses were observed at low energy PCMI failure of high burnup fuels. Therefore, the evaluation of mechanical energies generated by PCMI failure was added in the safety assessment of RIE to confirm the integrity of reactor pressure vessel.
Limit defined: fuel burnup(BU)<25GWd/tU: fuel enthalpy increase =<110cal/g
25GWd/tU= <BU<40GWd/tU: =<85cal/g
40GWd/tU= <BU<65GWd/tU: =<50cal/g
65GWd/tU= <BU<75GWd/tU: =<40cal/g (provisional)

Q2. No.

Q3. No.

Q4. PCMI failure threshold of high burnup fuels affected.

Q5. Conservative methods.

Q6. Unaffected.

Korea

Q1. ***Definition:*** the following 2 criteria are applied, though they are not sufficient ;
the uniform strain(including elastic and plastic) during transient condition < 1%,
no fuel melting.
Rationale: prevent the failure of cladding from the excessive pellet-cladding interaction.

Q2. Yes : the above 2 criteria, however, is not separately defined as criteria for PCI.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative method.

Q6. Unaffected so far.

Spain

Q1. ***Definition:*** failure probability due to PCMI shall be low the following 2 criteria are applied, though they are not sufficient; the uniform strain (including elastic and plastic) during transient condition < 1%, no fuel melting.
Rationale: prevent the failure of cladding due to PCMI.
Limit: no explicit limit defined.

Q2. No.

Q3. No.

Q4. Unaffected so far.

Q5. –

Q6. Need for upgraded models to reflect the high burnup phenomena.

Sweden

Q1. ***Definition:*** The failure probability of the fuel rods caused by the PCMI during anticipated transients and accidents shall be low.
Rational: This provides basis for cladding integrity by avoiding PCMI type failures
Limit defined: Explicit safety limit have not been defined. Licensee must show that fuel failure probability during anticipated transients and accidents due to PCMI is small.

Q2. Linear heat generation rate (LHGR) shall not exceed the thermal mechanical limit specified as a function of fuel burnup. This curve is defined for each fuel design separately.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. Best estimate and conservative methods.

Q6. Fuel design codes have been updated to cover models for phenomena, which are relevant for fuel burnup up to maximum licensed fuel burnup.

Ria fragmentation

Finland

Q1. ***Definition:*** The fragmentation and melting of the fuel rod shall be prevented.
Rational: provides basis for core coolability by avoiding fuel fragmentation.
Limit defined: The fuel enthalpy of any fuel rod may not exceed the value of radial average enthalpy, 963 J/g UO_2 (230 cal/g).

Q2. No.

Q3. No.

Q4. Criterion unaffected so far (licensed fuel bundle maximum burnup is 40 MWd/kg U).

Q5. Conservative methods.

Q6. Unaffected due to the low fuel burnup limit (licensed fuel bundle maximum burnup 40 MWd/kgU).

Germany

Q1. There is no maximum energy deposition value defined in the regulation. The German licensing practice requires that no fuel rod would fail during a rod ejection accident (PWR) and during rod drop accident (BWR), respectively.
Rationale: prevent the failure of cladding due to RIA.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. Best-estimate methods using 3D neutronic analysis.

Q6. Unaffected so far.

Hungary

Q1. According to the criterion the energy increase shall not at any location in any fuel rod cause the exceeding of the radial average enthalpy value 963 J/gUO2 (230 cal/g). The criterion has the function of limiting the energetic fuel-coolant interaction in case of postulated accidents.

Q2. No.

Q3. No.

Q4. No change up to the present.

Q5. The evaluation method is conservative.

Q6. No change up to now.

Japan

Q1. ***Definition:*** Two criteria exist. In order to avoid fragmentation due to fuel melting, a safety limit ensures that maximum radially averaged fuel enthalpy must not exceed 230 cal/g minus x cal/g corresponding reduction of melting point due to burnup and additives. Another criterion defines amount of fragmented fuel pellets and mechanical energy generation caused with PCMI failure of burnup fuels.
Rational: to avoid fragmentation due to fuel melting. Fragmentation due to PCMI failure is allowed, but licensees must show tolerable consequences in their evaluation with assumptions defined in the criterion.
Limit defined: 230 cal/g minus x cal/g corresponding reduction of melting point due to burnup and additives.

