

# Loss of Coolant Accident

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Mr Peter Rudling, Chairman of the Board of ANT International

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## Executive Summary

For decades, the design safety acceptance criteria defined in 1972 by US-NRC to deal with design-basis accidents (DBA) remain unchanged, despite significant evolution of the fuel management schemes adopted in most of the commercial Light Water Reactors (LWRs) around the world. Fuel discharge burnups, for instance, increased progressively from 33GWd/tM in 1972 to more than 60GWd/tM 20 years later without questioning the applicable design limits. The fuel community waited for the unexpected result of the RIA test REPNa1 carried-out in October 1993 in CABRI on a high burnup fuel rod (64GWd/tM), to realize that burnup impacts significantly the fuel behavior in transients and may have a direct impact on standard DBAs safety acceptance criteria. Immediately, most of the regulators launched ambitious R&D programs to document the burnup effects on fuel behaviors during DBA transients with the objective to update the design acceptance criteria of the 2 most impacted accidents, Reactivity Initiated Accidents (RIA) and Loss of Coolant Accidents (LOCA). In April 2006, a semi-integral LOCA test has been carried-out in Halden test reactor on a high burnup (91GWd/tM): the test reveals new phenomena such as fuel fragmentation and relocation in the ballooned region and fuel fragments dispersal in the coolant.

The present document aims to update the previous ZIRAT report on LOCA, dated 2005. It accounts for the considerable knowledge that had been accumulated in the intervening period for water-cooled reactors (PWR, BWR and VVER).

Large amounts of new experimental data have been gathered and summarized. The description of computer codes and ECCS evaluation models has been updated and an overview is given on updated LOCA safety acceptance criteria in USA, France, USA, Belgium, Czech Republic, Canada, Belgium, Germany, UK, Hungary, Japan, Spain, Sweden, Switzerland.

### LOCA, a key Design Basis Accident.

The design of nuclear power plants consists in analyzing the consequences of hypothetical accidents so that suitable mitigating systems can be designed and implemented. For design basis accidents (DBAs), a fundamental acceptance criterion is that there should be no or very limited radiological consequences to the public. In order to fulfil this criterion, derivative acceptance criteria have been formulated to design (i) the emergency core cooling systems of the facility (ECCS) and (ii) the nuclear fuel in the reactor core.

Reactivity Initiated Accidents (RIA) are used to cover accidents resulting from excessive nuclear power while Loss-of-coolant accidents (LOCA) are used to address transients resulting from lack of cooling. LOCAs can be differentiated between several categories depending on the size of the postulated break in the primary coolant system. For design basis accidents (DBAs), a double-ended pipe break (called Large Break Loss of Coolant Accident or LB-LOCA) is postulated on one of the cold legs of a PWR or in one of the recirculation pump intake lines of a BWR. In order to mitigate the consequences of this break, it is necessary to design the emergency core cooling systems (ECCS) so that the fuel is cooled efficiently during all phases of the accident. This requirement naturally leads to a criterion ensuring that the structural integrity of the fuel rods is maintained and that the fuel rods geometry remains coolable throughout the whole LOCA sequence and even after, in the long term, to face potential post-LOCA events (e.g., seismic events).

Intermediate or small break LOCA (IB-LOCA and SB-LOCA respectively) cover the spectrum of events where the break in the primary circuit is less than a double-ended pipe break (LB-LOCA) and does not necessarily lead to rapid blow-down and complete uncovering of the core. In some countries, e.g., France in 2017, because double-ended pipe breaks are low occurrence and are precluded by the anti-whipping systems, the regulators agreed that industry could analyze LB-LOCA cases considering realistic hypothesis to assess the radiological consequences of the event. However, as such, the anti-whipping systems must be considered as intrinsic safety components of the reactor coolant system (RCS). In addition, LB-LOCA has been definitively excluded by design for the new NPPs generation, thanks to the break preclusion concept. As a result, penalized LOCA analysis are now focused on IB-LOCA and SB-LOCA. Together with this change of paradigm, the thermo-mechanical design of the reactor structures must be carried out considering the same break size as the one used for the LOCA analysis.

Along the same line, the US-NRC recognized in years 2000s that double-ended guillotine LB-LOCA is highly unlikely and current regulatory treatment is inconsistent with risk insights. US-NRC developed

then the Transition Break Size (TBS) approach, based on the frequencies of the different types of breaches. All LOCAs below the TBS should meet the current 50.46 analysis/acceptance criteria while LOCAs above the TBS are analyzed with relaxed standards. The TBS rulemaking process hasn't come to an end, but discussions are starting again, within a risk-informed framework.

### **Investigating all phenomena occurring during LOCA transients is the new trend.**

Several physical phenomena affect the fuel rods during a LOCA thermal transient: heating rates and peak temperatures have a direct effect on cladding mechanical properties, which in turn will impact ballooning-burst conditions of the fuel rod cladding (ductile mode), while cladding oxidation during the high temperature phase of the LOCA transient will embrittle the cladding due to oxygen and hydrogen pickup (brittle mode).

Most fuel vendors using the 1973 US-NRC standard for LOCA adopted the isothermal parabolic rate constant published by Baker & Just in 1962 to calculate oxygen uptake in cladding exposed to steam during a LOCA thermal transient. This correlation was used to determine the historical LOCA acceptance criteria, i.e., the max Equivalent Cladding Reacted (ECR) and the max Peak Cladding Temperature (PCT), to ensure fuel core coolability. Later data have shown that the Baker-Just (B-J) oxidation correlation is conservative and overpredicts the reaction rate, opposite the more realistic Cathcart-Pawell (C-P) correlation. This is true for modern 1%Nb alloys, but not for Zr4: in case steam pressure remains at high levels (e.g., SB- or IB-LOCA cases), B-J correlation is well suited while C-P correlation tends to underpredict Zr4 cladding oxidation.

### **Improved testing to better define LOCA acceptance criteria**

During the last 2 decades, nuclear industry and the regulators developed new testing conditions to better simulate the physical phenomena occurring during a LOCA transient, to provide fuel performance codes with reliable validation data, and ultimately, to update the acceptance criteria defined in 1973.

Regarding the **ductile mode** (i.e., ballooning and burst of the fuel rod cladding during the heating phase), it has been shown that the behaviour of the fuel cladding is affected by the pre-transient hydrogen content and strongly influenced by the temperature distribution axially and azimuthally. Axial temperature profile is dependent on the LOCA scenario under study while azimuthal temperature gradients are influenced by the three-dimensional heat transfer mechanisms which take place in the surrounding of each fuel rod. The latter, often called "bundle thermal effect" is important as it has a direct impact on the balloon size and subsequent level of flow blockage and fuel relocation (i.e., on PCT). Meaningful experimental simulations must therefore accurately reproduce these physical mechanisms. This implies the use of realistically heated fuel rod simulators, realistic conditions of surface heat transfer, and the use, if possible, of multi-rod assemblies to reproduce the heat-transfer conditions in the sub-channels between rods. These prototypical test conditions are well suited to validate Multiphysics codes, which in turn, enable proper transfer from separate effect tests results to in-reactor conditions.

It must be noted that most of the tests are performed on single rods. However, it has been confirmed that these experimental test conditions are conservative and tend to enhance the effect of the plenum pressure on balloon sizes. On the other hand, co-planar and non-coplanar deformation, with strains up to and including those leading to mechanical interaction between fuel rods have been demonstrated experimentally in LOCA experiments with multi-rod arrays and simulated reflood cooling, but none of them has produced deformations which would inhibit fuel rod cooling. As such, no specific acceptance criteria has been required to limit flow blockage ratio: it has been agreed that the current acceptance criteria (i.e., ECR and PCT) are sufficient to ensure core cooling.

Regarding the **brittle mode** (i.e., cladding embrittlement due to high temperature oxidation), it has been shown that irradiated cladding can be favorably replaced by pre-hydrided and thinned cladding specimens to simulate end-of-life in-reactor corrosion and hydrogen pick-up.

### **Updating historical LOCA acceptance criteria to include new physical phenomena**

In the 1973 Rule-Making Hearing the U. S. Atomic Energy Commission (AEC) staff and commissioners believed that retention of ductility was the best guarantee against potential fragmentation of fuel cladding under various types of not-so-well-quantified loading, such as thermal shock, hydraulic, and

seismic and post-seismic forces, and the forces related to handling and transportation. However, investigations conducted after the 1973 Rule-Making Hearing show that for oxidation temperatures  $\leq 1204^{\circ}\text{C}$ , the 17% oxidation limit (calculated with Baker-Just correlation) is adequate to ensure survival of unconstrained or fully constrained low burnup fuel rod claddings under quenching thermal shock. It was also shown that the 17% ECR limit (accurately determined based on measured phase layers thickness) is adequate to ensure retention of ductility and resistance to 0.3-J impact failure in non-irradiated, non-ruptured, two-side-oxidized Zircaloy cladding tubes.

Primary rationale of the **17% oxidation criterion** was retention of cladding ductility at temperatures higher than  $275^{\circ}\text{F}$  ( $135^{\circ}\text{C}$ ), i.e., the saturation temperature during reflood. Importantly, the 17% ECR threshold is tied with the use of the Baker-Just (B-J) correlation. If a best-estimated correlation other than Baker-Just (e.g., Cathcart-Pawel (C-P) correlation) had been used, the ECR threshold would have been  $<17\%$ . To complement this oxidation criterion, the  **$1204^{\circ}\text{C}$  peak cladding temperature (PCT)** limit was selected based on slow-ring-compression tests that were performed at  $25\text{-}150^{\circ}\text{C}$  and was confirmed by using other evaluation techniques. As such, the  $1204^{\circ}\text{C}$  PCT and the 17% ECR limits are inseparable, and constitute an integral criterion.

To account for the significant burnups extension and the chemical evolution of modern alloys during the last decades, regulators require to update the LOCA acceptance criteria, mainly by considering the effects on ECR of pre-transient hydrogen uptakes. In addition, US-NRC, because the inner side corrosion layer (less than  $\sim 10\mu\text{m}$ ) is a potential oxygen source, requires that double side oxidation, during the high temperature phase, must be considered, even away from the burst opening.

Among the large variety of tests used to assess or verify the LOCA limits, the following 2 types of tests are fundamental:

1. Post-Quench Ductility (PQD) tests, adopted by the US-NRC and other countries applying the 10CFR50.46 guidelines: after investigating different types of experimental options, the staff decided to use the Ring Compression Tests (RCT) of double-sided oxidized specimens to determine the Post Quench Ductility (PQD) as a function of Hydrogen content and time at high temperature (ECR).
2. Strength-based tests, adopted in Japan and in France, to determine LOCA acceptance criteria, including phenomena not covered by the RCT approach, such as the transient secondary hydriding in the necks regions, above and below ballooned / burst areas. The approach is based on semi-integral out-of-pile LOCA tests, on pre-hydrided (potentially thinned claddings to account for in-reactor pre-transient corrosion) and pressurized fuel rodlets. The limits are based on the tensile axial resistance upon quench of the rodlet.

### Experimental data base remaining gaps

The various thresholds used to define the LOCA limits or the consequences of FFRD are based on experimental tests results whose boundary conditions are overly conservative.

This is acceptable in a licensing approach where comfortable safety margins are available, but the nuclear industry is continuously searching for enhanced operating conditions (e.g., higher operating flexibility, higher burnups, higher enrichments, longer cycles, etc.) resulting in higher fuel duty. As such, it might be interesting to reconsider some of these experimental conservatisms, to refine the mechanistic approach and to find new margins.

Among the remaining gaps to be addressed through better test conditions, we can mention:

- Simulate the depressurization (blowdown) phase of a LOCA transient to investigate the effect on fuel pellet fragmentation, of transitioning from a fuel pellet parabolic thermal gradient to a homogeneous temperature distribution,
- Along this line, perform in-pile semi-integral LOCA tests starting with an initial linear heat rate equivalent to the in-reactor operational LHR, to investigate the effects of initial pellet-cladding closed gap and decay heat on the overall fuel rod behavior (e.g., effects on axial gas communication, on cladding deformation profile, on TFGR and on fuel fragments sizes),

- Include spacer grids in a LOCA fuel bundle test to evaluate the effect of the spacer grids (cold spots) on axial gas communication, cladding strain profile, transient fission gas releases (TFGR), fuel fragmentation and axial fuel relocation (FFRD),
- Perform LOCA tests on high burnup fuel bundle to study the steric effects on ballooning-burst and subsequent temperature excursion,
- Perform semi-integral LOCA tests on high burnup doped fuel to investigate the effect on TFGR and fragmentation,
- Design an in-pile LOCA test enabling measuring the actual occurrence time of the successive phenomena leading to FFRD and evaluate the related consequences,
- Update the LOCA database to include new types of fuel (ATF)

### Improved simulation tools and methods to address the new challenges

Identifying and investigating experimentally all physical phenomena playing a role during a LOCA transient, require developing simulation tools to analyse the tests and to make the link with the real in-reactor cases. The various LOCA-related phenomena are captured quantitatively by a range of predictive computer modelling codes. To predict fuel behavior during a LOCA, three types of codes are needed: reactor physics codes, fuel behavior codes (steady-state and transient), and thermal hydraulics (system) codes. Multi-rod transient codes are able to capture interactions in and amongst rods that influence possible flow blockage and channel coolability. These codes need to be well documented, verified and validated, and be subject to uncertainty qualification, and be kept at the state-of-the-art regarding predictive capability as understanding of the phenomena increases.

In parallel, advanced LOCA analysis methodologies, for both PWRs and BWRs, have been developed over the years, to reflect the shift from conservative approaches to more nuanced, probabilistically informed methods. This transition marks a substantial development in nuclear safety analysis, aiming to enhance the accuracy and reliability of the ECCS design assessment. Historically, methodologies such as the CQD<sup>1</sup> methodology dominated LOCA analysis. CQD, deeply conservative in nature, utilized safety margins that were deliberately large, intended to account for the widest possible range of uncertainties without detailed quantification. These methods typically over-predicted the severity of potential accidents, without proper quantification of the available margins, ensuring safety but often at the cost of operational efficiency.

The shift towards Best-Estimate Plus Uncertainty (BEPU) methodologies marked a pivotal change in LOCA analysis. Techniques such as Westinghouse's ASTRUM and later Full Spectrum LOCA (FS-LOCA), and AREVA's (now Framatome) RLB-LOCA, incorporate sophisticated statistical methods to manage and quantify uncertainties.

Recent publications in the nuclear engineering field continue to focus on refining these methodologies, often emphasizing the need for further integration of multi-physics modelling and enhanced computational capabilities. Trends in recent research include exploring more mechanistic models for fuel behavior, improving the integration between thermal-hydraulic and neutron-kinetic models, to analyse more accurately and effectively redefined LOCA worst scenarios together with, in many countries, updated LOCA acceptance criteria.

Observed trends and gaps in LOCA safety analysis are condensed below:

- **Adoption of BEPU Analysis:** The change for more penalizing LOCA acceptance criteria has led to the increasing adoption of BEPU analysis. This method emphasizes using best estimate calculation codes combined with statistical uncertainty analysis, marking a shift toward more sophisticated evaluation techniques. While BEPU analysis outcome, like 95/95 estimates for PCT (peak clad temperature), ECR (local maximum clad oxidation) and Core Wide Oxidation, are accepted as acceptance criteria verification by regulatory bodies, there are still discussions

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<sup>1</sup>CQD methodology means Code Qualification Document methodology according to the definition given in WCAP-12945, although CQD was described as Conservative Qualification Design (methodology) in some papers.

concerning their interpretation in the context of the determination of the expected safety margins or about their immediate usefulness for Reload Safety Evaluation (specifically in case of mixed cores).

- **Variation in Break Coverage:** Different countries are evolving their approaches to analyse the worst-case scenario for LOCA based on their level of occurrence:
  - In France and Sweden, there is a notable shift toward analysing worst intermediate breaks, while large break LOCA are only studied using best estimate assumptions.
  - Meanwhile, in countries like the USA or Belgium, a comprehensive analysis covering the full spectrum of potential breaks is required.
- **Consideration of FFRD:** Notably, fuel fragmentation models can be found in some LOCA codes, however relocation and dispersal models are largely overlooked in LOCA safety assessments in many countries.
- **Incorporation of Localized Phenomena:** There is an emerging trend in modern LOCA analysis methodologies to explicitly include localized phenomena that were previously considered within the inherent conservatism of the LOCA safety approach. For example:
  - In the USA detailed analysis of double-side oxidation effects away from the burst region is now required.
  - France is focusing on the potential impacts on PCT of fuel relocation within the ballooned regions in contact with adjacent ballooned rods.

To conclude, significant new data have been accumulated on the effects of irradiation, alloying elements, pre-oxidizing and pre-hydrating of the cladding, steam purity (addition of nitrogen or air) and high external steam pressure. Based on these findings, new licensing frameworks have been defined in most of the countries. Any new cladding material (e.g., ATF) has to be studied accordingly. It is important to continue developing fuel behaviour computer codes, associated with uncertainty analysis methods, to capture quantitatively the increased understanding of phenomenology. A specific effort should focus on multi-rod configurations to analyse core coolability.

# 1 Introduction

During a LOCA the core must be coolable, and structural integrity must be maintained to ensure long term coolability of the reactor core.

The fuel cladding temperatures during a LOCA may increase over 1 000°C, resulting in:

- Steam oxidation of the cladding
- Cladding ballooning and burst if the rod internal pressure is large enough. The ballooning may enable fuel fragmentation and relocation in the balloon area, such increasing the local power density leading in turn to higher peak cladding temperature (PCT), while the rod burst may lead to double side cladding oxidation, secondary hydrogen uptake, and embrittlement from the inside.
- Cladding embrittlement, specifically during the quenching phase through: 1) formation of an oxide layer and 2) dissolution of oxygen in the underlying Zr metal as well as 3) cladding hydrogen pickup during the steam oxidation of the fuel cladding
- Finally, if the high temperatures are maintained long enough, the Zr oxidation reaction may become so exothermic that fuel rod cannot be cooled.

Fuel safety criteria in most countries are based on the United States Nuclear Regulatory Commission (US-NRC) criteria which aims at limiting PCT (Peak Cladding Temperature), ECR (Equivalent Cladding Reacted) and H<sub>2</sub> production. Historically the LOCA limits (PCT and ECR) were defined based on a post-LOCA zero cladding ductility approach while the Japanese and the French criterion are based on the failure threshold value, determined in the integral thermal shock tests under restrained conditions.

Various LOCA testing methodologies have been used:

- Separate effect tests
- Semi-integral LOCA tests (in pile and out-of-pile)

To predict fuel behaviour during a LOCA, three types of codes are needed: reactor physics codes, thermomechanical fuel performance behaviour codes (steady-state and transient), and thermal hydraulics (system) codes. The trend is to develop calculation codes mixing thermomechanical behaviour and local thermal hydraulics.

The idea of this report is to tell the “LOCA story”, from physics to acceptance criteria, including a short background on the physical phenomena occurring during a LOCA focusing on the data and information which are used in the current design approaches used for BWRs, PWRs, and VVERs.