Q2. No: RIAfuel failure limit regarding PCMI failure (PCMI in RIA conditions in Cat. I) gives number of failed rod, and then amounts of fragmented fuels.

Q3. No.

Q4. Criteria significantly affected. Enthalpy corresponding to decrease of fuel melting point must be reduced from absolute enthalpy limit, 230 cal/g. PCMI failure threshold was reduced for high burnup fuels. Regarding fragmentation due to PCMI failure, amount of fragmented fuel pellets and evaluation formula for mechanical energy generation were newly adopted.

Q5. Conservative.

Q6. Unaffected so far.

Korea

Q1. ***Definition:*** a radially averaged maximum enthalpy of fuel rod should be less than 280 cal/g.
Rationale: prevent widespread fragmentation and dispersal of the fuel and generation of pressure pulses in the primary system.

Q2. Yes: but the licensed limits are different with plant type. For the plants originated from Westinghouse design, limit of 225 cal/g and 200 cal/g is defined for unirradiated and irradiated fuel rod, respectively. For the plants originated from CE design, 280 cal/g is defined.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative method.

Q6. Unaffected so far.

Spain

Q1. ***Definition:*** fragmentation and melting should be avoided.
Rationale: maintain core coolability and primary system integrity.
Limit: the radially averaged fuel enthalpy should not be higher than 280 cal/g.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. Conservative methods.

Q6. Unaffected so far.

Sweden

Q1. ***Definition:*** The fragmentation and melting of the fuel rod shall be prevented.
Rational: This provides basis for core coolability by avoiding fuel fragmentation.
Limit defined: The fuel enthalpy of any fuel rod may not exceed the value according to the table below.

Table. Limit for fuel fragmentation.

Pellet Burnup [Mwd/kgUO_2]	Radial Average Enthalpy [cal/gUO_2]
0	230
33	230
40	100
50	60
60	30

Q2. No.

Q3. No.

Q4. Criterion unaffected.

Q5. Conservative methods.

Q6. Unaffected due to restriction in control rod worth and. low fuel burnup in Swedish reactors.

Non runaway oxidation

Finland

Q1. Definition: This criterion is not used in Finland
Rational: Maximum temperature limit for cladding during accidents is 1200 C .
Limit define: –

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Germany

Q1. –

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Hungary

Q1. There is no existing criterion for the limitation of the oxidation for example in case of main coolant pump locked rotor accident. It is expected and safety analysis confirmed that the DNBR criterion and the temperature reactivity feedback are able to limit the oxidation at acceptable level.

Japan

Q1. –

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Korea

Q1. No : it is believed that the core can maintain coolable geometry during Non-LOCA accidents.

Q2. Though a definite limit is not defined in regulatory guide, the 2700°F criterion is used for the locked rotor accidents(Westinghouse plants) to assure core coolability.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative method.

Q6. Unaffected so far

Spain

Q1. No. Clad temperature is limited to 2700º F in all cases.

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Sweden

This criterion is not used.

Q1. –

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

LOCA pct

Finland

Q1. ***Definition:*** The highest temperature of the cladding may not exceed 1200 ºC in case of an accident
Rational: The temperature rise of the cladding has been limited to the level, below which the oxidation of the cladding as a consequence of metal-water reaction is still controllable.
Limit defined: The highest temperature of the cladding may not exceed 1200 ºC in case of an accident.

Q2. LHGR for fresh fuel in WWER reactors is limited by LOCA. Maximum LHGR limit for fresh fuel is 325 W/cm.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion (licensed fuel bundle maximum burnup is 40 MWd/kg U).

Q5. Conservative method.

Q6. Unaffected due to low fuel burnup limit.

Germany

Q1. ***Definition***: the maximum PCT shall be maintained below 1200°C.
Rationale: to avoid cladding embrittlement by excessive oxidation due to fuel-coolant interaction, and to maintain coolable core geometry.

Q2. PCT < 1200ºC

Q3. No.

Q4. Unaffected.

Q5. Conservative methods.

Q6. Unaffected.

Hungary

Q1. The definition of the criterion is as follows. The highest temperature of the cladding reached in accident conditions shall not exceed 1200°C. The 1200°C temperature limit is one key parameter ensuring that cladding embrittlement remains acceptable.

Q2. No.

Q3. Radial and axial peaking factors together with the maximum permitted linear heat rate contribute to the fulfilment of the safety criterion. The limiting values depend on fuel design and they are given in the Tech. Spec.

Q4. There is no change so far.

Q5. Evaluation methodology is conservative.

Q6. No influence.

Japan

Q1. ***Definition:*** Safety limit ensures that the maximum cladding temperature shall not exceed 1200 ºC.
Rational: to avoid excessive embrittlement of cladding.
Limit defined: 1200 ºC

Q2. Yes: 1200 ºC

Q3. No.

Q4. Criteria unaffected so far.

Q5. Conservative.

Q6. Unaffected so far.

Korea

Q1. ***Definition:*** the maximum PCT shall maintain below 2200^{o}F

Rationale: to avoid cladding embrittlement due to excessive oxidation and preserve a coolable core geometry after LOCAs.

Q2. Yes : the maximum PCT limit of 2200^{o}F is defined.

Q3. Vendor and utility also use this criterion for fuel and ECCS design.

Q4. Unaffected so far.

Q5. Conservative method are used for the licensing calculation. Best-estimate methods are also used for analyzing realistic behaviors after LOCAs and evaluating safety margin. One best-estimate method is under review for the approval to the licensing calculation.

Q6. Unaffected so far.

Spain

Q1. ***Definition:*** the maximum PCT shall be maintained below 2200^{o}F.
Rationale: to avoid cladding embrittlement by excessive oxidation due to fuel-coolant interaction, and to maintain coolable core geometry.
Limit: the maximum PCT is limited to 2200^{o}F.

Q2. PCT<2200º F

Q3. No.

Q4. Unaffected.

Q5. Conservative methods.

Q6. Unaffected.

Sweden

Q1. ***Definition:*** The highest temperature of the cladding may not exceed 1204 ºC in case of an accident
Rational: The temperature rise of the cladding has been limited to the level, below which the oxidation of the cladding as a consequence of metal-water reaction is still controllable.
Limit define: The highest temperature of the cladding may not exceed 1204 ºC in case of an accident.

Q2. LHGR limit for BWR fuel and Fq limit for PWR fuel.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. Conservative method.

Q6. Unaffected.

LOCA oxidation

Finland

Q1. ***Definition:*** An excessive embrittlement of the cladding shall be prevented.
Rational: provides basis for core coolability and handling and transportation of the fuel after the accident.
Limit defined: It shall be shown that

- the cladding is not oxidized during an accident to the degree that it cannot withstand the loads caused by the accident, for example stresses due to thermoshock during quenching at the late phase of a loss of coolant accident. In estimating the total thickness of the required ductile part of the cladding, attention has to be paid to the external and possible internal oxidation of the cladding during the accident and to the preceding oxidation during normal operation. Additionally, the chemical interactions between uranium dioxide and cladding material in connection with cladding collapse have to be taken into consideration. Also the loads caused by the handling, removal transport and storage of the fuel rod bundle after the accident have to be included in this assessment;
- the oxygen adsorbed during normal operation and during an accident does not excessively embrittle the cladding. The effect of the adsorbed oxygen on the cladding shall be experimentally determined.

Q2. LHGR limit for fresh fuel in WWER reactors is limited by LOCA. Maximum LHGR limit is 325 W/cm.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion (licensed fuel bundle maximum burnup is 40 MWd/kg U.

Q5. Conservative method.

Q6. Unaffected due to low fuel burnup limit.

Germany

Q1. According to RSK guidelines the maximum oxidation depth of the cladding calculated will not exceed 17% of the actual cladding tube wall thickness at any point. This provision ignores the corrosion layer which might have build up during normal operation because it is assumed that this layer does not contribute to both the strength of the cladding and the protection against oxidation..
Rationale: coolable core geometry and handling of fuel after LOCA.

Q2. Yes: the 17% ECR is used.

Q3. No.

Q4. Unaffected so far. Investigation for Niob alloys are underway.

Q5. Conservative methods.

Q6. Unaffected so far.

Hungary

Q1. Corresponding to the criterion the oxidation level of the cladding shall nowhere exceed 17% of the cladding wall thickness. This means that locally 17% of the Zr is converted into ZrO2. The oxidation limit is the second parameter responsible for the acceptable degree of embrittlement.

Q2. No.

Q3. See LOCA-PCT point 3.