It is a challenge to summarize relevant LOCA information since the topic is very large and includes information from many different technical areas. Contrary to many other reports on LOCA, this report also contains information about LOCA performance of Accident Tolerant Fuel (ATF) and LOCA in Small Modular Reactors (SMRs). To make this report as readable as possible, the report information has been divided into three parts:

- Executive Summary for management that would like to get the “big picture” of LOCA and, the remaining issues to be resolved,
- Main body of the report providing technical information on various aspects of LOCA for engineers familiar with the LOCA topic and,
- Appendices on various LOCA topics with a greater deal of details.

The main body report structure is outlined below:

- Principles of reactor safety in Section 2,

- Description of prototypical LOCA transients (PWR, BWR, VVER, SMR and Spent Fuel Pool) in terms of timing, temperature, temperature rates, coolant pressure in Section 3,
- Description of the physical phenomena in Section 4 such as:
  - Oxidation and hydrogen pickup effects on fuel clad embrittlement
  - Ballooning, Burst, Co-planar deformation and Coolant blockage (ductile phase of the LOCA transient, Rupture (or brittle phase of LOCA transients) and
  - Rupture
  - Conclusions
- LOCA Safety acceptance criteria and their background in Section 5 as follows:
  - Historical LOCA acceptance criteria
  - Current US regulations and modifications being considered
  - LOCA acceptance criteria as they are considered in various countries
  - LOCA rulemaking in France, an example of a phenomenological approach
  - LOCA safety analysis acceptance criteria regarding FFRD issues
  - Technical basis for defining ATF LOCA acceptance criteria
  - Remaining gaps
- LOCA testing methodologies in Section 6
- Computer code and methods for LOCA in Section 7
- LOCA SMR relevant information in Section 8

The information about the Appendices is provided below:

- Reactor Characteristics in Appendix A -
- Fuel Assembly Designs in Appendix B -
- Materials used in fuel assemblies in Appendix C -
- Fuel Fragmentation, Relocation and Dispersal (FFRD) in Appendix D -. This is an excellent section to get an understanding of FFRD for those not too familiar with this topic.
- LOCA safety analysis acceptance criteria regarding FFRD issues in Appendix E -
- LOCA testing methodologies (Separate-effects and integral tests- in- and out-of-reactor) in Appendix F -. This is an expanded part of the information given in Section 6 in the main part of the report:
  - Separate Effect Tests
  - Out-of-pile semi-integral LOCA tests on single fuel rods, irradiated and defueled or prehydrated
  - Out-of-pile and in-pile and semi-integral LOCA tests on single real or simulated fuel rods
  - In-pile and out-of-pile semi-integral LOCA tests on fuel assemblies

- Remaining experimental gaps
- Structured frameworks for developing, validating, and quantifying uncertainties in BE LOCA analysis methods in Appendix G -.

## 2 Principles of reactor safety

### 2.1 Nuclear reactor design

The basic philosophy of the nuclear power plant design can be described as defence in depth, expressed in terms of three levels of safety.

- The first level of safety is to design the reactor and other system components so that they will operate with a high degree of reliability. This involves:
  - A requirement that there should be an inherent stability against a reactivity increase, e.g. through a quick-acting negative temperature (or power) reactivity coefficient.
  - The submittal of a proposed *quality assurance* (QA) program by the utility application for a permit to construct the nuclear power plant. The purpose is to provide assurance that the design of the plant is satisfactory and that construction and operation will be carried out in a manner that complies with the accepted codes and standards.
  - The construction of *redundant*<sup>2</sup> components and systems.
  - The construction of barriers in the nuclear reactor systems to limit radioactivity escape into the environment. For *BWRs* and *PWRs/VVERs*,
    - the first barrier to the escape of radioactivity from the fuel is the fuel cladding,
    - the second barrier is the primary coolant boundary and,
    - the third barrier is the containment structure.
    - The fourth barrier is a filter (in some countries only) outside the containment
- The objective of the second safety level is to provide the reactor with a protection system to safely accommodate a range of conceivable abnormal situations. The protection system includes various instruments for measuring operating variables and other characteristics of the overall nuclear plant system. If the instruments detect a transient that cannot be corrected immediately by the control system, the reactor is shut down automatically by the protection system. Also, the reactor operator can cause an independent (manual) trip if there are indications that an unsafe condition may be developing.
  - In a *PWR/VVERs*, when a reactor trip signal is received, the electromagnetic clutches holding up the control rods are de-energized by an automatic cutoff of the electric power. The rods will then drop into the reactor core. In addition boric acid solution (boric acid solution) can be injected from the chemical and volume control system, CVCS, to provide a backup to the control rods if required.
  - In a *BWR*, a rapid shutdown is obtained by forcing the control rods up into the core by hydrostatic pressure; at the same time, power to the recirculation pumps is cut off. In addition, the *BWR* reactivity can be decreased by injection of an aqueous solution of sodium pentaborate. Although the reactor shutdown cooling system is not generally regarded as a component of the protection system, shutdown cooling is nevertheless an essential aspect of reactor protection to ensure that the decay heat in the fuel will not overheat the fuel.

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<sup>2</sup> The term redundancy refers to the use of two or more similar systems in parallel, so that the failure of one will not affect the plant operation. They are of special importance in systems, such as instrumentation, shutdown controls, and emergency cooling, upon which safety depends.

- The third level of safety is the inclusion of engineered safety features which provide additional protection to the public during a postulated accident. The potential consequences of these accidents are analysed in a conservative manner to determine the adequacy of the engineered safety features to mitigate them. The major engineered safety features are:
  - the emergency core-cooling system, *ECCS*, to supply water to the reactor core in the event of a loss-of-coolant accident,
  - the containment vessel (or structure) to provide a barrier to the escape to the environment of radioactivity that might be released from the reactor core,
  - the cleanup system for removing part of the radioactivity and heat that may be present in the containment atmosphere, and
  - hydrogen control to prevent formation of an explosive hydrogen-oxygen mixture in the containment.

## 2.2 Safety criteria

*The overall objective of reactor safety is the prevention of radiation-related damage to the public from the operation of commercial nuclear reactors.*

To meet this objective safety criteria are introduced to avoid fuel failures during normal operation, or to mitigate the consequences from reactor accidents in which substantial damage is done to the reactor core. The current safety criteria were developed during the late 60s and early 70s based upon a number of experiments on Zry-2 and Zry-4 fuel claddings of essentially nonirradiated fuel and some limited experiments with fuel with low and intermediate burnups. The reason being that it was thought at that time that fresh fuel had the smallest margins towards the safety criteria during these accidents. This information was used to develop the fuel safety criteria for these accidents as well as the related analytical methods (computer codes).

The main idea in the development of the current safety criteria was that the consequences of postulated accidents are inversely proportional to their probability, Table 2-1. To ensure that these safety criteria are met, certain *fuel design criteria* (mechanical, nuclear and thermal-hydraulic criteria) must be met, see Figure 2-1.

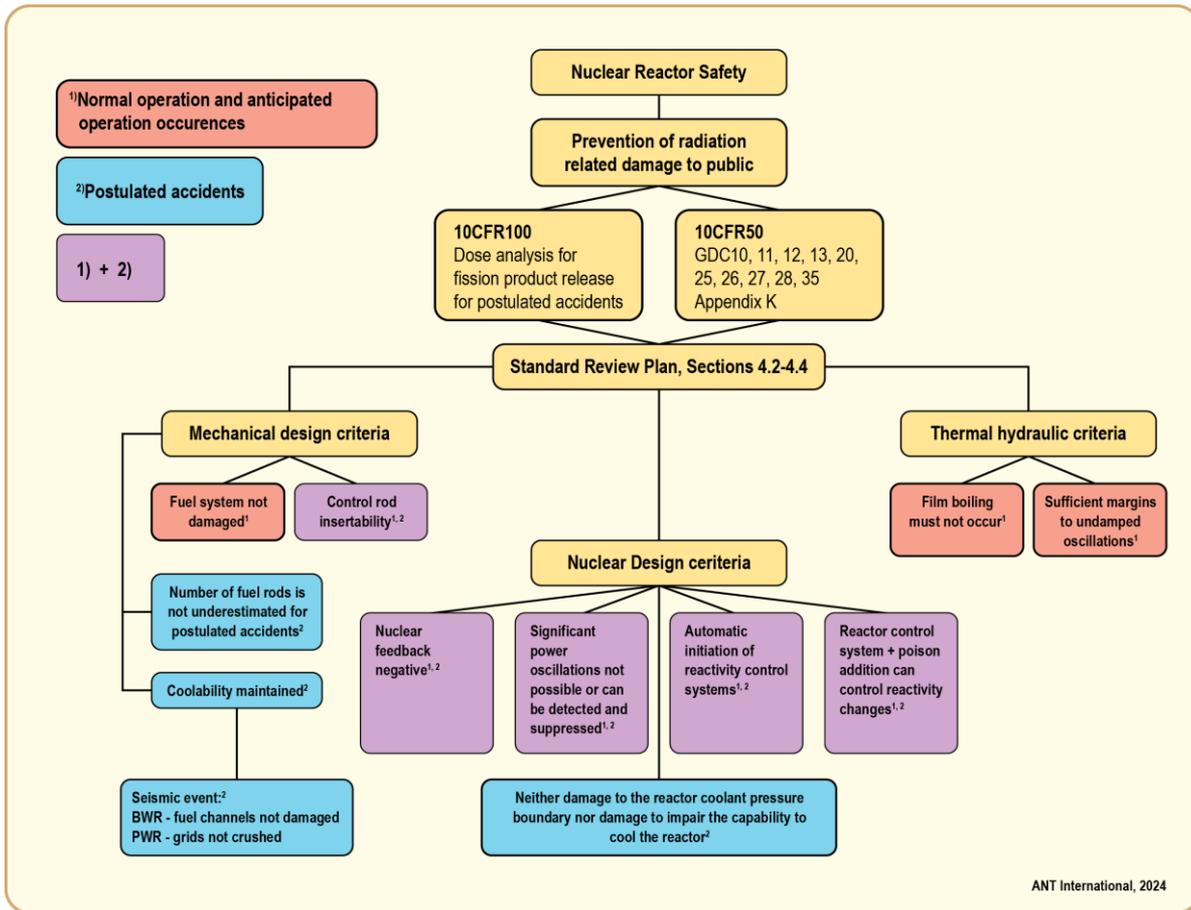


Figure 2-1: Outline on relation between 10CFR50, 10 CFR100 and SRP section 4.2-4.4.

The consequences of operational events and postulated accidents are inversely proportional to their probability, as shown in Table 2-1.

Events with a probability varying from  $\sim 1$  to  $10^{-2}$ /yr are characterized as AOOs (Anticipated Operational Occurrences) and are Class II events. For these more probable transients, safety criteria do not allow fuel failures and only a very small number of fuel rods in the core are allowed to experience boiling crisis. More specifically, the DNBR (Departure from Nucleate Boiling Ratio) for PWRs and the CPR (Critical Power Ratio) for BWRs shall be determined so that with a certain probability the CHF (Critical Heat Flux) is not exceeded.

Examples of Class II events that may result in an increase in thermal power are:

- Coolant temperature decrease.
- Control material removal.
- System pressure increase.
- Decrease in cooling effectiveness.

All other events with a probability less than  $10^{-2}$ /yr are characterised as (postulated) accidents. For the less probable accidents, fuel failures are allowed but the criteria are usually established to ensure core coolability during the event itself and for the long term. Postulated accidents may be divided into two parts, namely:

- Class III events have low probability and the potential for small radioactive release outside the plant site. In these postulated events the core would remain covered with water. Reactors are supposed to be able to restart after the event, provided damages are fixed and safety authorities agree. Examples of these events are:
  - Small pipe break.
  - Loss-of-flow accident.
- Class IV events have very low probability and are potentially more severe; these are called DBAs (Designed Basis Accidents). Some of the plant systems are potentially damaged, and the power plant is unlikely to restart. The most severe DBA is considered to be a complete (double-ended) rupture of a large pipe, ranging in diameter from 0.61 to 1.07 m (about 2 to 3.3 feet), in the primary coolant circuit of a PWR or in the recirculation pump intake line of a BWR. These DBAs are denoted Large Break Loss-of-Coolant Accidents (LBLOCAs). Other DBAs are:
  - Earthquakes, Tornadoes, and Flooding.
  - Control Element Ejection, (Reactivity Initiated Accident or RIA).
  - Spent-Fuel Handling Accident.

Beyond DBAs, the events are considered as more severe in the sense they can lead to core melting. The goal is then to ensure that the existing safety systems are adequate to limit the doses consequences at the boundary of the power plant.

Severe accidents (i.e., beyond the design basis accidents) are classified in DEC-A (Design Extension Conditions) for accidents leading to limited damage to the reactor core, and DEC-B for accidents leading to partial core melt.

Table 2-1: Probability of different reactor events.

Class	Event types	Acceptance criterion	Probability per year	Examples
I	Normal operation	No fuel failures	1.	Full power operation, refuelling
II	Anticipated transients		1-10 <sup>-2</sup>	Loss of feed-water, pump trip, turbine trip
III	Anticipated transients with additional equipment failures	Fuel failures OK, but fuel should retain coolable geometry	<10 <sup>-2</sup>	Break outside containment, small primary break, turbine trip without bypass with scram on second signal
IV	DBAs		<10 <sup>-4</sup>	Large break LOCA, control rod ejection (PWR) or control rod drop (BWR)
	DEC-A and DEC-B		<10 <sup>-6</sup>	Station blackout, LOCA with large leak from drywell to wetwell

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The objectives of the *fuel system*<sup>3</sup> safety review are to provide assurance that:

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<sup>3</sup> Fuel system consists of assemblies of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods;

- (1) The *fuel system* is *not damaged*<sup>4</sup> as a result of normal operation and AOOs.
- (2) *Fuel system* damage is never so severe as to prevent control rod insertion when it is required.
- (3) The number of *fuel rod failures*<sup>5</sup> is not underestimated for postulated accidents.
- (4) *Coolability*<sup>6</sup> is always maintained during and after the accidental transient.

Objective (1) in the above list is formalized in General Design Criterion 10, GDC10 [10 CFR Part 50 Appendix A, 1990]. The application of *GDC10* is described in the Standard Review Plan, SRP [USNRC, 1981]. The *fuel system*<sup>7</sup>, nuclear, and thermal and hydraulic designs are covered in SRP sections 4.2, 4.3 and 4.4, respectively. Section 4.2 in SRP identifies a number of *fuel system* failure mechanisms that actually have occurred in commercial reactors, as well as hypothesized *fuel system* failure mechanisms. For each of these *fuel system* failure mechanisms, SRP section 4.2 lists a corresponding design limit intended to accomplish objective (1) in the list above. These design limits are called SAFDLs. Thus, the SRP does not include any design limits to address potential new *fuel system* failure mechanisms related to more recent fuel designs and/or reactor operation strategies.

Fuel rod failures must be accounted for in the dose analysis required by [10 CFR Part 100, 1995], for postulated accidents.

The general requirements to maintain control rod insertability and core coolability appear in the General Design Criteria, e.g., GDC 27 and GDC 35. Specific coolability requirements for the loss of coolant accidents, LOCA, are provided in [10 CFR Part 50].

The *fuel system* design bases must take into account the four safety objectives described on the previous page. The Specified Acceptable Fuel Design Limits (SAFDLs) which are discussed below, do this. In a few cases the SAFDLs provide the design limit but in most cases, it is up to the fuel designer to recommend a design limit value, taking into account a specific failure mechanism. The fuel designer must also provide the background data used for determining the design limits (those specified by the NRC as well as those proposed by the specific fuel designer) to ensure that the design limits are both necessary and sufficient. To get the license for a new type of fuel, the fuel vendor must provide all the data relative to this specific fuel design to demonstrate that the design limits are still applicable and sufficient.

Specific failure mechanisms for the *fuel system* (including *the fuel rod*) and licensing criteria related to classes I and II operation and classes III and IV events are discussed in the following subsections.

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holddown spring, connections, spacer grids and springs; end plates; channel boxes; and reactivity control elements that extend from the coupling interface of the control rod drive mechanism in the core.

<sup>4</sup> Not damaged means not only that the fuel integrity is maintained, i.e., no release of radioactivity, but also that the fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements GDC10 and the design limits that accomplish this are called SAFDLs.

<sup>5</sup> Fuel rod failure means that the fuel cladding has been breached and radioactivity from the fuel get access to the coolant.

<sup>6</sup> Coolability means that the Fuel Assembly (FA) retains its rod-bundle geometry with adequate hydraulic section enabling removal of residual heat, even after a severe accident.

<sup>7</sup> Fuel system consists of assemblies of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; holddown spring, connections, spacer grids and springs; end plates; channel boxes; and reactivity control elements that extend from the coupling interface of the control rod drive mechanism in the core.

## 2.3 Fuel operating margins

To ensure that fuel does not fail during normal operation or anticipated operational occurrences, and to ensure that coolability is maintained during postulated accidents, fuel design criteria (such as maximum rod internal pressure, peak fuel temperature, critical heat flux ratios, peak cladding temperature and max oxidation during *LOCA*, etc.) are specified or agreed by the Regulators. In most countries the Regulators are applying the same criteria as USNRC.

The fuel vendors are using fuel performance codes to assess the values reached by a series of key parameters (e.g., fuel temperature, rod internal pressure, cladding temperature, oxidation level, etc.) to verify that the fuel design limits are not reached (Figure 2-2). For each fuel design criteria based on fuel temperature, there will be a thermal limit varying with burnup (because thermal conductivity degrades with burnup) and the most limiting thermal limit will establish the operating regime for a given fuel design, see Figure 2-3.

Also, cycle specific analysis are done either by the fuel vendor or the utility to ensure that the core loading is appropriate and that thermal limits will not be exceeded. Finally, the utility must supervise the core with the core monitoring systems to ensure that the design thermal limits are not exceeded.