Q4. The official Hungarian limit is not changed. However, the Russian vendor and also Hungarian tests proved that for the Zr1%Nb cladding material the 18% limit ensures acceptable conservatism.

Q5. The evaluation methodology is conservative.

Q6. No affect so far.

Japan

Q1. ***Definition:*** Safety limit ensures that the calculated stoichiometric oxidation shall not exceed 15% of the cladding thickness.
Rational: to avoid excessive embrittlement of cladding.
Limit defined: 15%ECR.

Q2. Yes: 15%ECR.

Q3. No.

Q4. Criteria unaffected so far.

Q5. Conservative.

Q6. Unaffected so far.

Korea

Q1. ***Definition:*** the total oxidation(during normal and transient condition) after LOCA shall be less than 17% of the cladding thickness before oxidation.
Rationale: same as the LOCA-PCT.

Q2. Yes : the 17% limit is defined.

Q3. Vendor and utility also use this criterion for fuel and ECCS design.

Q4. Unaffected so far.

Q5. Conservative method.

Q6. Unaffected so far.

Spain

Q1. ***Definition****:* excessive clad embrittlement should be prevented.
Rationale*:* coolable core geometry and handling of fuel after LOCA.
Limit: the total oxidation(during normal and transient condition) after LOCA shall be less than 17% of the cladding thickness before oxidation.

Q2. Yes: the 17% limit is used.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Sweden

Q1. ***Definition:*** An excessive embrittlement of the cladding shall be prevented.
Rational: This provides basis for core coolability and handling and transportation of the fuel after an accident.
Limit defined: It shall be shown that the cladding is not oxidised during an accident to the degree that it cannot withstand the loads caused by the accident, for example stresses due to thermal shock during quenching at the late phase of a loss of coolant accident. In estimating the total thickness of the required ductile part of the cladding, attention has to be paid to the external and possible internal oxidation of the cladding during the accident and to the preceding oxidation during normal operation. Also the loads caused by the handling, removal transport and storage of the fuel rod bundle after the accident have to be included in this assessment.

Q2. The total cladding oxidation should not exceed 17 %.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. Conservative method.

Q6. Unaffected.

LOCA H release

Finland

Q1. ***Definition:*** The amount of hydrogen caused by the chemical interaction between coolant and cladding shall not exceed 1% of the amount, which could be developed, if the whole active cladding surrounding by fuel pellets would react with the coolant.
Rational: This criterion limits the amount of an hydrogen in the primary circuit and the containment during the accident. It also limits the number of failed fuel rods in the core.
Limit define: The amount of hydrogen caused by the chemical interaction between coolant and cladding shall not exceed 1% of the amount of hydrogen, which could be developed, if the whole active cladding surrounding fuel pellets would react with the coolant.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. –

Q6. Unaffected.

Germany

Q1. ***Definition:*** the total amount of hydrogen generation by clad-coolant interaction shall be less than 1% of the hypothetical amount generated by the complete metal-water reaction in the core.
Rationale: to prevent excessive hydrogen generation that could threaten containment integrity.

Q2. The 1% limit is defined.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Hungary

Q1. The criterion limits the amount of hydrogen released during the accident. The maximum permitted value is 1% of the hydrogen which would be generated if the entire core cladding material were converted into ZrO_2 during the coolant-cladding interaction. The limitation determines one parameter for the design of containment integrity.

Q2. No.

Q3. No.

Q4. No.

Q5. Evaluation is conservative.

Q6. No.

Japan

Q1. ***Definition:*** Safety limit ensures that the amount of hydrogen generation by metal-water reaction shall be sufficiently low from the viewpoint of maintaining integrity of the containment vessel.
Rational: to limit the contribution of metal-water reaction for all the hydrogen generation rate to a certain level.

Q2. Yes: but not directly associated. The amount of hydrogen generation shall be sufficiently lower than 1% of total hydrogen generation in the core.

Q3. No.

Q4. Criteria unaffected so far.

Q5. Conservative.

Q6. Unaffected so far.

Korea

Q1. ***Definition:*** the total amount of hydrogen generation shall be less than 1% of the hypothetical amount generated by the metal-water reaction in the core.
Rationale: to prevent the excessive hydrogen generation threatening containment integrity.

Q2. Yes : the 1% limit is defined.