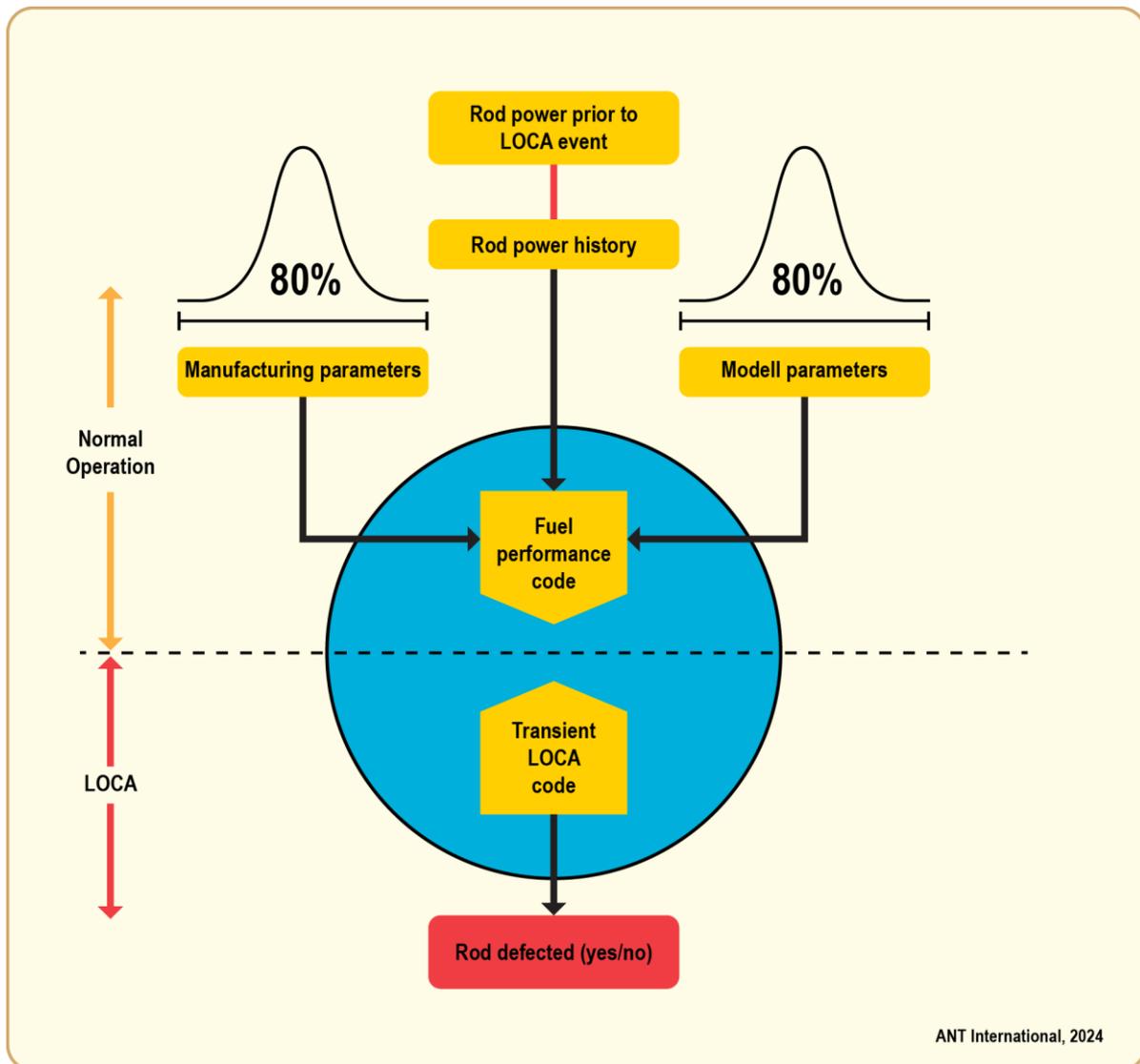


Figure 2-2: Schematics showing the process of establishment of thermal limits, modified figure according to [Heins, 2004].

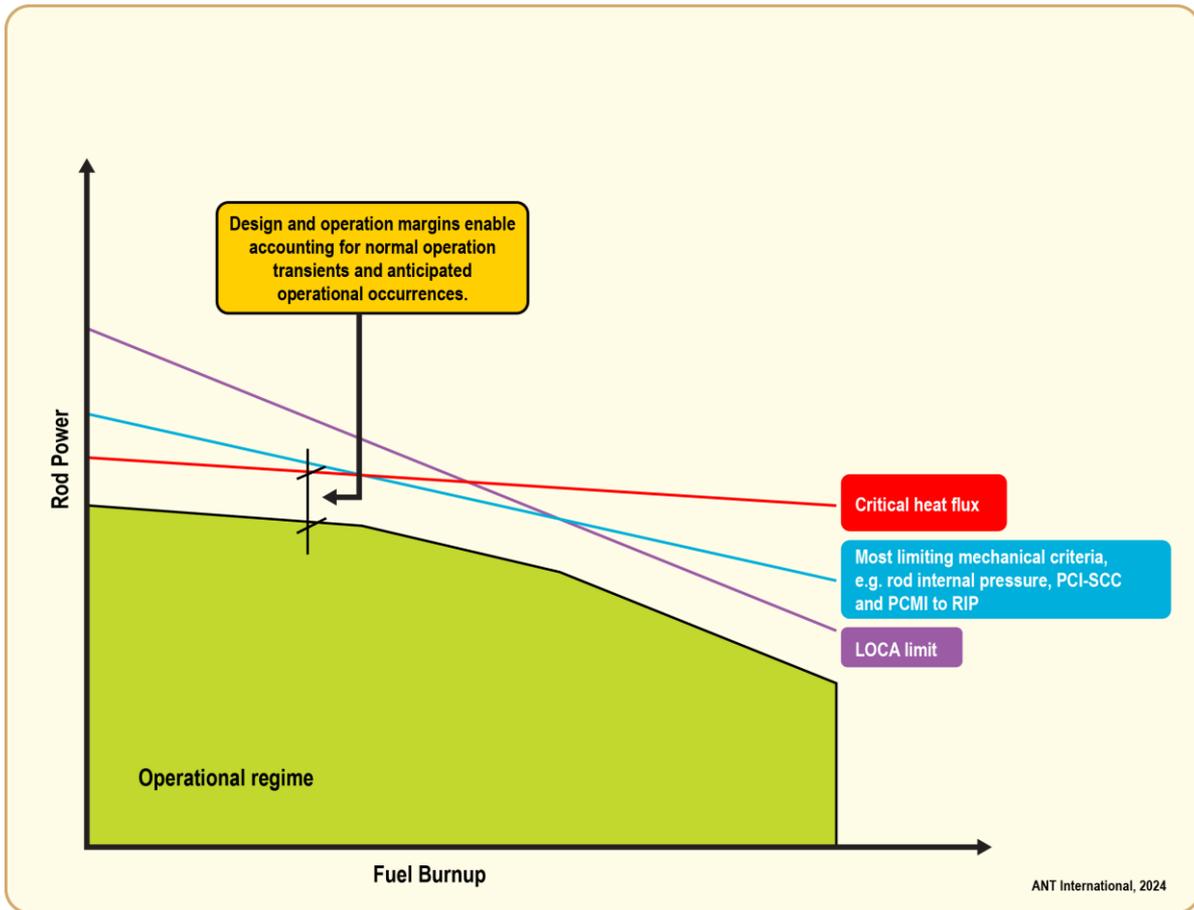


Figure 2-3: Schematic showing the operational regime. The red line represent the maximum rod power allowable to ensure that boiling crises will not occur during stationary condition or during a class II transient. The blue line represents the maximum rod power allowable to ensure that fuel failure will not occur during class I and II operation, which in most reactors is related to the maximum allowable rod power leading to pellet cladding interaction and cladding stress corrosion cracking (PCI-SCC). The purple line represents the maximum rod power allowable to ensure that fuel will retain coolability during a Large Break LOCA event.

## 2.4 Difference between LOCA and DEC-A accidents

During and after a LOCA it must be ensured that:

- The core remains coolable during and after the event (which means that the maximum allowable coolant blockage is limited) and,
- Fuel fragments dispersal in the coolant (FFRD), if any, should remain limited to ensure no local flow blockage nor risk of re-criticality. One way to demonstrate FFRD has no consequences is to avoid cladding burst of the high burnup fuel rods (i.e., it means that PCT of high burnup fuel rods should stay below the typical burst temperature, around 750°C).

*During a design basis LOCA, only steam is assumed to be in contact with the core, while during a DEC-A or DEC-B transients also nitrogen (from air) may be in contact with the core.*

### 3 Description of prototypical LOCA transients in terms of timing, temperature, temperature rates, coolant pressure

#### 3.1 Introduction

Loss-of-coolant accidents (LOCA) can be differentiated between several categories depending on the size of the postulated break in the primary coolant system. For the design basis accidents (DBAs) a Large Break Loss of Coolant Accident, *LBLOCA*, - a “guillotine” (or double-ended) break is postulated on one of the cold legs of a *PWR*, Figure 3-1, or in one of the recirculation pump intake lines of a *BWR*,

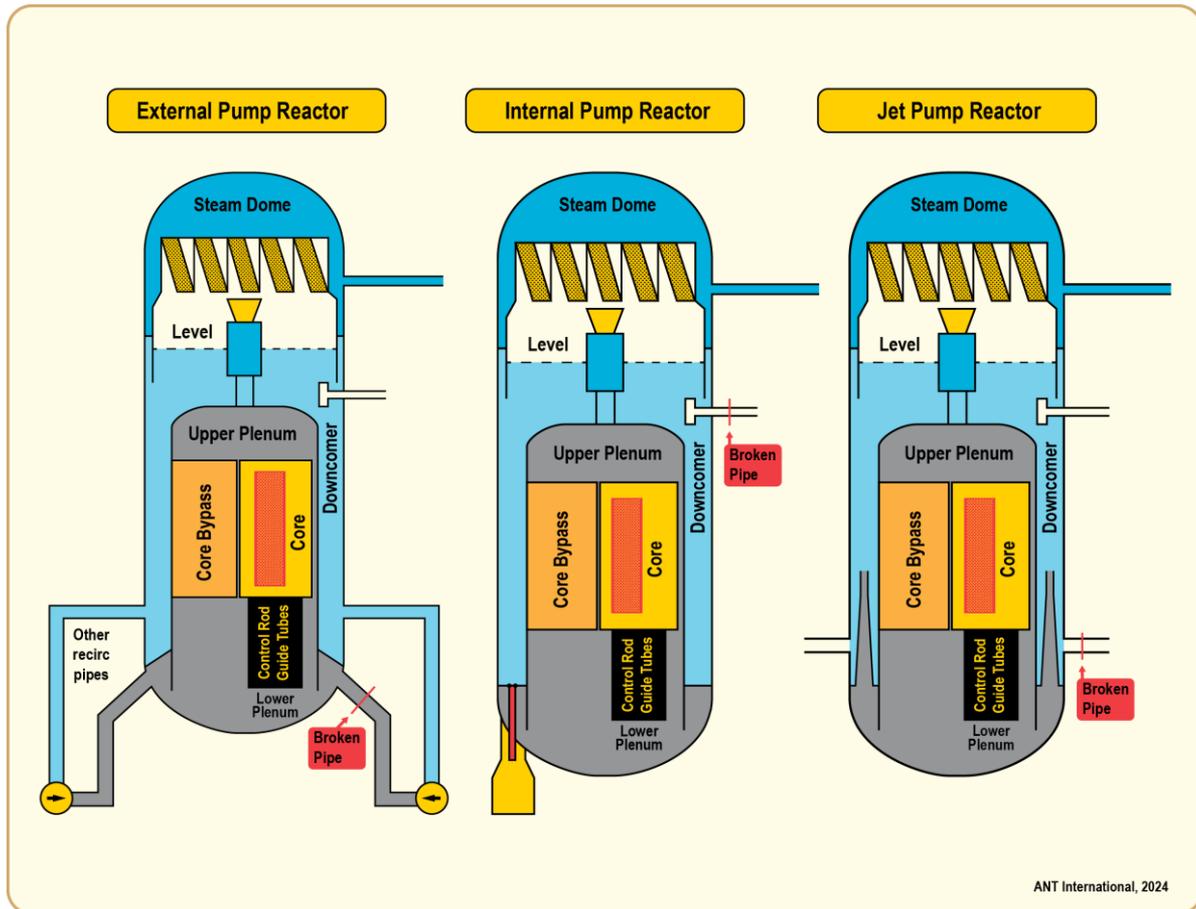


Figure 3-1: Basic BWR Geometries. Nine Mile Point 1 in upstate New York (the only External pump reactor existing today). The red box in the figure designates the location of the postulated pipe break, modified figure according to [ALARA Engineering, 2002].

Figure 3-2, an intermediate (IBLOCA) or small break *LOCA* (SBLOCA) cover the spectrum of events where the break in the primary circuit is less than a major one and does not necessarily lead to rapid blow-down and complete uncovering of the core. In some countries (e.g., France in 2017) the regulators agreed to analyse *LBLOCA* cases considering best estimate hypothesis to assess the doses consequences and to design the filtering systems. As a result, DBA *LOCA* analysis (considering penalized approaches) is limited to the bounding cases within the whole spectrum of IBLOCA or SBLOCA. This choice is funded on the fact that a double-ended break is low occurrence and is prevented by the anti-whipping structures included in the Reactor Coolant System (RCS) design. Along this change of paradigm, it has been requested to use the same bounding break size to perform the thermo-mechanical design of the reactor structures (usually the 2 approaches are decoupled and not always based on the same loading assumptions).

Along the same line, the US-NRC recognized in years 2000s that double-ended guillotine LBLOCA is highly unlikely and current regulatory treatment is inconsistent with risk insights. US-NRC developed then the Transition Break Size (TBS) approach, based on the frequencies of the different types of breaches. All LOCAs below the TBS should meet the current 50.46 analysis/acceptance criteria while LOCAs above the TBS are analysed with relaxed standards. To our knowledge the TBS rulemaking process didn't go to the end but discussions are starting again.

During a *LBLOCA*, the primary system pressure drops, and most of the reactor water is expelled into the containment. In this very initial phase of a LOCA fast heatup of the cladding takes place due to temperature equilibration from the hot fuel pellets to the cladding. The drop in pressure would activate the protection system and the reactor would be tripped. The fission chain reaction in the core would thus be terminated, however, decay heat would continue to be released at a high rate from the fuel. The various Emergency Core Cooling Systems, *ECCS* subsystems must then provide sufficient cooling to minimize overheating and fuel cladding damage. The steam flow limiters and isolation valves, inside and outside the containment vessel, would close automatically to prevent the spread of possibly contaminated steam.

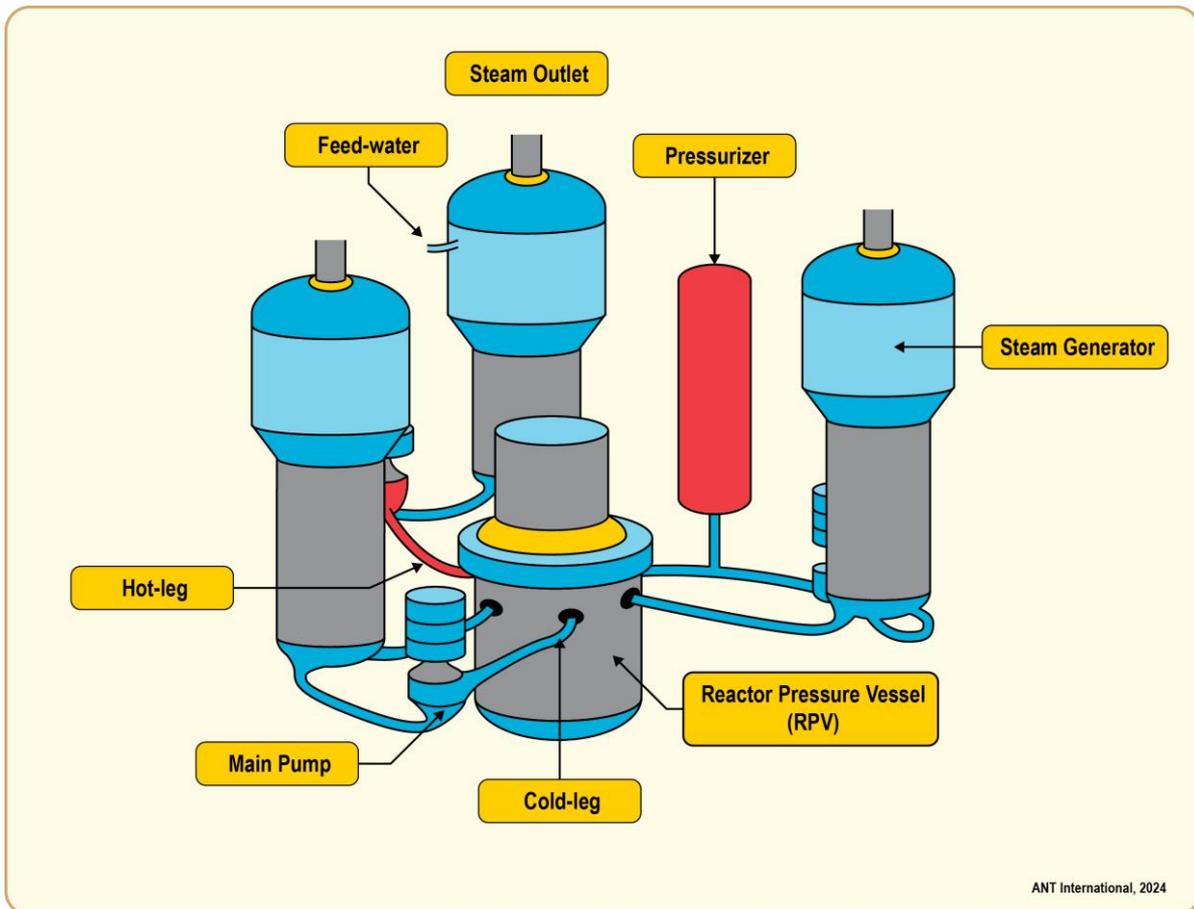


Figure 3-2: PWR geometry, modified figure according to [ALARA Engineering, 2002].

A LOCA can also occur in the Spent Fuel Pool, SFP. Adequate cooling of the spent fuel in the SFP can principally be lost either by malfunction of the pool cooling system (loss of cooling accidents) or by loss of the pool water inventory (loss of coolant accidents, LOCAs). There are many similarities between the phenomena involved in reactor LOCAs and SFP loss of cooling/coolant accidents, the latter tend to be slower, due to the relatively low power of the fuel and the large water volume in the pool. Another important difference is that there are three physical barriers between the nuclear fuel and the environment when the fuel is in the reactor (the fuel cladding, the envelope of the primary circuit and the containment building), whereas there is usually only one physical barrier (the fuel cladding) when the fuel is stored in SFPs.

## 3.2 LOCA in PWRs, VVERs and BWRs

### 3.2.1 BWRs

For a design basis LOCA in a jet pump BWR, Figure 3-1 (see figure to the far right) and Figure 3-3, the following phases can be identified:

- *Nucleate boiling* - The recirculation pump in the intact line would continue to function while coasting down. Some water would continue to flow through the reactor core and nucleate boiling could continue. Subsequently, the rate of water flow through the core would drop off and the temperature would rise. During this time, water would be flowing out of the break, and the level in the reactor vessel would fall.
- *Blowdown* - When the level of water outside the shroud reaches that of the break, assumed to be at a recirculation pump intake line below the core bottom, blowdown would occur. There would then be an escape of steam from the reactor vessel and the pressure would decrease fairly rapidly. As a result, the water in the lower plenum, would undergo violent boiling (flashing). Part of the mixture of steam and water would pass up through the core and reduce the fuel cladding temperature somewhat. This would be followed by a short core heatup period.
- *Core spray* - By the time blowdown and lower-plenum flashing are almost over, signals of the low water level in the reactor vessel and of the increase in drywell pressure would have activated the *ECCS* system.
- *Reflooding* of the core would occur by accumulation of water in the lower plenum of the reactor vessel. When the water level has reached the bottom of the core a steam-water mixture flowing up through the core will cool the fuel rods resulting in large thermal clad stresses.