Q3. Vendor and utility also use this criterion for fuel and ECCS design.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Spain

Q1. ***Definition:*** the total amount of hydrogen generation by clad-coolant interaction shall be less than 1% of the hypothetical amount generated by the complete metal-water reaction in the core.
Rationale: to prevent excessive hydrogen generation that could threaten containment integrity.

Q2. The 1% limit is defined.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Sweden

Q1. ***Definition:*** The amount of hydrogen caused by the chemical interaction between coolant and cladding shall not exceed 1% of the amount, which could be developed, if the whole active cladding surrounding by fuel pellets would react with the coolant.
Rational: This criterion limits the amount of an hydrogen in the primary circuit and the containment during the accident. It also limits the number of failed fuel rods in the core.
Limit defined: The amount of hydrogen caused by the chemical interaction between coolant and cladding shall not exceed 1% of the amount of hydrogen, which could be developed, if the whole active cladding surrounding fuel pellets would react with the coolant.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup have not been affected the criterion.

Q5. Not applicable.

Q6. Unaffected.

LOCA long term cooling

Finland

Q1. ***Definition:*** The nuclear power plant shall be brought to and maintained for long term at a safe and stable shutdown state after accident conditions.
Rational: Provides basis for decay heat removal after accidents.
Limit define: no explicit limit define.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup has not been affected the criterion.

Q5. Conservative.

Q6. Unaffected.

Germany

Q1. ***Definition:*** After LOCA the core temperature shall be maintained at acceptable low values and the long term decay heat removal must be assured.
Rationale: to remove the decay heat.
Limit: there is no explicit limit.

Q2. No.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Hungary

Q1. According to the criterion the fuel flow channels shall not become blocked so that the coolability of fuel is endangered. Preservation of the coolability for long term is the only way to avoid accident progression. The degree of the permissible blockage is design dependent. Hungarian experiments revealed that in a spectrum of LOCA conditions the blockage never exceeded 80% of the flow area and the remaining 20% is enough for sufficient cooling and removal of the remanent heat in case of Zr1%Nb cladding.

Q2. No.

Q3. No.

Q4. For the new BNFL VVER-440 fuel the vendor proved that the coolability is preserved.

Q5. Evaluation is conservatively made.

Q6. No.

Japan

Q1. ***Definition:*** Safety limit ensures the removal of decay heat for long time in any case of change in the fuel geometry.
Rational: to remove decay heat from FP including long-lived isotopes, which is one of the main heat sources during the accident.

Q2. No.

Q3. No.

Q4. Criteria unaffected so far.

Q5. Conservative.

Q6. Unaffected so far

Korea

Q1. ***Definition:*** After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.
Rationale: to remove the decay heat and keep the core low temperature.

Q2. Yes: not explicit criteria on the core temperature and the time period. The long-term cooling capability is assured by core temperature after reflood, no re-criticality, no boron precipitation.

Q3. Vendor and utility evaluate long-term cooling capability of ECCS by using same definition.

Q4. Unaffected so far.

Q5. Conservative method.

Q6. Unaffected so far.

Spain

Q1. ***Definition:*** After an accident, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.
Rationale: to remove the decay heat.
Limit: there is no explicit limit.

Q2. No.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Sweden

Q1. ***Definition:*** The nuclear power plant shall be brought to and maintained for long term at a safe and stable shutdown state after accident conditions.
Rational: Provides basis for decay heat removal after accidents.
Limit defined: No explicit limit defined.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup has not been affected the criterion.

Q5. Conservative.

Q6. Unaffected.

Seismic loads

Finland

Q1. ***Definition:*** This criterion is not relevant in Finland, only 0.1 g acceleration have been taken into account in design of structures and components.
Rational: Finland is seismically very stable area.
Limit : –

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Germany

Q1. ***Definition:*** For LOCA condition no seismic loads are considered additionally. Generally the components of the plant have to be designed such that seismic loads can be taken without detrimental consequences.
Rationale: to retain the capability to bring the reactor to a safe state and maintain coolable geometry of fuel assemblies.

Q2. No.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Hungary

Q1. Concerning the seismic loads it is required to prove that the fuel and core structures are not endangered in case of an earthquake. Combined seismic and LOCA loads are not considered. The frequency of the design earthquake is 10-4/y.

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Japan

Q1. –

Korea

Q1. ***Definition:*** core coolability and control rod insertability should be assured under seismic loads.
Rationale: bring the reactor to a safe state and retain the coolable geometry of fuel assemblies under seismic loads.

Q2. Yes : according to the above requirement, the following criteria are defined:
(a) no fuel rod fragmentation
(b) no deformation of fuel assembly impairing control rod insertion
(c) no crushing of spacer grid affecting cooling geometry of fuel assembly

Q3. No.

Q4. Unaffected so far.

Q5. Conservative method.

Q6. Unaffected so far.

Spain

Q1. ***Definition**:* core coolability and control rod insertability should be assured under seismic loads. ***Rationale:*** to retain the capability to bring the reactor to a safe state and maintain coolable geometry of fuel assemblies.

Q2. (a) no fuel rod fragmentation,
(b) no deformation of fuel assembly impairing control rod insertion
(c) no crushing of spacer grid affecting cooling geometry of fuel assembly.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Sweden

Q1. ***Definition:*** This criterion is only used for 2 reactors in Sweden (Forsmark 3 and Oskarshamn 3). A Swedish response spectrum is used.
Rational: Sweden is seismically a very stable area.
Limit defined: Plastic deformation is not allowed.

Q2. No.

Q3. No.

Q4. No.

Q5. Conservative methods.

Q6. Unaffected.

Hold-down forces

Finland

Q1. ***Definition:*** This criterion is not relevant in Finnish NPPs.
Rational: Fuel bundle is surrounded by the fuel boxes both in BWRs and WWERs and structure of the bundle does not include holddown springs.
Limit defined: –

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Germany

Q1. ***Definition:*** the holddown capability of the fuel assembly should exceed the vertical hydraulic lift-off loads during normal operation.
Rationale: prevent fuel assembly lift-off from the lower core plate.
Criteria: non lift-off of fuel assembly due to hydraulic loads during normal operation and anticipated operational occurrences, with the exception of pump overspeed.

Q2. –

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Hungary

Q1. At present there is no Cat. I criterion in Hungary. When a new fuel design is approved then the vendor proves that the assembly will not lift-off in operational conditions.

Q2. –

Q3. –

Q3. –

Q4. –

Q5. –

Q6. –

Japan

Q1. –

Korea

Q1. ***Definition:*** the holddown capability of the fuel assembly should exceed the vertical hydraulic lift-off loads during normal operation.
Rationale: prevent fuel assembly lift-off from the lower core plate.

Q2. Yes : non lift-off of fuel assembly due to hydraulic loads during normal operation and anticipated operational occurrences, with the exception of pump overspeed, is defined.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative or statistical method is used according to fuel type.

Q6. Unaffected so far.

Spain

Q1. ***Definition****:* the holddown capability of the fuel assembly should exceed the vertical hydraulic lift-off loads during normal operation.
Rationale: prevent fuel assembly lift-off from the lower core plate.
Criteria: non lift-off of fuel assembly due to hydraulic loads during normal operation and anticipated operational occurrences, with the exception of pump overspeed.

Q2. –

Q3. No.

Q4. Unaffected so far.

Q5. Conservative methods.

Q6. Unaffected so far.

Sweden

Q1. ***Definition:*** The PWR fuel assembly should not lift from the bottom nozzle.
Rational: If the fuel lift from the bottom nozzle the cooling of the assembly is deteriorated.
Limit defined: No limit is defined, but the hold down forces should not plastically deform the assembly.

Q2. No.

Q3. No.

Q4. The criterion is unaffected, but because of the problems with bowing of the fuel in the Ringhals PWR reactors it is of concern when buying or loading new fuel assemblies.

Q5. Conservative methods.

Q6. The methods have been developed to more adequate calculate the hold down force and its influence on the fuel structure.

Criticality

Finland

Q1. ***Definition:*** Criticality during the fuel transportation, storage and handling shall be prevented.
Rational: Provides basis for criticality safety for fuel transportation, storage and handling.
Limit defined: The margin to criticality shall be at least 5% deltaK/K taking into account certain accident conditions specified in the guide.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup has not been affected the criterion.

Q5. Conservative method.

Q6. Unaffected.

Germany

Q1. ***Definition:*** criticality should be avoided during fuel transportation, storage and handling.
Limit: margin to criticality of at least 5% $\Delta K/K$.

Q2. No.

Q3. No.