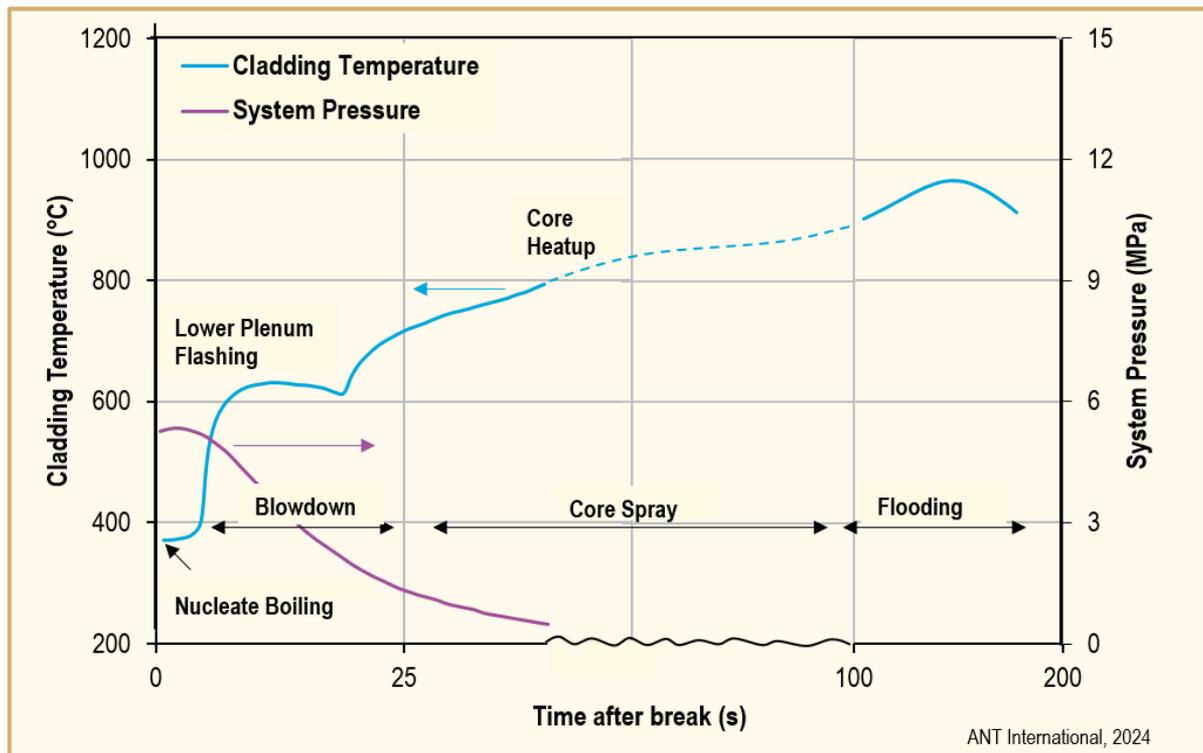


Figure 3-3: BWR LBLOCA in a jet pump reactor.

Internal pump reactors, Figure 3-1 (see middle figure) have similar clad temperature evolution as that in the jet pump reactor, Figure 3-3.

However, the LOCA oxidation event is much more severe in an external pump reactor, Figure 3-1 (see figure to the far left) and Figure 3-4, compared to both a jet and internal pump reactor. Today, only one external pump BWR reactor exists, Nine Mile Point in upstate New York, USA. Due to this reactor design, the ECCS will not be able to fill the core with water, however, rewetting of the hot fuel clad surface will occur at the point when the Leidenfrost temperature has been reached resulting in a quench of the clad temperature. Comparing Figure 3-4 with Figure 3-3, it is obvious that the time at high clad temperature is about one order of magnitude larger for an external pump reactors compared to that of e.g. a jet pump reactor. The much longer time at high clad temperature in the external pump reactor will often result in limiting LOCA thermal limits for this reactor design.

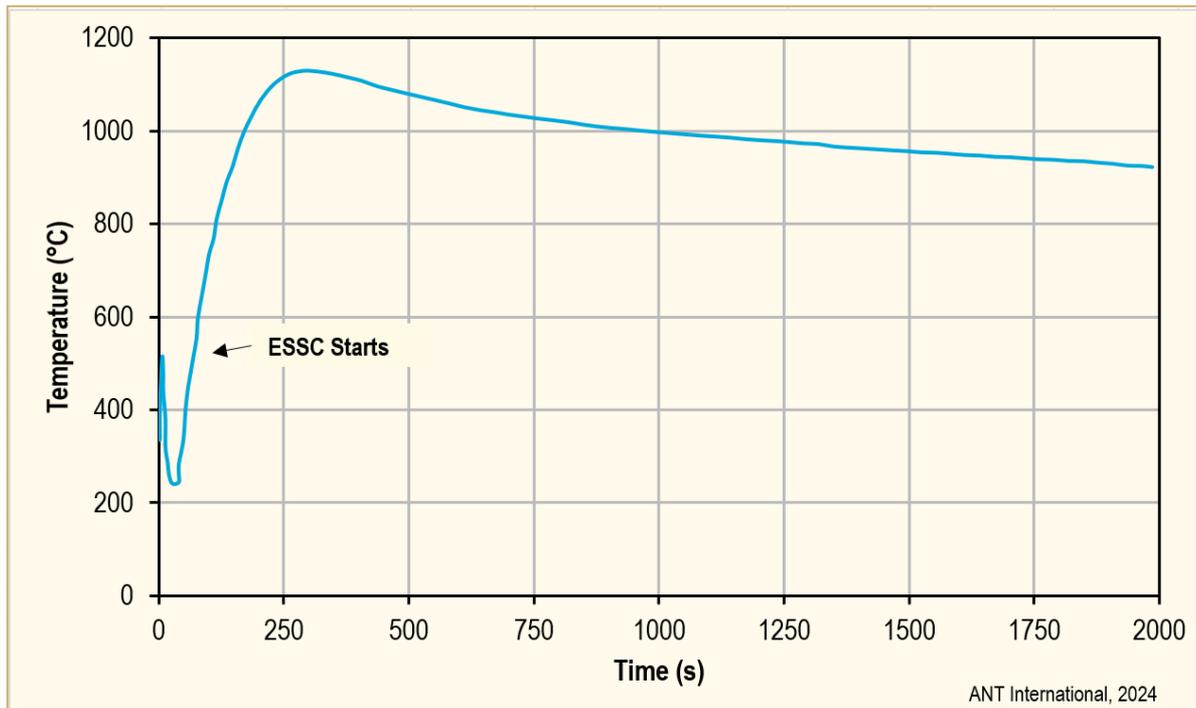


Figure 3-4: Fuel clad surface temperature evolution during a LOCA in an external pump BWR, modified figure according to [ALARA Engineering, 2002].

### 3.2.2 PWRs

The design basis *PWR LOCA* may be separated into roughly three phases:

- *blowdown*, in which coolant is expelled from the reactor vessel, Figure 3-5.
  - During the initial blowdown stage, the primary system pressure drops more or less rapidly, depending on the break size Figure 3-6. Propagation of pressure waves may occur during this phase imposing mechanical loads that could damage the reactor core systems.
  - In the subsequent, saturated blowdown stage, steam voids are formed and the steam-water mixture flows out through the break until the system pressure becomes about equal to the containment pressure.

The emergency shutdown of the reactor core at the very beginning of the transient immediately cancel power generation in the fuel, however, its residual heat and degraded cooling conditions will increase the temperature of the fuel and the cladding. During this process, the temperature of the periphery of the fuel pellets will increase while the centreline temperature will decrease. This inversion of the initial radial temperature profile within the fuel pellets are expected to generate thermomechanical stresses susceptible to lead to fuel fragmentation and fission gas releases.

- After a few seconds the fuel rods are cooled to some extent by the steam-water flow, and the cladding temperature may drop for a short time before heating up again. The magnitude of this first temperature peak depends on the LOCA scenario.

Subsequently, as the enthalpy of the fluid increases, the critical heat flux falls below the maximum flux resulting in a decrease in the heat-transfer coefficient and a corresponding increase in the cladding temperature. The resulting fuel clad temperature increases, and the rod internal overpressure may result in burst failure of some of the hotter rods. The ballooning and burst phenomena depends on the temperature field surrounding the fuel rod which is far from being uniform within a fuel bundle: The higher the azimuthal thermal gradient around the fuel rod, the smaller the balloon sizes.

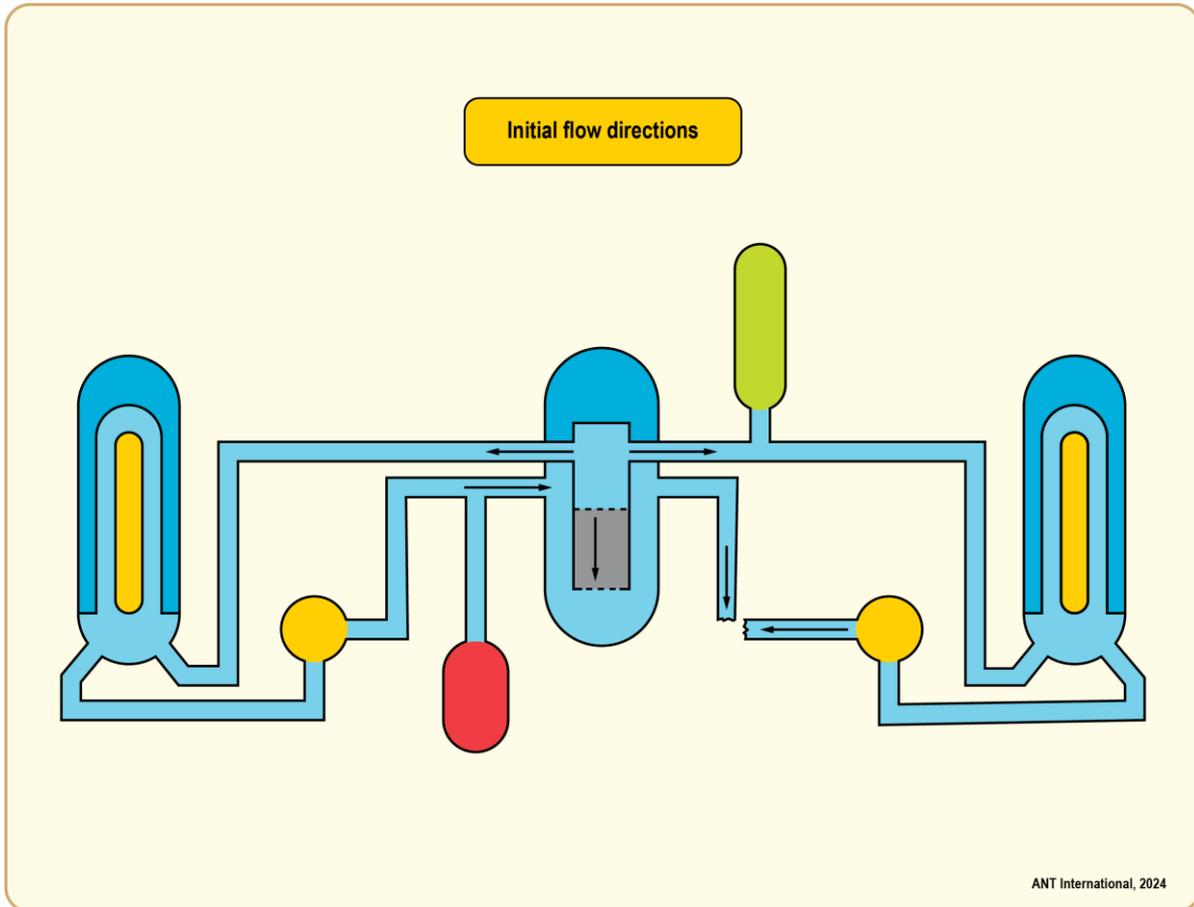


Figure 3-5: Cold-leg break in *PWR* – blowdown phase, modified figure according to [ALARA Engineering, 2002]

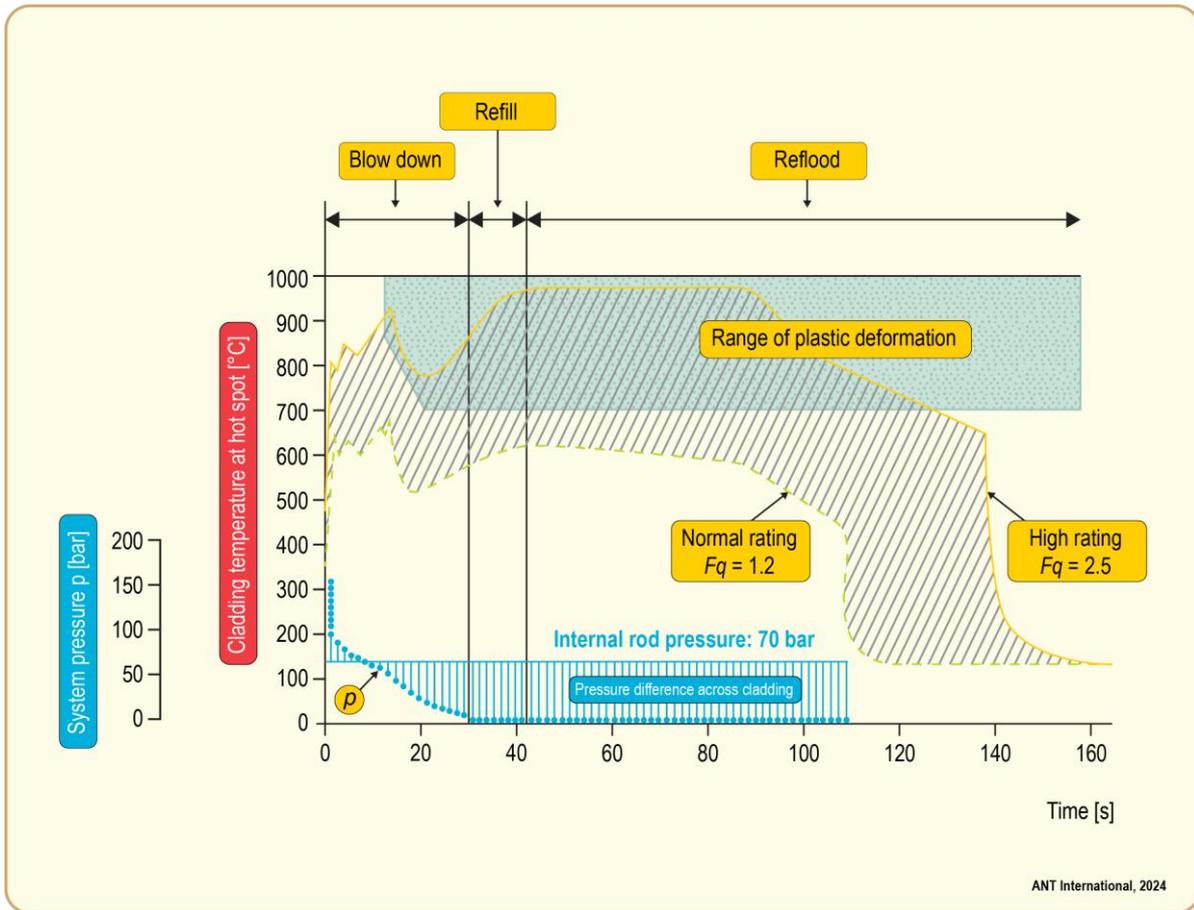


Figure 3-6: PWR LBLOCA. A pressure of 70 bar corresponds to 995 psi, a temperature of 1000 °C corresponds to 1832 °F modified figure according to [Erbacher & Leistikow, 1987].

- *refill*, when the *ECCS* begins to fill the vessel with water, up to the bottom of the core,
  - The decrease in the *PWR* primary system pressure during blowdown activates the *ECCS*. Also borated water is injected into the reactor vessel providing some cooling of the fuel. There is very little cooling of the fuel during the refill phase and such cooling mainly occurs by steam-water mixture convection.

*reflood*, when the water level rises sufficiently to cool the core, Figure 3-7.

- Reflooding of the core commences when the water level reaches the bottom of the fuel rods resulting in a dramatic decrease in fuel clad temperature imposing significant thermal clad stresses.

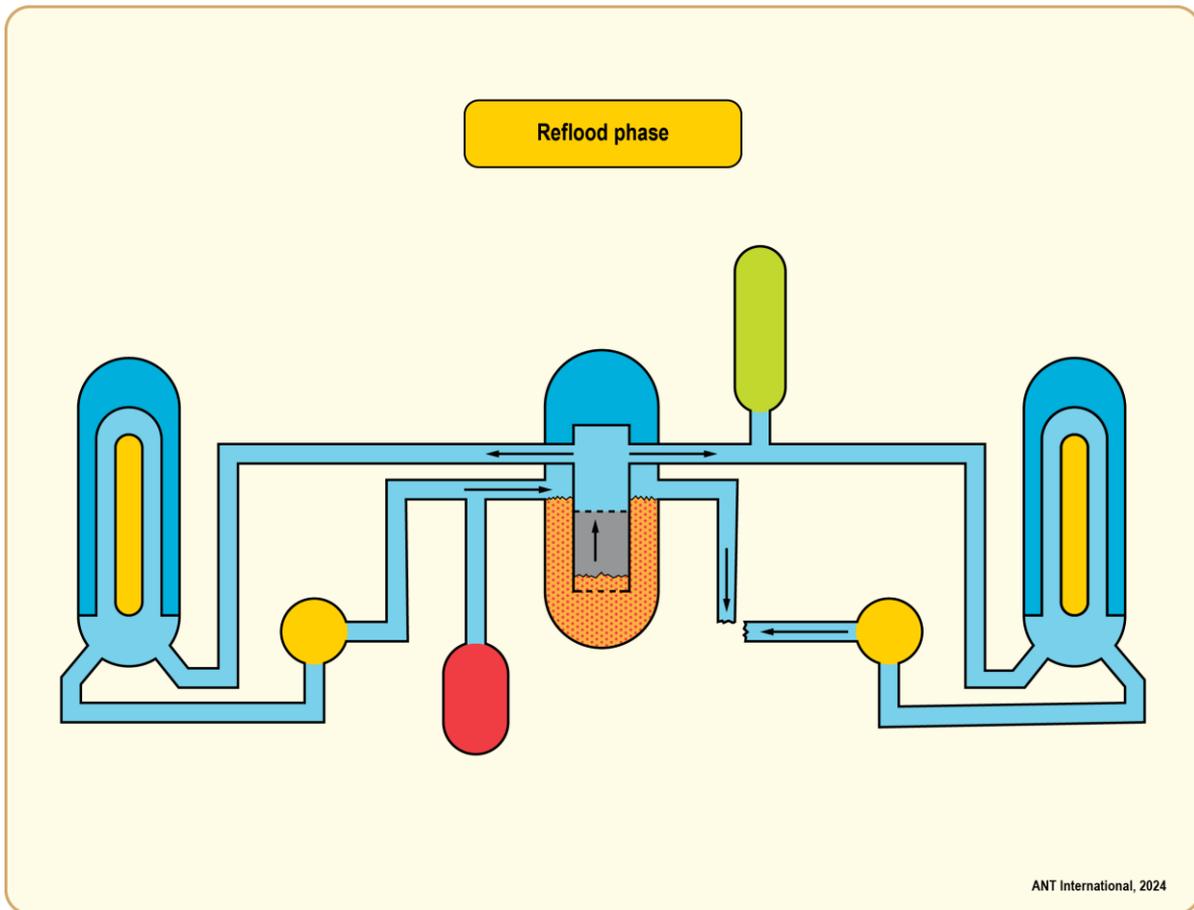


Figure 3-7 Cold-leg break in PWR, reflood phase, modified figure according to [ALARA Engineering, 2002].

The fuel and fuel clad temperature is governed by the fuel rod decay heat and the ECCS cooling conditions. Thus, the delayed energy release (decay heat) from the decay of fission products and actinides in irradiated fuel is an important factor in the analysis of postulated loss-of-coolant accidents (LOCAs) and evaluation of emergency core-cooling system (ECCS) performance.

After the fission process ends, the accumulated radionuclides continue to release recoverable energy in the form of beta and gamma rays. This delayed energy represents about 7% of the total recoverable energy from fission [Gauld, 2006]. Approximately 25% of this residual energy is released in the first 10 s after fission, and about 50% is released by 100 s after fission.

Accident analyses typically consider the decay heat rate for times shorter than about 10 s after fission. Calculation of decay heat in this time frame is widely performed within the nuclear industry using decay heat standards, such as ANSI/ANS<sup>8</sup>-5.1 [ANS, 2005].

Figure 3-6 shows the effect of decay heat, where one can see that larger fuel rod power prior to the LOCA event results in an increase in stored decay heat, thus resulting in increased fuel and cladding temperatures during the LOCA event.

<sup>8</sup> American National Standards Institute/American Nuclear Society

### 3.2.3 VVERs

The development of VVER reactor types began in the Soviet Union in the 1960s and still continues. The electrical power output of these reactors ranges from 70-210 MWe for the first generation to 1300 MWe for the latest versions of VVERs currently offered by vendors. The most common reactor types in operation are the VVER-440 and VVER-1000. However, there have been numerous design modifications worldwide for specific sites. For example, the VVER-440 V-213 and VVER-440 V-230 are both VVER-440 types, however, their safety systems differ significantly, resulting in varied responses during postulated accidents such as LOCA.

This chapter focuses mainly on the performance of the VVER-1000 V-320 (VVER-1000/320) during a Loss of Coolant Accident (LOCA). It is important to note that the information presented may not be fully applicable to all VVER designs due to the wide range of reactor types, power uprates, design modifications, and safety system upgrades.

The VVER-1000/320 reactors are among the most common Russian-designed Pressurized Water Reactors (PWRs) worldwide, with 25 units currently operating in Russia, Ukraine, Bulgaria, and the Czech Republic. Additionally, the newer Gen III and Gen III+ VVER reactors are based on this design. The VVER-1000/320 is a 4-loop reactor with a design thermal power of 3000 MWth and an electric output of 1000 MW. These reactors share some similarities and differences in their safety injection systems compared to other PWRs as summarized in Table 3-1.

Table 3-1: ECCS comparison for various PWR designs in comparison with VVER-1000 and VVER-1200 designs

		Westinghouse (3-loop)	Siemens (Konvoi)	Framatome (P4)	VVER- 1000	VVER- 1200	EPR	APR1400	Hualong
					V-320	V392m			
Thermal Power (MWth)		3000	3600	3800	3000	3600	5000	4000	3600
HPIS	Number of trains	3	4	2	3	2	4	4	2 or 3
	Shutoff head (MPa)	18	11	12.0–10.2	10.6	8.6	9.8	11–14.5	~9
	Capacity per pump (kg/s)	50	62	68–136	78	83	40	50	~30
Accumulators	Number of accumulators	3	8	4	4	4 (HA-1) 8 (HA-2)	4	4	2 or 3
	Total capacity (m <sup>3</sup> )	41 x 3 (36 + 5)	272	188	60 x 4 (50 +10)	60 x 4 (HA-1) 120 x 8 (HA-2)	600 (400 + 200)	68 x 4	50 (35–38)
	Discharge pressure (MPa)	4.4	2.5	4.4	6	5.9 (HA-1) 1.5 (HA-2)	5.5	4	5.5
	LPIS	Number of trains	3	4	2	3	2	4	-
	Shutoff head (MPa)	~ 1.5	0.9	2.3–2.0	2.5	~ 2.5	2.35	-	N/A
	Capacity per pump (kg/s)	406.8	306	120–280	207.2	250	138.8	-	~130

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The VVER-1000/320 is a four-loop PWR with safety systems (ECCS and containment spray system) similar to those in Western PWRs but with unique characteristics. For instance, the Hot Legs (HL) and

## 4 Description of the physical phenomena occurring during a LOCA transient

### 4.1 Introduction

The Loss-of-Coolant-Accident (LOCA) event starts by the decrease and then the loss of coolant flow due to a break in a coolant pipe while, at the same time, the reactor is depressurized, scrammed and shut-down. The fuel starts heating up due to its decay heat until the Emergency Core Cooling Systems (ECCSs) are activated and cools the fuel. Hypothetical LOCA events are analysed for each reactor to assure that the safety criteria, defined by the regulators, for the reactor system and the fuel, are met.

The design basis LOCA is a break in a pipe that provides cooling water to the reactor vessel. Analyses are performed for a variety of break sizes and locations to demonstrate that the ECCS can maintain the fuel in a coolable geometry as described in Section 3. The limiting break is typically in one of the cold, main coolant pipes of a PWR or one of the intake pipes to the recirculation pump of a BWR.

The LOCA process starts by the decrease and ultimate loss of coolant flow while the reactor is depressurized, Figure 4-1. The loss of coolant flow decreases heat removal from the fuel, increasing the fuel temperature and causing a significant temperature rise of the cladding. The decrease in system pressure causes an outward pressure differential and a hoop stress in the cladding wall. The result is the plastic deformation, or ballooning of the cladding. Ballooning may also result in fuel fragmentation<sup>22</sup> and subsequent axial fragment relocation in the ballooned area<sup>23</sup> that may impact the cladding temperature as well as the Equivalent Cladding Reacted (ECR<sup>24</sup>) in the later phase of LOCA.

Depending on the temperature, the cladding ductility and the rod internal pressure, the cladding will either stay intact or may burst which will allow steam to oxidize the fuel clad inner surface. In addition, some of the hydrogen released by the water/zirconium corrosion reaction inside the burst fuel may be picked up by the cladding resulting in very high local hydrogen concentrations (1000-3000 wtppm H). A fuel cladding with such high hydrogen concentrations will be very brittle even though the cladding is not oxidised at all, i.e. ECR is 0. The fuel clad axial temperature distribution will determine the axial elevation of the ballooned and burst fuel rods in the assembly. The axial and azimuthal fuel clad temperature distribution is a result of the heat transfer mechanisms at the surfaces of the cladding.

Increased clad temperature, time and steam pressure will increase oxidation. The clad temperature may increase further if the clad ballooning becomes large enough to restrict coolant flow and if fuel relocation occurs. The heat liberated in the zirconium-water reaction may contribute to the temperature increase. At the point when burst occurs, double-sided oxidation will occur of the fuel clad resulting in a faster embrittlement effect.

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<sup>22</sup> Fuel fragmentation is enhanced by the strong evolution of the temperature radial profile in the fuel pellets after the power shutdown: the initial parabolic temperature profile in the fuel pellets starts immediately to flatten, i.e., while the central zone is cooling down, the periphery of the pellet (high burnup structured region or HBS) will heat up, generate macroscopic stresses in the pellet and pressurize the gas bubbles at the grain boundaries, such favouring their decohesion and fragmentation.

<sup>23</sup> Fuel relocation can be described as any physical movement of fuel pellets or fuel fragments within the cladding. Fuel relocation may occur, if during LOCA a section of the fuel rod experiences ballooning, by slumping of fuel fragments from upper location in the ballooned section.

<sup>24</sup> The ECR is defined as the total thickness of cladding that would be converted to stoichiometric ZrO<sub>2</sub> from all the oxygen that are contained in the fuel cladding as ZrO<sub>2</sub>, and oxygen in solid solution in the remaining clad metal phase.

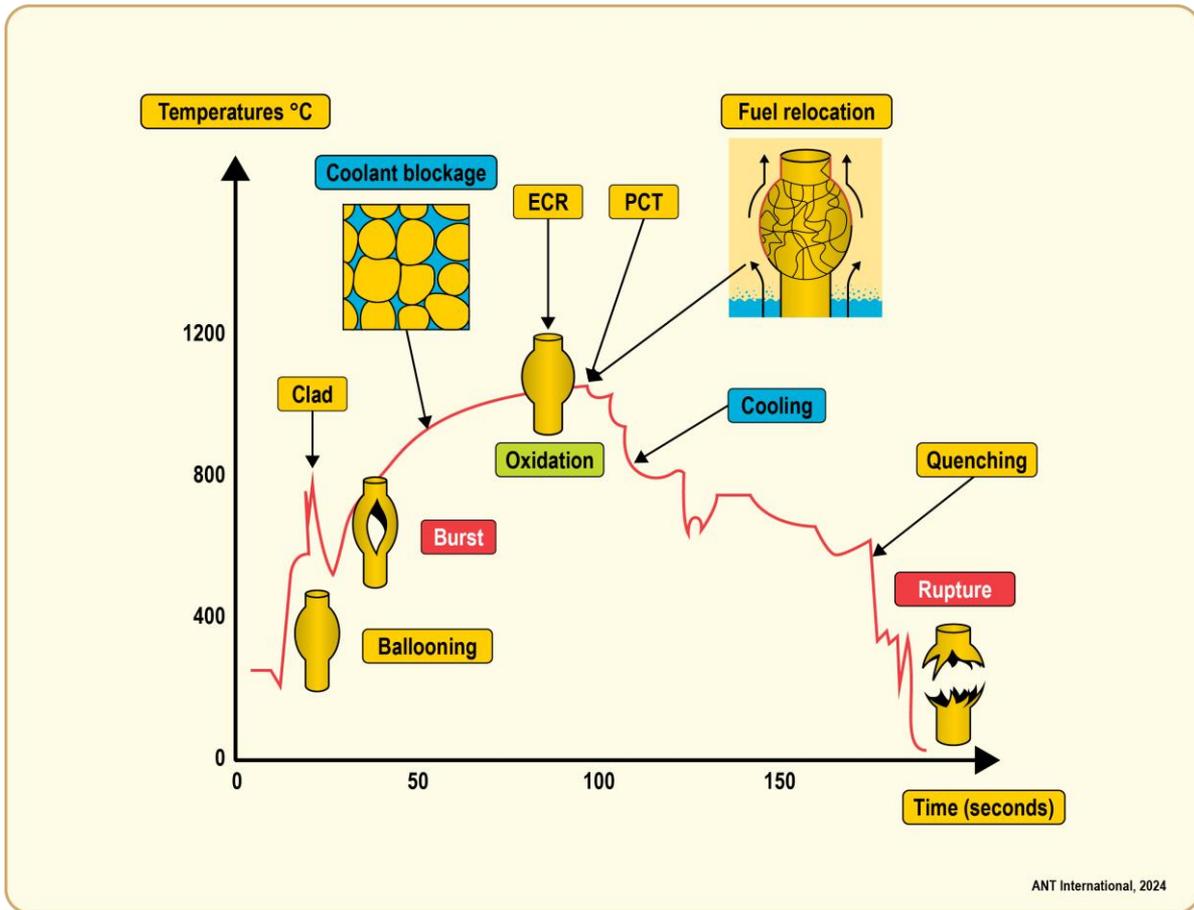


Figure 4-1: Typical temperature evolution during a PWR LOCA. Temperature rapidly increases initially after LOCA initiation because of temperature equalization from the hot fuel to the cladding (during normal operation, there is a strong temperature gradient).

Ballooning of the fuel rods may result in *blockage* of the coolant sub-channel that in turn may impact the fuel coolability. If large fuel clad burst strains occur at the same axial elevation, *co-planar deformation*, in the FA, the coolability may be significantly degraded.

The increasing temperatures and presence of steam will cause the intact cladding to oxidize on the outside surface OD<sup>25</sup> and the burst cladding to oxidize on both the outside and inside surfaces OD and ID (two-sided oxidation). The oxidation process at the high LOCA temperatures will increase the oxygen and hydrogen content in the cladding, reducing its ductility and resistance to rupture. Two-sided oxidation can have significant effects on the fuel rod resistance during the quench phase of the transient and the Post Quench Ductility (PQD) of the cladding, because of high, but localized HPU in addition to the high temperature oxidation.

The length of time at PCT is determined by the reactor system and regulated by the Equivalent Clad Reacted (ECR) limit, defined as the total thickness of cladding that would be converted to stoichiometric ZrO<sub>2</sub> from all the oxygen contained in the fuel cladding as ZrO<sub>2</sub> and oxygen in solid solution in the remaining metal phase. Since the 1950s many measurements of the rate of reaction of Zircaloy in steam have been made and in 1973 the USNRC adopted the isothermal parabolic rate constants published by [Baker & Just, 1962] for licensing calculation of oxygen uptake in cladding exposed to steam in LOCA thermal transients. Later data show that the Baker Just oxidation correlation (B-J correlation) is conservative and that the Baker-Just correlation overpredicts the reaction rate (Figure 4-2). ECR

<sup>25</sup> Regulatory bodies are also currently considering ID oxidation of intact cladding via transfer of oxygen from the fuel pellets.

calculations in the US have been changed to the Cathcart-Pawel equation (C-P correlation) by the US-NRC, Figure 4-2 although the LOCA limits (PCT and max allowable ECR) used by the designers were defined based on the B-J correlation. It has been shown also that the B-J correlation is better suited to simulate the effect of steam pressure on the oxidation of Zr based alloy (i.e., to represent small break LOCA scenarios where coolant pressure is maintained longer, during the high temperature oxidation phase).

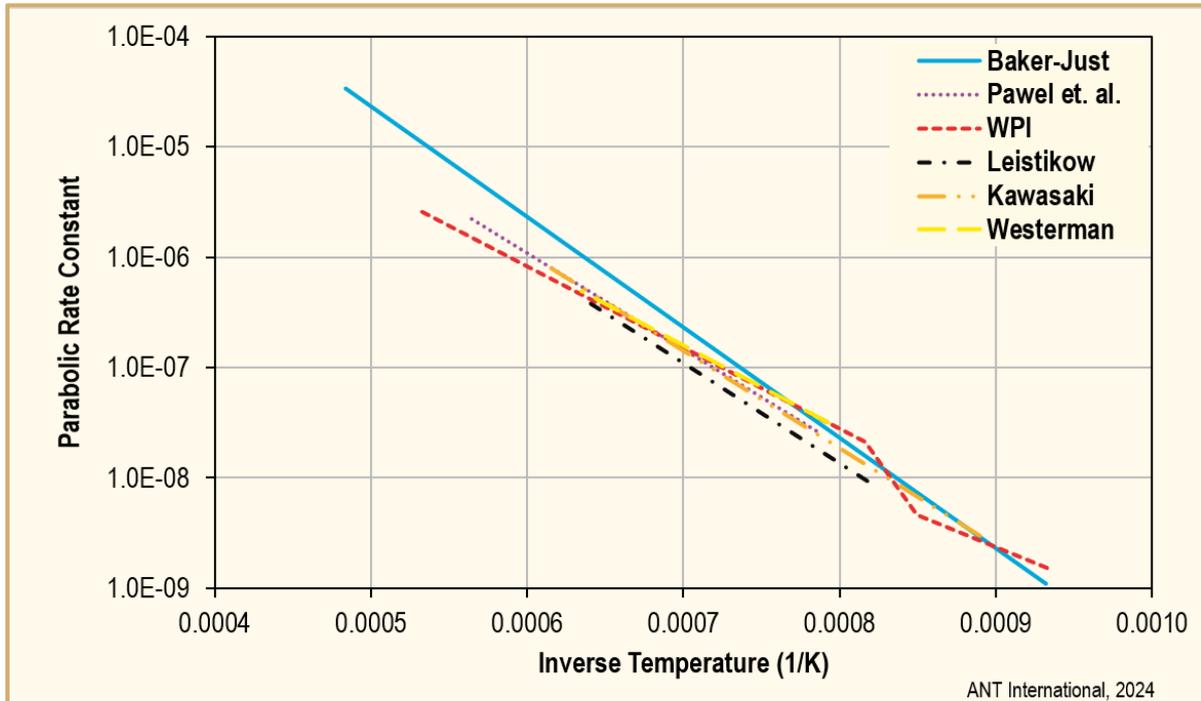


Figure 4-2: The parabolic rate constant as a function of inverse temperature as determined by different researchers. The unit for the parabolic rate constant is  $(\text{g O}_2/\text{cm}^2)^2 \text{ s}^{-1}$ , modified figure according to [NEA Report No. 6846, 2009]

The cladding continues to oxidize until the ECCS becomes effective and a PCT is reached. The maximum PCT is regulated to be a maximum of 2200 °F by the USNRC and 1204 °C internationally.

The ECCS activation will stop the temperature rise and start cooling the core by injection from the bottom of the core in a PWR/VVER and from the top in a BWR. The re-entrance of the coolant via the ECCS will quench and cool the cladding. The “cooling” process as shown on Figure 4-1 is relatively slow until the emergency coolant contacts the hottest fuel rods in the reactor core. At that point, in the range of 400° to 800 °C, identified as “quenching temperature” in Figure 4-1 the water from the ECCS will reduce further the cladding temperature at a rapid rate (1°–5 °C/sec) by re-wetting. Thermal shock due to the sudden change in heat transfer conditions may *rupture* the cladding if the cladding ductility, dependent on the extent of oxidation and hydriding that occurred during the LOCA transient (i.e., degree of cladding embrittlement enhancement in addition to hydrogen content prior to the transient) is insufficient to accommodate: 1) the strain due to the sudden change in heat transfer conditions or 2) the differential expansion of FA components (implying axial tensile loadings to the fuel rods). The ability of the cladding to withstand the thermal stresses will depend on the extent of oxidation that occurred during the heat-up stage of the transient. Since, however, all the hydrogen is likely to be in solution at these temperatures, hydrogen is unlikely to contribute to the embrittlement during the heat up phase except to the extent that its presence increased diffusivity of oxygen in the  $\beta$  phase (Zr has two different crystallographic structures, a low temperature hcp structure -  $\alpha$  phase- and a high temperature fcc

structure,  $\beta$  phase, see Figure 4-3<sup>26</sup>) at high temperatures and the oxygen solubility and increased oxygen content, in turn, increased the embrittled condition of the cladding.

Upon further cooling, the hydrogen stabilized  $\beta$  Zr phase will decompose below the  $\alpha$ - $\beta$  transformation temperature, Figure 4-4, and the hydrogen will precipitate in the form of hydrides in the  $\alpha$  phase and embrittle and potentially crack the cladding.