Q4. Credit to burnup is needed due to fuel enrichment increase.

Q5. Conservative methods.

Q6. Need for experiments and code validation at high burnup values.

Hungary

Q1. There is a requirement that for the storage and transport facilities it has to be proved that the value of k_{eff} remains below 0.95 in the worst condition. If single failure is considered in the accident scenarios the k_{eff} value shall be less than 0.98.

Q2. –

Q3. –

Q4. –

Q5. –

Q6. –

Japan

Q1. ***Definition:*** Criticality during the fuel transportation, storage and handling shall be prevented.
Rational: Provides basis for criticality safety for fuel transportation, storage and handling.

Q2. No.

Q3. No.

Q4. Unaffected.

Q5. Conservative method.

Q6. Unaffected.

Korea

Q1. ***Definition:*** safety limit of effective multiplication factor keff ensures that fuel storage facility will not be critical during normal and credible abnormal conditions.
Rational and Limit: analysed keff shall not exceed 0.95 or 0.98(optimum moderation).

Q2. Yes: operating limit(Tech. Spec.) boron concentration of fuel pool water and fuel burn-up limit to enrichment defined.

Q3. No.

Q4. Unaffected so far.

Q5. Conservative, statistical method.

Q6. Unaffected.

Spain

Q1. ***Definition****:* criticality should be avoided during fuel transportation, storage and handling.
Limit: margin to criticality of at least 5% Delta K/K. If credit to soluble Boron is given, reactivity without Boron should be less than unity.

Q2. The Tech Specs contain the limits on Boron concentration and fuel burnup (if credit to burnup is given).

Q3. No.

Q4. Credit to burnup is needed due to fuel enrichment increase.

Q5. Conservative methods.

Q6. Needs for experiments and code validation at high burnup values.

Sweden

Q1. ***Definition:*** Criticality during the fuel transportation, storage and handling shall be prevented.
Rational: Provides basis for criticality safety for fuel transportation, storage and handling.
Limit: The value of k_{eff} shall be below 0.95 for normal conditions and the value of k_{eff} shall be less than 0.98 for accident conditions.

Q2. No.

Q3. No.

Q4. New fuel/core designs including fuel burnup has not been affected the criterion.

Q5. Conservative method.

Q6. Unaffected.

B. OPERATIONAL/LICENSING CRITERIA

CPR/DNB operating limit

Hungary

Operational DNBR limit is not existing. The hot subchannel outlet enthalpy is limited. It shall not exceed the saturation enthalpy belonging to the operational pressure. The aim of the limit is to avoid cavitation caused corrosion of the cladding.

LGRH limit

No reponses.

PCI

Hungary

According to the requirement the fuel design shall take into account the stress corrosion phenomenon and ensure that the PCI condition exists only for acceptable period of time. To avoid PCI caused fuel failure the operational instructions shall include limits for the manoeuvring. The operational instructions are design specific.

Coolant activity

Hungary

To limit the number of leaking fuel rods the coolant activity is limited. The presently accepted value is

$3.7 \cdot 10^{7}$ Bq/l for I-131, and

$3.7 \cdot 10^{9}$ Bq/l for the total activity.

Gap activity

Hungary

The gap activity in the safety analysis is determined according to the methodology proposed in the following EU research report:

"Realistic methods for calculating the releases and consequences of a large LOCA", report EUR 14179 EN, 1992.

Source term

Hungary

According to the procedure accepted at present for the determination of source term in the safety analysis the release of the gap activity from the whole core has to be supposed in case of a large break LOCA.

Control rod drop time

No reponses.

RIA fuel failure limit

Hungary

In spite of the fact that in most of cases the DNBR criterion ensures the prevention of fuel failure there is a requirement in Hungary (Cat. I.) limiting the enthalpy rise in AOOs. The limit is 586 J/gUO_2 (140 cal/g) and it means the radially averaged enthalpy at the worst local position.

It should be added that as in most of the countries there is another fuel failure criterion (Cat. I.) namely the melting point of the fuel. According to this criterion the fuel centerline temperature must not reach the melting point at the hot spot of the fuel pins.

Cladding stability

No reponses.

C. DESIGN CRITERIA

Crud deposition

Hungary

General rules are formulated as design criteria. According to these rules the vendor shall prove that:

- crud deposition;
- stresses and hydriding;
- fretting and wear;
- diameter increase and elongation

will remain below acceptable levels during operation and if it is necessary, also in AOO conditions.