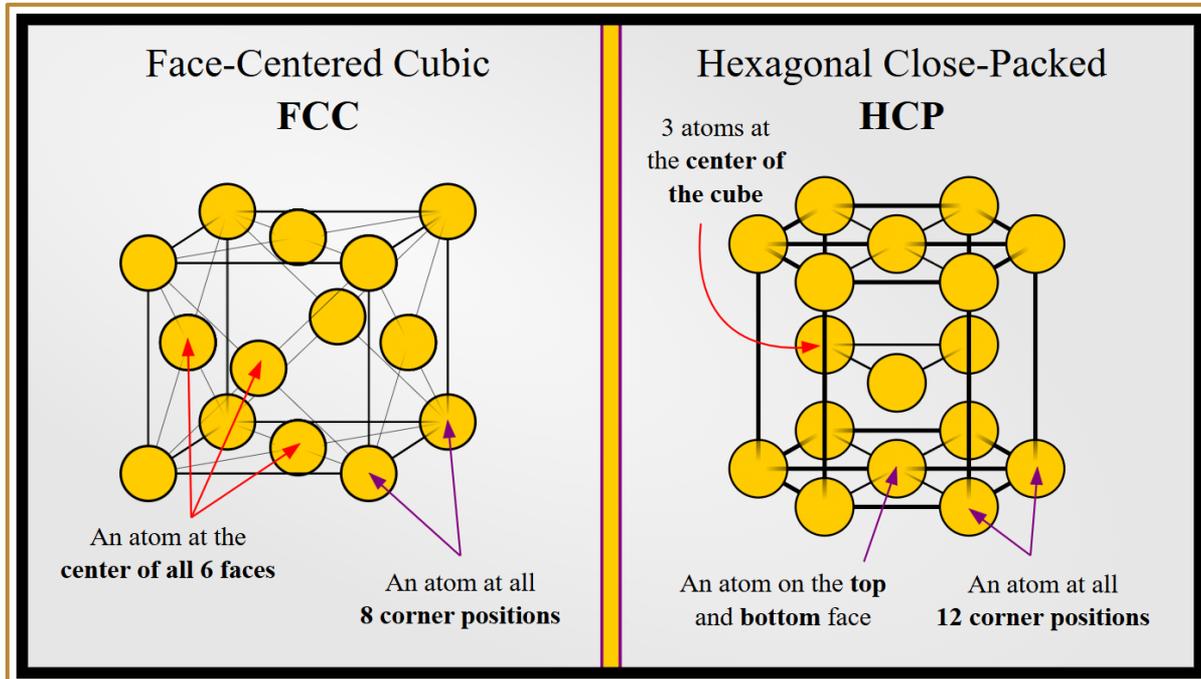


Figure 4-3: Schematics showing the difference between FCC and HCP structure.

<sup>26</sup> FCC and HCP are both close-packed with a 74% atomic packing factor, 12 nearest neighbours, and the same interstitial sites. However, HCP only has 3 slip systems, while FCC has 12 slip systems, which lead to very different mechanical properties. The most direct difference between FCC and HCP crystals is in the atomic arrangements. The face-centred cubic structure is a cube with an atom at all 8 corner positions, and at the centre of all 6 faces. The hexagonal close-packed structure is a hexagon with an atom at all 12 corner positions, an atom on the top and bottom face, and 3 atoms in the centre.

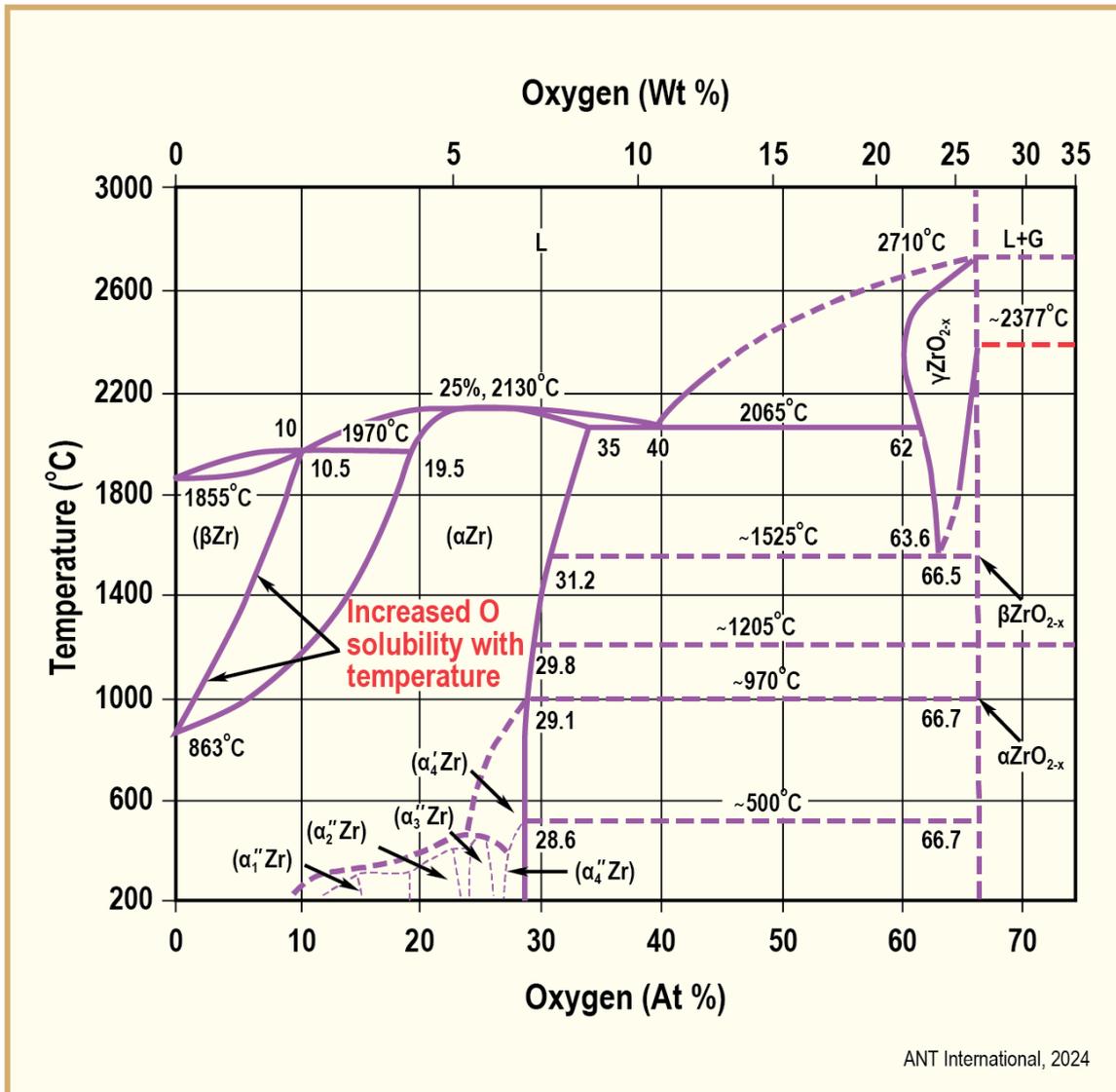


Figure 4-4: Zr-O phase diagram, modified figure according to [Abriata, et al., 1986].

At temperatures of about 135°C (saturation temperature after the LOCA event) the forces imposed on the FA by the coolant, by handling, by transport and by hypothetical accidents such as seismic events have the potential of cracking the embrittled cladding. The embrittlement effect of hydrogen (in form of hydrides) on ductility during this phase of the LOCA is much stronger than during the quenching phase. The volume fraction of hydrides precipitated will be a function of the cooling rate.

Preserving post-quench integrity of the cladding has been used by the US-NRC as an easy way to demonstrate that massive shattering of the fuel bundle rods within the reactor core (i.e., with a risk of flow blockage or re-criticality, such challenging long-term cooling of the core), is unlikely. The post-quench ductility is based partly on the properties of the former  $\beta$  zone, since the ZrO<sub>2</sub> and oxygen stabilised  $\alpha$  zones are too brittle to sustain a load.

The embrittlement criteria are based on properties of the prior  $\beta$  layer measured on post-simulated LOCA tests of unirradiated and irradiated Zr based alloys, either by using ring compression tests, see Appendix F - [Hobson & Rittenhouse, 1972] or axial loading applied to the fuel rods during the quenching phase of semi-integral LOCA tests. In addition, embrittlement by oxidation at high temperature is limited by the time of exposure, i.e., through the Equivalent Cladding Reacted coefficient or ECR, calculated by the Baker-Just equation. Critical values were defined by the US-NRC [Hobson & Rittenhouse, 1972] as PCT  $\leq 1204^\circ\text{C}$  and ECR  $\leq 17\%$  (it should be noted that the oxide thickness of the samples was never actually measured during the tests used to defined these critical values).

Nevertheless, they have been confirmed later on more prototypical test conditions, including, for instance, the potential effect of transient secondary hydriding which occur only on semi-integral LOCA tests).

*It is important to note that variables in the oxidation cycle as well as the subsequent cooling and quenching cycles assumed in the DBA will have a significant effect on the post-LOCA properties and these include:*

- The rate of temperature increase to the Peak Cladding Temperature, PCT, Figure 4-1 will affect the total oxidation ratio.
- The oxidation of the cladding during the high LOCA temperatures phase will increase the oxygen content in the cladding, reducing its ductility and resistance to rupture.
- Maintaining PCTs at <1200 °C will result in improved Post Quench Ductility (PQD).
- Decreased time at PCT will decrease the ECR, Equivalent Cladding Reacted, and also improve PQD.
- Slow cooling rates from PCT to the quenching temperature will increase the PQD.
- Lower quenching temperatures, such as 600 °C, instead of 800 °C will increase the PQD.
- Lower quenching rates from the quenching temperature will also increase the PQD.

There are several test methods to evaluate the effects of oxidation and quenching on the integrity of the cladding.

*Mechanical tests, e.g ring compression tests, see Appendix F -".* The samples used are usually unirradiated, pre-hydrated and oxidized to represent the pre-quench conditions. The alternative is to remove the PQD ring specimens from a rodlet after an integral thermal shock test, which includes the ballooning and oxidation cycle. In that case however the ballooned-burst region, as well as the transient secondary highly hydrided area, below and above the burst opening, cannot be tested according to the PQD ring compression tests.

*Semi-integral tests (either out-of-reactor or in-reactor ), see Appendix F -,* appear to approach LOCA conditions more closely and consist of isothermal heating of the fuel to the selected oxidation temperature and then quenched with water from a selected temperature at a selected rate. In Japan and in France, the fuel rodlets are submitted to a mechanical tensile axial load during the quenching phase. Critical LOCA limits are subsequently defined based on the survivability of the tested rodlets.

In summary, during and after the LOCA regulations requires that:

- A. The core remains coolable (which means that the maximum allowable coolant blockage is limited) and,
- B. No fuel dispersal occurs (which means that total cladding rupture is not allowed; it is assumed that the cladding burst is so small that only fission gases are released).
- C. The maximum amount of hydrogen generated by the oxidation reaction of the cladding will be 0.01 times (or 1% of) the hypothetical amount generated if all the cladding adjacent to the fuel columns were to react.

An embrittled fuel cladding could potentially result in loss of fuel coolable geometry due to fuel clad fragmentation. To ensure coolability during a LOCA, A. and B. above, there are two criteria that are addressing clad embrittlement:

- *Peak Cladding Temperature, PCT, < 1200-1204°C* dependant on country (or 2200°F in US), see Section 5.3.

- *Equivalent Cladding Reacted, ECR*<sup>27</sup>, < 15-18% dependant on country, See Section 5.3.

Modelling of the performance of the fuel during the LOCA requires knowledge of the mechanical properties of the cladding at the initiation of and during the LOCA as determined by the reproduction of the LOCA conditions in the test methods. The LOCA conditions used for modelling, such as the temperature–time history, also need to be representative of the conditions expected in the reactor that is being evaluated.

The response to a LOCA can vary from alloy to alloy, among the different varieties of a single alloy and the different power histories the fuel rods had been exposed to at the time of the LOCA. Similarly, the different mechanical property tests that simulate LOCA conditions can give different results for the same material variant. Awareness of these potential differences is important.

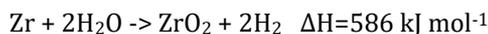
Since the mechanical properties of the prior  $\beta$  zone could change as a function of extended burnup (e.g., pre-transient hydrogen content, transient fission gas release, fuel fragmentation and relocation, etc.,) the US-NRC and other organizations have evaluated the effects of extended burnup on the fuel rods properties and the potential need for changing the criteria. The US-NRC and other regulators have issued updated rules, maintaining the former PCT value while ECR is expressed as a function of the cladding pre-transient hydrogen contents.

In the next subsections, the following topics will be described: Ballooning, burst, oxidation, cooling/quenching and embrittlement/rupture.

## 4.2 Oxidation and Hydrogen Pickup Effects on Fuel Clad Embrittlement

### 4.2.1 Introduction

Corrosion of zirconium alloys is a thermodynamic and electrochemically based process:



Prior to a hypothetical LOCA, Zr alloys (essentially pure Zr with a small amount of alloying elements, see Appendix C -for more information about alloys being used) are affected by the following parameters, see Figure 4-5:

- The microstructure of the Zr alloy-metal surface.
- The water chemistry and the hydraulic conditions.
- The Zr alloy temperature (at the metal/oxide interface).

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<sup>27</sup> The *ECR* is defined as the total thickness of cladding that would be converted to stoichiometric  $\text{ZrO}_2$  from all the oxygen that are contained in the fuel cladding as  $\text{ZrO}_2$ , and oxygen in solid solution in the remaining clad metal phase.

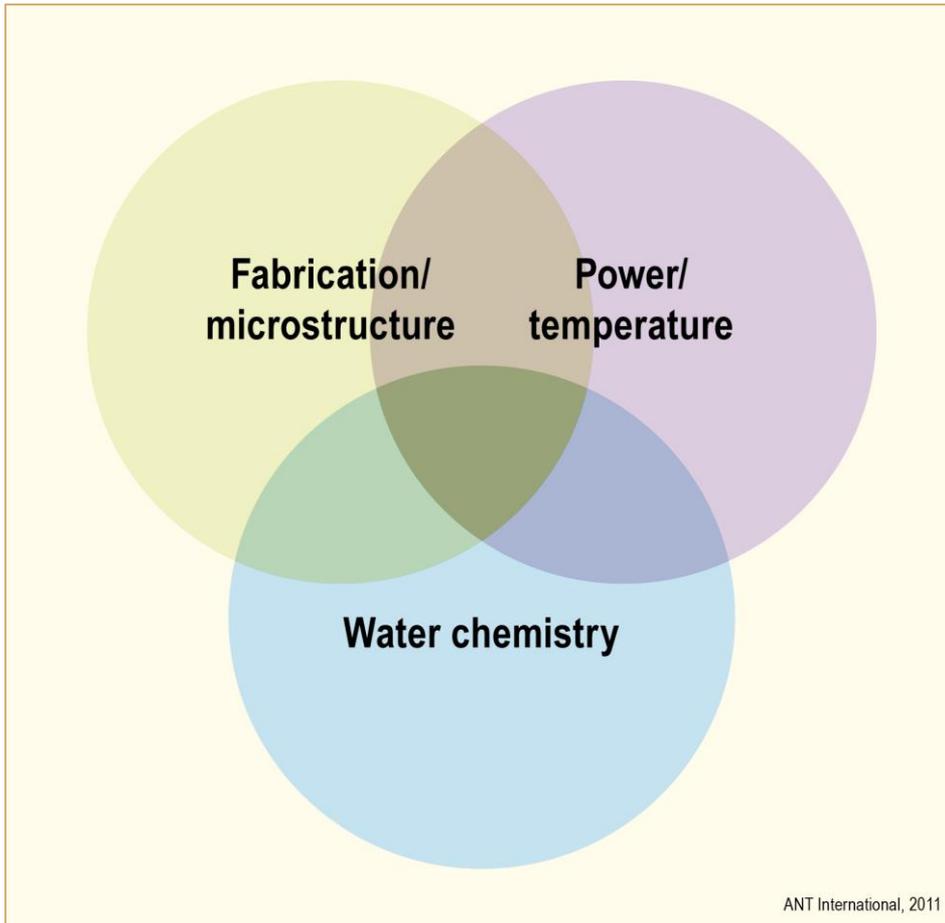


Figure 4-5: Parameters impacting corrosion performance of Zr Alloys.

Irradiation affects the metal microstructure, the oxide properties and the water chemistry.

The in-reactor oxidation/corrosion of zirconium alloys, i.e., prior to a hypothetical LOCA, generates a thin protective black oxide. As the zirconium oxide grows in thickness the outer part of the oxide (phasing the water/steam phase) is transformed into a greyish porous oxide. The oxide grows inwards into the zirconium alloy material because of oxygen diffusion from the coolant through the growing oxide scale to the oxide/metal interface.

During the oxidation/corrosion process, a certain fraction of the hydrogen in the water molecule, from the zirconium alloy corrosion process, is picked up by the zirconium alloy. This HPUF, depends on zirconium alloying content but also on temperature, water chemistry, and reactor start-up procedure. The total amount of hydrogen that is picked up by the zirconium alloy is the product of the corrosion rate and the HPUF.

The corrosion and hydriding process of zirconium alloys is schematically shown in Figure 4-6.

## 5 LOCA Safety acceptance criteria and their background

### 5.1 Historical LOCA acceptance criteria

#### 5.1.1 Historical Background

##### 5.1.1.1 Introduction

Historically the LOCA limits have been defined by the US-NRC after the 1973 Public Hearing, at a time the discharge burnups were lower than 33 GWd/tM compatible with the in-reactor corrosion performance of Zr based alloys (e.g., Zry4, Zry2).

For several decades, the rules didn't evolve while the fuel burnups were continuously increasing. In October 1993, the RIA REP-Na1 test in the CABRI sodium loop in France led to a premature failure of high burnup fuel rod, with fuel dispersal, at an enthalpy much lower (30 cal/g) than the historical limits (230 cal/g for failure and 280 cal/g for fuel dispersal). This unexpected result confirmed that historical safety limits should be reviewed, to account (i) for the new operational conditions (i.e., higher burnups, longer cycles, higher duties), (ii) for the new cladding materials developed by the fuel vendors to mitigate in-reactor corrosion and hydriding (e.g., 1% Nb Alloys, MDA, duplex claddings) and (iii) for the higher performance of the simulation tools enabling including the new physical phenomena susceptible to take place during the various LOCA phases.

A comprehensive review of all the Specified Acceptable Fuel Design Limits (SAFDLs) provided in the US-NRC Standard Review Plan 4.2 and the fuel-related General Design Criteria of 10CFR50.46, has been carried out by EPRI in the late 1990's and published in 2006 [EPRI Report, 2006]. The process developed to conduct this review consisted of four stages.

1. Establish Baseline for Current Fuel System Designs
2. Perform Burnup Effects Screening Evaluation
3. Perform Comprehensive Burnup Effects Evaluation
4. Establish New Baseline for Licensing Extended Burnup

The outcome of this review is that all design criteria but two can remain unchanged provided actual properties evolution with burnup are properly simulated in the codes and methods used by the designer. The two design criteria which requested significant changes and further investigations were the RIA limits and the LOCA limits.

Within this context, US-NRC launched its high Burnup Plan late 1990s to document the impact of fuel burnup on those design limits [Chung, H.M. et al., 1996]. Similar initiatives took place in France and in Japan.

However, to correctly interpret the results of such investigations, and if necessary, to establish new embrittlement thresholds that maintain an adequate safety margin for high-burnup operation, it appears necessary to accurately understand the rationale, history, and data bases used to establish the current LOCA criteria, i.e., the maximum cladding temperature limit of 1204°C (2200°F) and the maximum oxidation limit of 17%. The third criterion, related to the maximum acceptable hydrogen production during the LOCA transient ( $\leq 1\%$ ) concerns the whole core structures. As such it is not included in this document on fuel safety design.

##### 5.1.1.2 Background

###### The 1972 US Atomic Energy Commission Public Hearings

In 1967, an Advisory Task Force on Power Reactor Emergency Cooling [NRC, 1971], which was appointed to provide "additional assurance that substantial meltdown is prevented" by an Emergency Core Cooling system (ECCS), concluded that it is paramount to preserve the heat transfer area and the coolant flow geometry not only during the short-term portion of the temperature transient in the core but also for the long term.

Then a Public Rulemaking Hearing took place in 1972 and 1973 [Cottrell, W.B., 1974] and the current LOCA embrittlement criteria (i.e., 1204°C peak cladding temperature and 17% maximum oxidation limits) have been defined, for zero- or low-burnup UO<sub>2</sub> fuel rods with Zircaloy claddings [Hache, G. and Chung, H.M., 2001].

### The Primary Objectives of the LOCA Criteria

Following the Advisory Task Force recommendations, the U.S. Atomic Energy Commission (AEC) required that: "... fuel and clad damage that could interfere with continued effective core cooling is prevented" and that: "the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching."

When evaluating the evolution of core coolability during the various phase of a LOCA, designers usually distinguish the ductile mode (ballooning and burst of the cladding during the high temperature phase) and the brittle mode (failure of the fuel rod during or after the quenching phase). Our focus below is on the change in coolable geometry due to cladding embrittlement and failure.

### Metallurgy of Cladding Embrittlement

Results of metallurgical examination of fuel claddings tested in the 1960s at ANL, INL and ORNL (Argonne, Idaho and Oak-Ridge National Labs) showed that embrittlement was caused by severe microstructural modification of the cladding. Brittle cladding cross sections exhibited oxide layer, oxygen-stabilized  $\alpha$  phase layer and a region of acicular prior  $\beta$ -phase. The results were later reported by [Fujishiro et al, 1978].

As a result of these observations, the scientific community was alerted to the fact that oxidation of Zircaloys above the  $\alpha$ -to- $\beta$  transformation temperature results in the formation of inherently brittle phases, i.e., Zr oxide, oxygen stabilized  $\alpha$ -Zr (fcc structure), and diffusion of oxygen into the underlying  $\beta$  phase (bcc structure)- See section 4.1 for difference between  $\alpha$ -Zr and  $\beta$  Zr. This is shown schematically in Figure 5-1.

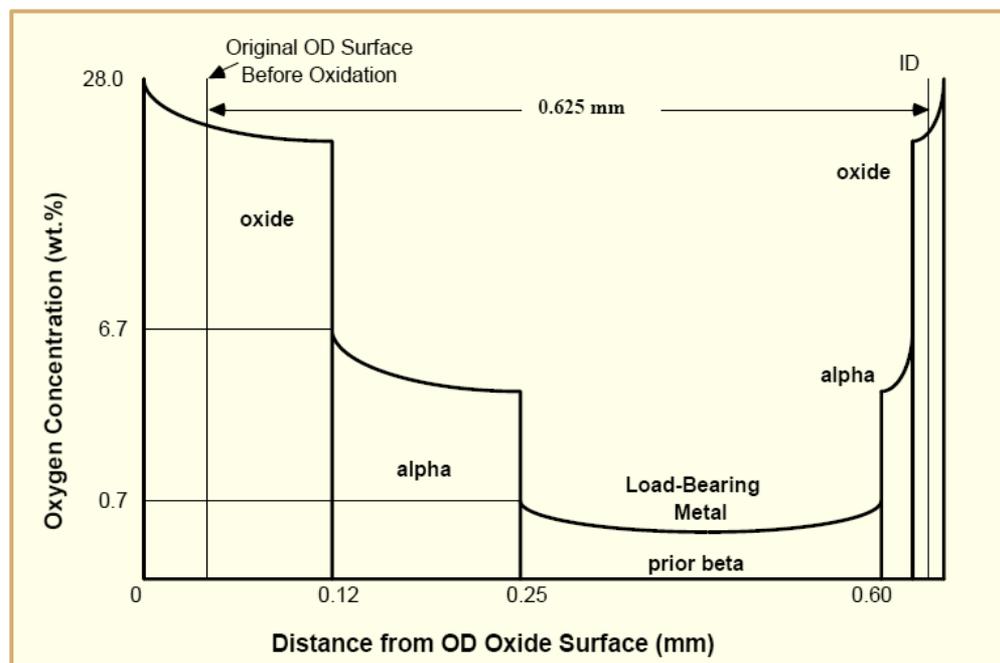


Figure 5-1: Schematic oxygen distribution in oxide, stabilized  $\alpha$  and prior- $\beta$  (transformed- $\beta$ ) layers in Zircaloy cladding after oxidation around 1200°C.

Significantly embrittled cladding can fragment during the quenching phase of a LOCA. The action of rewetting by ECCS water involves the collapse of the vapor film that covers the cladding outer-diameter (OD) surface prior to subsequent transition to nucleate boiling. This event takes place at a specific temperature, i.e., the Leidenfrost temperature. For oxidized Zircaloy-4 cladding rewetted by bottom-flooding water, ANL investigators reported that rewetting occurs in the range of 475-600°C [Chung, H.M. and Kassner, 1980]. The abrupt change in the heat transfer conditions induces large thermal-shock stresses, which in turn can fracture the cladding, if it is sufficiently embrittled by the transient oxidation.

To enable long-term cooling, regulator staff and commissioners mentioned, with other experts [Parsons, P.D. et al., 1986]: "The ability of the cladding to withstand the thermal-shock stresses of quenching during rewetting **or post-LOCA forces** is related to the extent and detailed nature of oxidation during the transient. The post-LOCA forces, which need to be taken into account, are the hydraulic, seismic, handling, and transport forces."

If post-LOCA forces must be considered (i.e., at a temperature below the Leidenfrost temperature), the risk of fragmentation cannot be precluded for 2 reasons: (i) the embrittlement effect of oxygen dissolved in the  $\beta$  phase is more pronounced at low temperature and (ii) the effect of pre-transient hydrogen and transient hydrogen up-take is more detrimental at temperature lower than 550°C.

### **Axial mechanical loading during the quench phase**

Later, assuming possible rod-to grid lock-up and differential thermal shrinkage between the fuel rod bundle and the guide tubes, the commissioners claimed that mechanical constraints during the quenching phase could occur. In recognition of this, the Regulatory Staff wrote during the 1973 Rule-Making Hearing that "the loads due to assembly restraint and rod-to-rod interaction may not be small compared to the thermal shock load and cannot be neglected [Atomic Energy Commission Rule-Making Hearing, 1972]." Nevertheless, the Commissioners added: "There is some lack of certainty as to just what nature of stress would be encountered during the LOCA.... (We want) to draw attention to the fact that it may not be possible to anticipate and calculate all the stresses to which fuel rods would be subjected during a LOCA transient. Although we believe the calculations of thermal shock stresses are worthwhile and informative, we agree with the regulatory staff that they are not sufficiently well defined to depend on for regulatory purposes."

Although reluctant to neglect the effect of axial mechanical loading during the quenching phase of the LOCA transients, the regulatory staff finally decided to consider that the safety margins provided by the proposed LOCA limits (17% ECR and 1204°C PCT) were sufficient to cover the "unknown phenomena", such as mechanical loadings, which could occur during a LOCA transient.

The available margins brought by the LOCA limits were confirmed later through a series of thermal shock tests performed on unconstrained cladding samples and partially constrained cladding tubes (Figure 5-2).

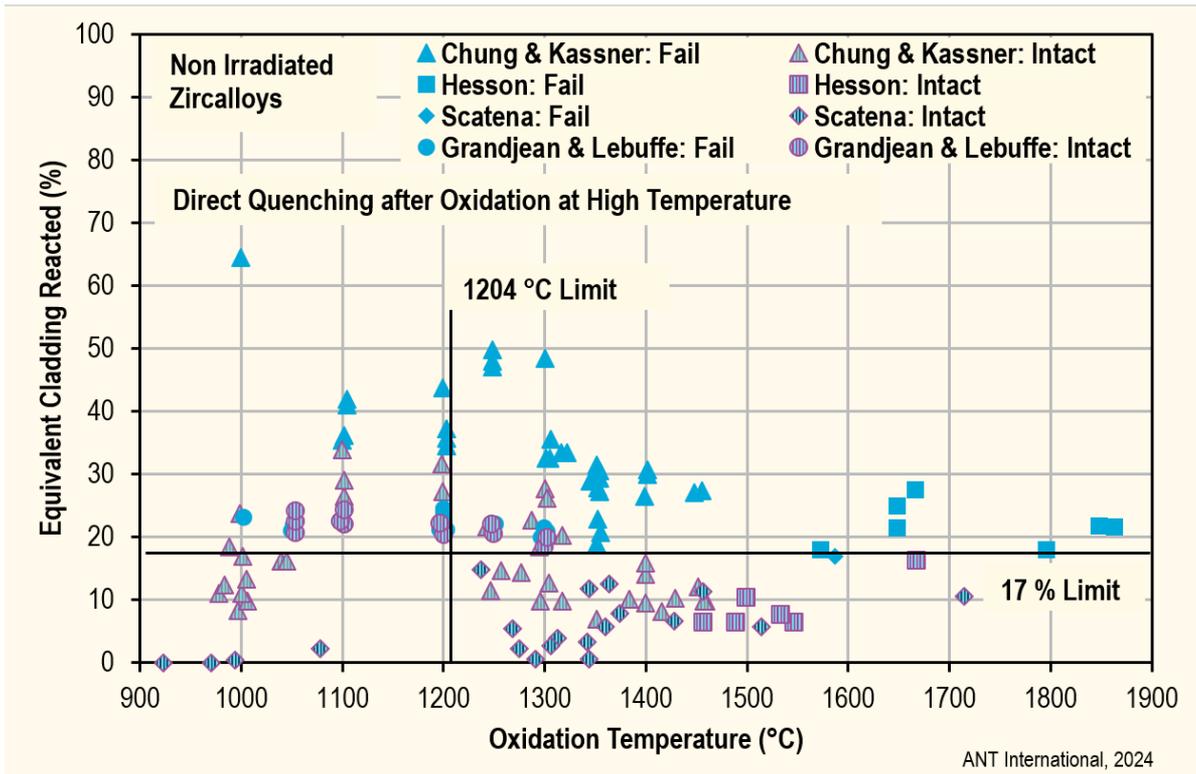


Figure 5-2: Failure boundary of partially constrained Zircaloy cladding tubes or unconstrained rings after oxidation at high temperature and penalizing direct quenching from peak oxidation temperature, modified figure according to [Chung, H.M. and Kassner] and [Grandjean, C., 1996].

Opposite to the US approach, the Japanese considered from early 1980's that the axial mechanical loading during the quench phase should be included in the LOCA limits definitions. To address this issue, Uetsuka et al. performed at JAERI, a series of quenching tests on cladding tubes under severely constrained conditions [Uetsuka, H. et al., 1983]. In these experiments, the cladding tube is fixed at the bottom but is allowed to freely elongate in axial direction during oxidation at high temperature. As a result, the cladding tube is free to expand axially because of thermal expansion and oxide-induced creep. At the end of the isothermal oxidation, the top-end of the tube is locked to the crosshead of an Instron tensile machine to monitor the load-time curve during the quenching phase. Thus, at the Leidenfrost temperature, the cladding tube is subjected to a combination of axial-tensile and thermal-shock stresses. The results of the tests are summarized in Figure 5-3 (constrained test conditions) and Figure 5-4 (unconstrained test conditions).

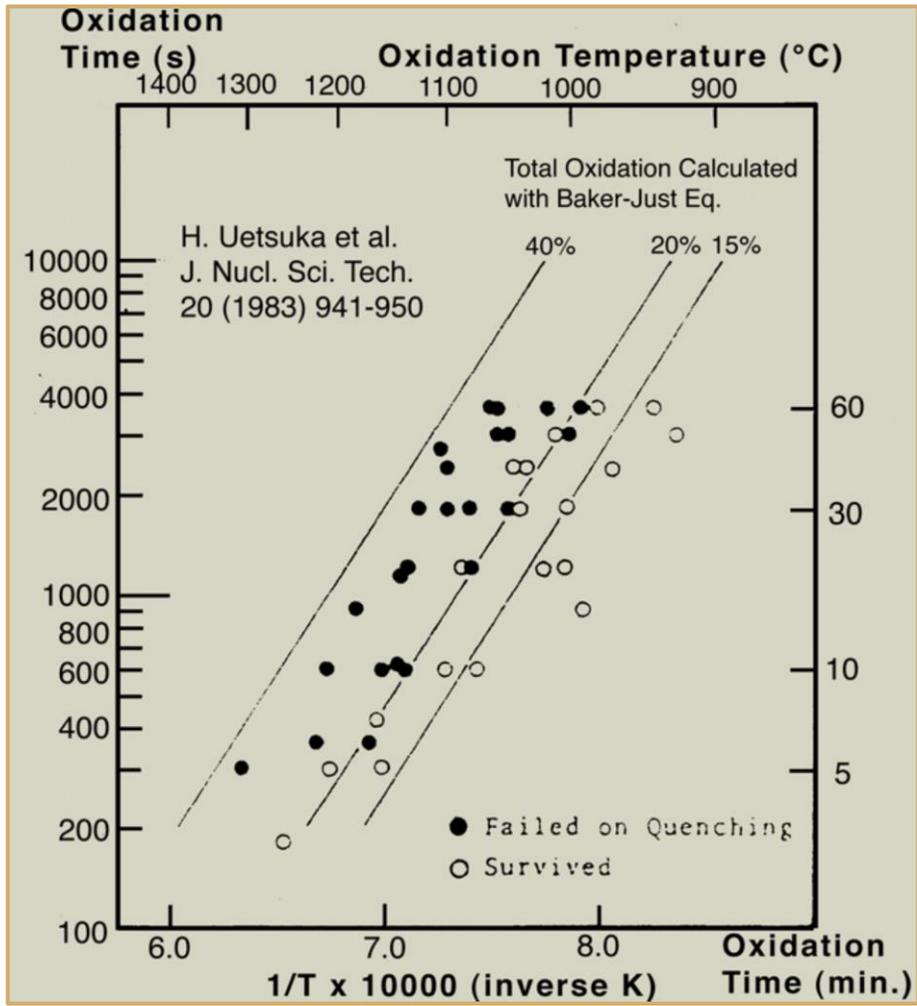


Figure 5-3: Failure-non failure boundary for fully constrained Zircaloy-4 after oxidation in steam and quenching as function of oxidation time and temperature; Total oxidation is calculated with Baker-Just equation, modified figure according to [Uetsuska et al., 1983].

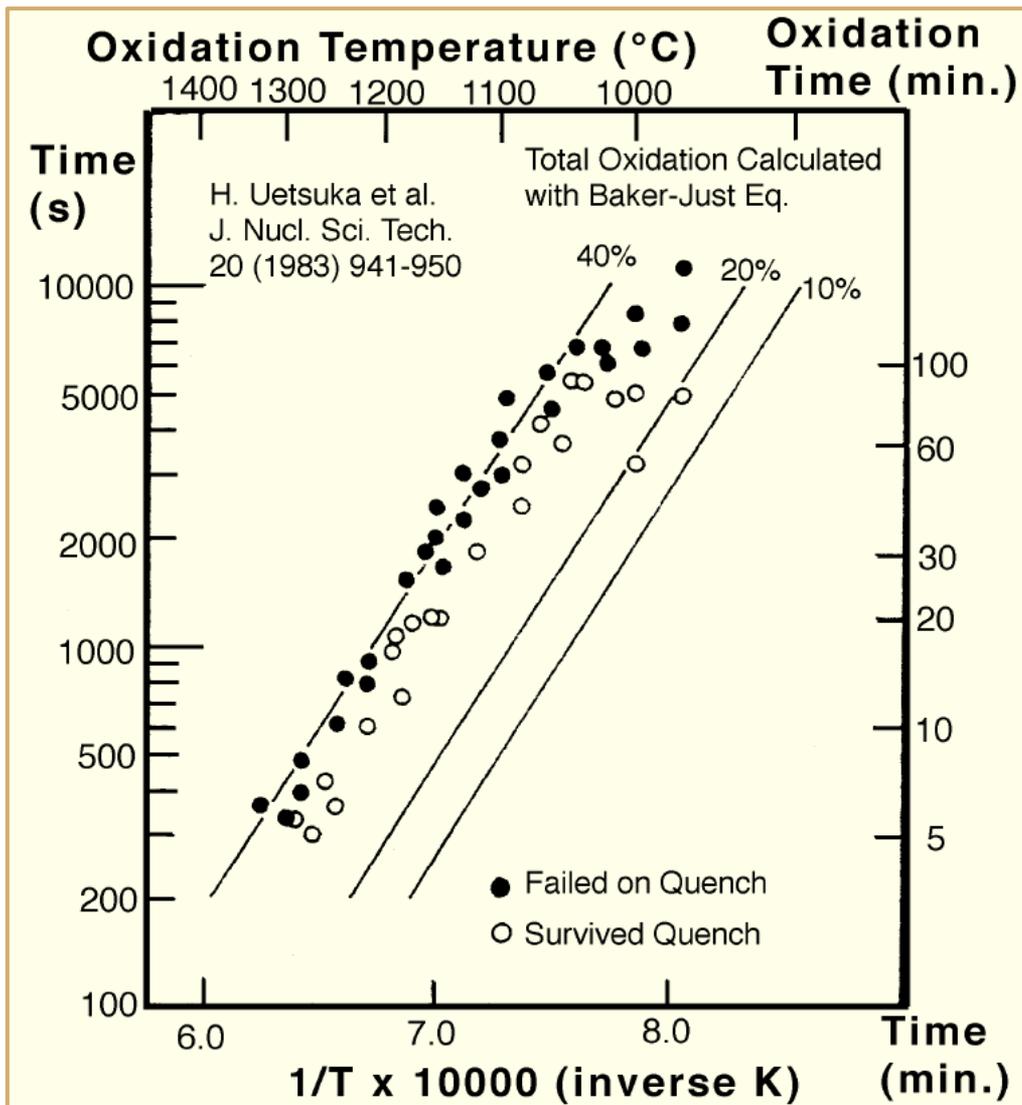


Figure 5-4: Failure-non failure boundary for unconstrained Zircaloy-4 after oxidation in steam and quenching as a function of oxidation time and temperature; Total oxidation is calculated with Baker-Just equation, modified figure according to [Uetsuska et al., 1983].

A comparison of the results from the two contrasting types of tests shows a significant effect of the mechanical constraint. The 17% oxidation limit, calculated with Baker-Just correlation, appears to be adequate for protection of constrained rods against thermal-shock failure (Figure 5-3), whereas a large margin is available for unconstrained rods (Figure 5-4).

However, the prototypicality of full constraint has been thoroughly discussed in Japan and in France (where a similar approach has been adopted in the 2010's). Separate effect tests and simulations calculations show that the additional axial mechanical loading during the quench phase is neglectable or very low (lower than 250N per rod) [Waeckel and al, 2015]. To assess the LOCA limits, it has been decided to choose a conventional mechanical loading of 540N /rod.

### 5.1.2 The oxidation criterion : $ECR \leq 17\%$ .

As indicated in a few reports [Parsons, P.D. et al., 1986] that reviewed the results of the LOCA-related tests performed before and after the 1973 Hearing, the 17%-ECR and 1204°C criteria were primarily based on the results of post-quench ductility tests conducted by Hobson [Hobson, D.O., and Rittenhouse, P. L., 1972] and [Hobson, 1973].