Stress/strain

Japan

BWR:

Primary stress shall not exceed the yield stress. Primary plus secondary stress shall not exceed the tensile strength.
Circumferential plastic strain of cladding is less than 1%.

PWR:

Stress shall not exceed the yield stress.
Circumferential plastic strain of cladding is less than 1%.

Oxidation

No reponses.

Hydride concentration

No reponses.

Tranports loads

Japan

PWR:

Fuel assembly should not be remarkably deformed under the 6G load during transportation (G: gravitational acceleration).

BWR:

No criteria.
The channel box surrounds fuel assembly.

FA fretting corrosion/wear

Japan

BWR:

No criteria.
Spacer springs keep fuel rod in contact with spacer.

PWR:

To avoid fuel failure due to fretting corrosion.
(Typical fretting corrosion thickness shall be lower than 10% of cladding thickness.)

Clad diameter increase

No reponses.

Cladding elongation

No reponses.

Radial peaking factor

No reponses.

3 D peaking factor

Japan

BWR:

No criteria.

PWR:

FQ×P should be less than 2.32
where
FQ: Total power peaking factor
P: Relative power.

PART II

Q1. ***What (regulatory approved) fuel designs are in use ?***
Q2. What fuel designs are being considered for the future ?

Finland

Q1. Current approved designs:
WWER 440: Zr 1% Nb design (Russian design) and standard zircaloy 4 cladding based design (BNFL) .
BWR: 10x10 design, late β-quenched cladding with and without inner liner or triclad cladding, some fuel types also new spacer designs.

Q2. Future designs: No plans for future designs.

Germany

The burn-up increases stepwise. For UO2 the following values are reached:
PWR: average batch discharge burn-up: 52 GWd/t, average fuel assembly burn-up: 60 GWd/t.
BWR: average batch discharge burn-up licensed: 50 – 53 GWd/t, average fuel assembly burn-up licensed: 55-57 GWd/t. A few lead assemblies have reached 62 GWd/t.

Enrichments of up to 4.6 wt% U235 resulting in average batch discharge burn-up values of 61 GWd/t are developed and in operation in PWR.

For one BWR fuel element enrichment of 4.1 wt% U235 has been approved leading to an average batch discharge burn-up value of 53 GWd/t.

Hungary

Q1. The currently approved designs are:
Russian type VVER-440 assembly with hexagonal lattice and Zr1%Nb cladding
BNFL VVER-440 design with cladding made of Zry-4 (low tin content).

Q2. In the future it is expected that the cladding material will change (e.g. Russian E635).

Japan

Q1. BWR 8x8, 9x9 Fuel
PWR 14x14, 15x15, 17x17 Fuel.

Korea

Q1. Currently Approved Designs:
Fuel type originated from Westinghouse design : 14X14 type, 16X16 type, 17X17 type, low-Sn Zircaloy-4 cladding
Fuel type originated from CE design : 16X16 type, low-Sn Zircaoy-4 cladding.

Q2. Fuel Design Considered for the Future:
ZIRLO cladding is about to be loaded.

Advanced fuel assembly for the Korean Standard Nuclear Plant is under development.

Spain

Q1. Currently Approved Designs:
Westinghouse fuel designs manufactured by ENUSA: 14X14 LO-LOPAR, 17X17 AEF and MAEF with Zirlo cladding and skeleton.
General Electric designs manufactured by ENUSA: GE-11, GE-12 and GE-14.
Siemens fuel manufactured by Siemens and ENUSA: 16X16 with Duplex or Zirlo cladding.
ABB fuel: SVEA-96.

Q2. Fuel Designs Being Considered for the Future.
Framatome AFA-3G with M5 cladding (demos loaded).
Siemens ATRIUM fuel (project to load demos).

Sweden

Q1. Current approved designs that are used:
BWR: SVEA-64, SVEA-100, SVEA96S, SVEA 96S Optima, KWU 9x9, ATRIUM-9, ATRIUM-10B, GE 11S, GE 12S, GE 14
PWR: AFA-2G 17x17, AFA-3G 15x15 and 17x17, Siemens HTP 17x17, Westinghouse performance + 15x15, Siemens Focus 15x15.

Q2. Future designs:
No plans for future designs.