Figure 5-5 summarizes the results of Hobson's ring compression tests performed at 23-150°C.<sup>44</sup>

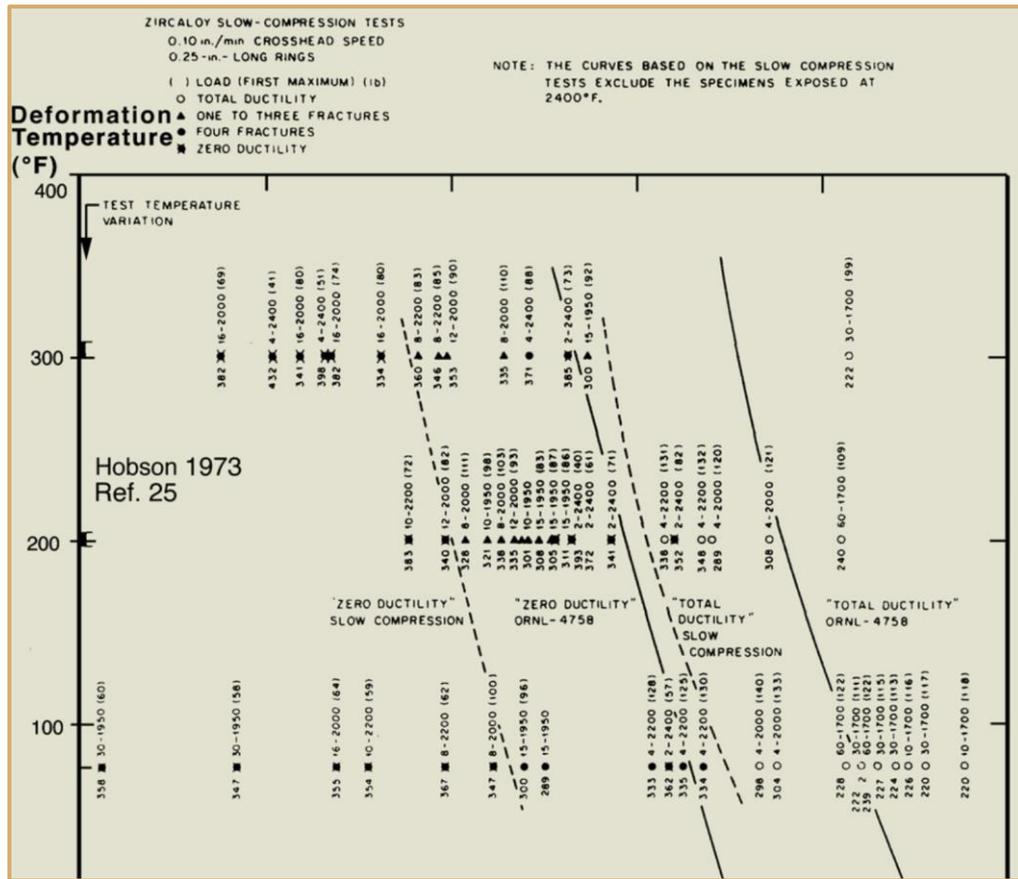


Figure 5-5: Ductility of two-side-oxidized Zircaloy rings as function of slow- or fast-compression temperature and fraction of transformed-β-layer, modified figure according to [Hobson, 1973].

Zircaloy-4 cladding tubes were oxidized in steam on two sides, followed by direct quenching into water. Then, short ring specimens cut from the oxidized tube were either compressed slowly to a total deflection of 3.8 mm or squashed by impact loading. After the test, the broken pieces of the ring were assembled back to determine the degree of brittleness. Zero ductility was defined based on the macroscopic geometry of the broken pieces and the morphology of the fracture surface on microscopic scale. Each data point in Figure 5 indicates failure type, test identification number, oxidation time in min., oxidation temperature in °F, and first maximum load in pound.

The dashed line on the left side of Figure 5 denotes the zero-ductility domain for slow-compression rate. This domain is valid only for oxidation temperatures lower than 2200°F or 1204°C.

Also note that the 17% ECR (equivalent cladding reacted) limit was derived based on zero ductility data obtained from slow compression test only, that is, excluding similar data obtained from impact-type ring-crushing test. If the latter type of data were used, the ECR limit would have been ≈14%. Subsequently further impact tests provided support for the 17% value.

The original criterion for embrittlement was defined in the absence of a good method of calculating oxygen distribution in the metallic layers, so regulators used the time required to accumulate 17% oxidation on the surface of the cladding as a surrogate measure of the time required to diffuse enough oxygen into the underlying metal to embrittle it. This is reasonable because both processes (oxide growth and oxygen diffusion in the metal) obey kinetics dependent on oxygen diffusion rate, but the precise value of 17% is particular to the correlation which has been used. Therefore, if the 17% ECR

<sup>44</sup> 135°C is the saturation temperature during reflood phase

criterion is specific to Baker-Just correlation, the same correlation must be used to determine the degree of total oxidation of the fuel rod cladding during the LOCA transient. **If an oxidation correlation other than the Baker-Just equation (e.g., Cathcart-Pawel correlation) is used, the threshold ECR would be lower than 17%.** This means that using a best-estimate correlation may not necessarily be conservative in evaluating post-quench cladding ductility.

After the 1973 Rule making hearings, several investigators tried to define an ECR criterion based on more physical parameters than the residual zero-ductility deduced from the Hobson ring compression tests.

In 1974, Pawel proposed a limit based on the oxygen diffusion in the cladding substrate (i.e., 0,7% oxygen in the  $\beta$  layer), but the approach was considered too difficult to be used within a licensing framework [Pawel, R.E., 1974].

In 1980, using an evaluation parameter consisting in measuring  $\alpha$  and  $\beta$  phases layers thicknesses rather than ECR (which is supposed to be more accurate), Hee Chung confirmed that the Hobson 17% ECR limit is equivalent to a 0.3-J impact failure limit, obtained on non-irradiated, pre-oxidized, ballooned and ruptured Zircaloy cladding tubes, as long as the peak cladding temperature remains below 1204°C, Figure 5-6 [Chung, H.M. and Kassner, 1980].

However, the 17% limit appears to be inadequate to ensure post-quench ductility for hydrogen uptake >700 wppm. Such hydrogen levels can be reached at high burnups for the lowest performance claddings or for all cladding types during a LOCA-like transient by secondary transient hydriding, in localized annular regions above and below the ballooned and burst area. This aspect, which is not covered by the US-NRC 17% ECR limit based on ring compression tests<sup>45</sup>, is accounted for by the Japanese and the French strength-based approach based on semi-integral LOCA tests including the secondary transient hydriding areas.

The hydrogen effect on O diffusion and residual ductility, therefore on the criteria, will be further discussed in the following sections. The LOCA limits and the corresponding licensing approaches adopted in several countries are gathered in Section 5.3.

More details on the Public Hearing and the way the 17% ECR limit has been defined and justified are described in the report written by [Hache, G. & Chung, H.M., 2001].

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<sup>45</sup> It seems that this effect of large hydrogen uptake was not known at the time of 1973 Hearing.

## 6 LOCA testing methodologies

### 6.1 Introduction and overview

During the LOCA event, the cladding is primarily embrittled by the formation of  $ZrO_2$  and oxygen stabilized  $\alpha$  Zr(O) as well as by hydrogen pickup during irradiation and high temperature steam oxidation during LOCA. The prior  $\beta$ -phase layer passes on the fracture resistance to the cladding and can prevent fragmentation during reflood. However, if the maximum LOCA oxidation temperature becomes too high also the  $\beta$ -phase will become brittle due to its increased oxygen solubility with increased temperature. Increased thickness and ductility of this phase will consequently reduce the risk of clad fragmentation during the thermal shock during LOCA quenching and the post-LOCA phase. The post-LOCA forces which need to be considered are the hydraulic, seismic, handling and transport forces. Specifically, the impact of hydrogen on clad ductility is very much dependant on temperature. The lower the temperature the larger the impact and consequently hydrogen has less embrittlement effect during the quenching phase as compared to post-LOCA events.

LOCA testing methodology includes many different techniques. An important part of the LOCA testing methodology is related to the emergency core cooling system (ECCS) acceptance criteria, which define the maximum temperature and degree of oxidation to avoid excessive embrittlement and failure of the fuel cladding. The criteria are mainly based on experimental data obtained in the 1970's - 80's.

A somewhat arbitrary classification of various types of tests could be the following:

1. Separate effects tests aimed at elucidating one phenomenon, e.g. mechanical tests.
2. Tests with moderate integration, tests in which the interaction between two or more phenomena is studied, e.g. thermal shock tests.
3. In-pile and out-of-pile semi-integral LOCA tests on single real or simulated fuel rods
4. In-pile or out-of-pile fuel bundles LOCA tests.

There have been essentially two different test methods (1. and 2 above) used in the past to assess the fuel clad embrittlement effect:

- *Mechanical tests*, Tests of samples that have been subjected to a high temperature oxidation of tubes in presence of steam. Tests used are e.g. ring compression tests (RCT), impact test<sup>49</sup>, three-point bending. The criterion here is brittle versus ductile fracture of the specimens. The ring compression tests, and brittleness criterion have the advantage of being consistent with the experimental basis of current LOCA criteria, and
- *Strength-based approach: axial loading during the quench phase* Tests with fuel claddings subjected to high temperature oxidation and subsequent quench with or without axial constraints. The criterion here is failure (clad rupture) versus survival of the fuel claddings during quenching. Using thermal quench test there is a need to assess the degree of axial constraints to be used in the test to be conservative.

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<sup>49</sup> The 0.3J impact criterion was based upon the recognition that cladding near ductile-brittle-boundary regime retains significant level of fracture toughness. The 0.3J impact energy corresponds to about 10 times of the impact energy that a burst and oxidised cladding needs to withstand to survive the thermal stresses generated during the quench phase of a LOCA transient (under no axial constraints). Also, 0.3J impact energy corresponds to about 10 times the impact energy generated during a seismic event.

It is not clear today what test should be used to ensure that the fuel retains “coolable geometry” during quenching and post-LOCA event. In Russia, in France and in Japan, the fuel rod LOCA behaviour is assessed in quenching tests (in Russia without clad restraint and in Japan or France with axial clad restraint). The criteria is that the fuel rod must not rupture during the quenching process. In the rest of the countries, the assessment is done by the ring compression test mentioned above. The ring compression test is more conservative (perhaps over-conservative) compared to the quenching tests. Thus, a material may pass the quench test but may not pass the ring compression test.

Comparing the two types of tests above, the former is more conservative (*perhaps too conservative*) (Figure 6-1).

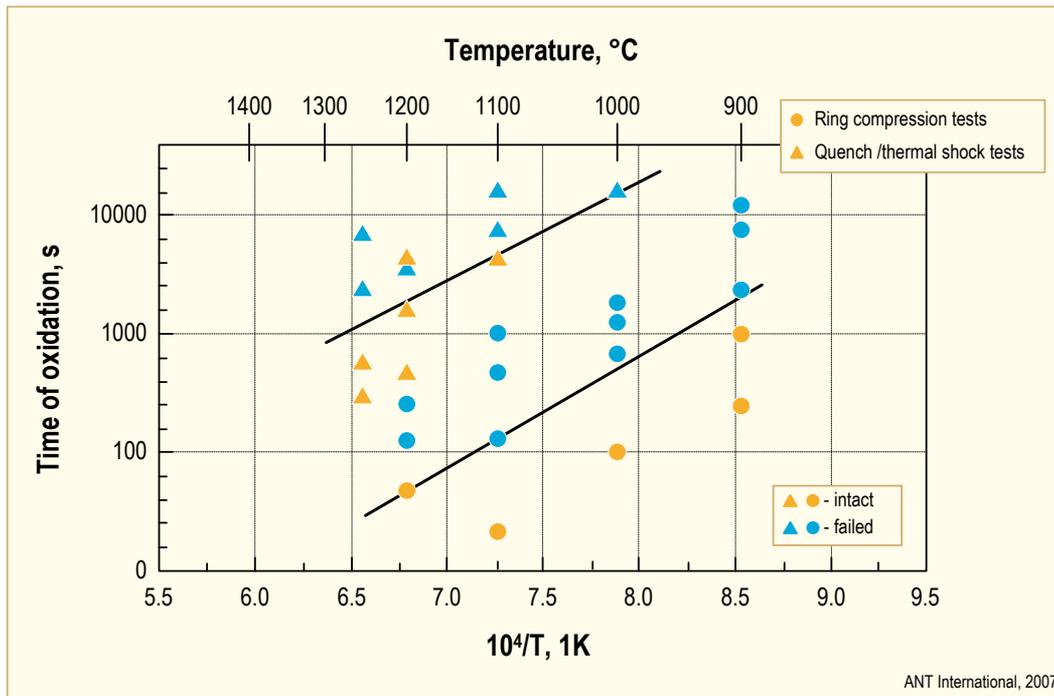


Figure 6-1: Cladding “failure” test data obtained in quench test and in RCTs on the same cladding material and in the same laboratory. As shown, the compression/brittleness tests give a far lower safety limit, typically ~250°C lower, after [Maroti, 2001].

When sufficiently embrittled, fuel cladding may fracture on cooling and break up into pieces thus losing a coolable geometry. The action of re-wetting involves the collapse of the vapour film and nucleate boiling commences. This event takes place at a more or less constant temperature, the Leidenfrost point. [Chung & Kassner, 1980], reported that rewetting occurs in the range 475-600°C that will change the heat transfer conditions inducing large thermal shock forces which may fracture the cladding if sufficiently embrittled by oxidation. Thus, during thermal shock, the maximum loading occurs in the range 475-600°C. At these temperatures and the cooling rates associated with quenching, hydrogen is in solution and has little effect on the fracture resistance of oxidised Zircaloy.

However, post-LOCA impact loading such as hydraulic, seismic and handling forces, occurs at lower temperatures, 100-200°C, where dissolved hydrogen or precipitated  $ZrH_{1.66}$  will deteriorate fuel cladding ductility much more than during the quenching phase.

During a *LOCA*, ballooning of the rod near the spacer grid, rod-grid spring chemical interaction and the friction between the fuel rod and spacer grids can restrict the axial movement of the cladding thus imposing clad tensile stresses during reflooding. In recognition of this, the AEC Staff wrote during the 1973 Rule-Making Hearing that “the loads due to assembly restraint and rod-to-rod interaction may not be small compared to the thermal shock load and cannot be neglected”. Subsequently, it was concluded that: “The staff believes that quench loads are likely the major loads, but the staff does not believe that the evidence is as yet conclusive enough to ignore all other loads (hydraulic, and seismic forces, and the forces related with handling and transportation)”.

Both in Russia, France and in Japan, the maximum allowable *ECR* during a *LOCA* transient is specified to ensure that the cladding can survive such a quenching without rupturing. In Russia a maximum value of 18 % *ECR* is used for Zr1Nb claddings assessed in quench tests without any axial constraints of the clad during quenching. In Japan a corresponding value of 15 % are used for Zircaloy claddings but assessed in quenching tests with significant axial constraints of the fuel clad.

Cold compression tests with non-prehydrided burst claddings (and high localised hydrogen concentrations) will fail the  $ECR < 17\%$  criterion. However, the Japanese thermal shock tests with non-pre-hydrided oxidised and burst cladding with some degree of axial restraints survives if  $ECR < 15\%$ . *The question is what type of test should be used and the selection depends on what loading situation during LOCA and post-LOCA is limiting and the degree of required conservatism in the criteria.*

In an excellent review by [Z. Hozer et al., 2023] all the different tests to study fuel behaviour during a *LOCA* are listed in Table 6-1 to Table 6-5.

Table 6-1: Phenomena resulting in fuel failure in *LOCA* tests [Z. Hozer et al., 2023].

PHENOMENA EXPERIMENTS	Fuel Failure during LOCA				
	Cladding Oxidation	Ballooning and Burst	Brittle Failure after Heavy Oxidation	Water Quench	Fuel Fragmentation and Dispersal
Edgar tests	X	X			
COCAGNE tests		X			
REBEKA tests	X	X			
AEKI/MTA EK burst tests	X	X	X		
JAERI and JAEA burst tests	X	X		X	
UK burst tests	X	X			
MRBT (ORNL) burst tests	X	X			
Russian burst tests	X	X			
ANL burst tests	X	X		X	X
EDF burst tests	X	X		X	
PBF tests	X	X		X	X
FR-2 tests	X	X			X
PHEBUS-LOCA test	X	X		X	
Halden LOCA tests	X	X			X
ACRR (SNL) tests	X				X
NRU MT-4 test	X	X		X	
LOFT LP-FP tests	X	X		X	X

FLASH tests (Grenoble, Siloe)	X	X		X	X
Studsvik LOCA test	X	X		X	X
CORA tests	X	X	X	X	
QUENCH-LOCA integral tests	X	X	X	X	
CODEX-LOCA integral tests	X	X	X	X	
PARAMETER tests	X	X		X	
MTA EK H uptake test					
DEFECT tests with defective fuel					
DEFEX secondary defect test Halden IFA-631 secondary degradation test					
OECD-IAEA Pairs Fuel Project	X	X	X	X	
ANT International, 2024					

Table 6-2: Activity release from fuel rods during LOCA events [Z. Hozer et al., 2023].

PHENOMENA	Activity Release from Fuel during LOCA				
	Noble Gas Release from the Fuel Rod- Steady State	Noble Gas Release from the Fuel Rod- Transient	Volatiles Fission Product Release from the Fuel Rod- Transient	Semi-Volatile Fission Product Release from the Fuel Rod- Transient	Fission Product Release from High Burnup Structure
Halden LOCA tests			X		
ACRR (SNL)			X	X	X
LOFT LP-FP tests		X	X	X	
FLASH tests (Grenoble, Siloe)		X	X		
GASPARD tests		X	X		X
HEVA tests		X	X	X	
VERCORS tests		X	X	X	X
VERDON tests		X	X	X	X
ITU FP tests		X	X	X	
ORNL FP tests		X	X	X	
CRL FP tests		X	X	X	
Halden FGR tests	X				X
FIRST-Nuclides leaching tests					
CRL defective					

fuel tests					
DEFECT tests with defective fuel					
DEFEX secondary defect test					
VVER NPP Iodine spiking					
PWR NPP Iodine spiking					
OECD-IAEA Paks Fuel Project		X	X	X	
ANT International, 2024					

Table 6-3: Activity transport from the reactor core to the environment during LOCA events [Z. Hozer et al., 2023].

PHENOMENA Activity Transport during LOCA									
EXPERIMENTS	Transport in the Primary Circuit (from Core to Break)	Deposition in the Primary Circuit. Retention by Primary Circuit Components	Transport in the Containment	Deposition on the Containment Wall	Deposition in the Containment Sump Water	Transport to the Environment Outside of Containment	Noble Gas Transport	Volatile Fission Product (I, Cs) Transport	Semi-Volatile Fission Product Transport
LOFT LP-FP tests	X	X					X	X	X
VERCORS tests	X	X					X	X	X
VERDON tests	X	X					X	X	X
CRL defective fuel tests									
DEFECT tests with defective fuel									
DEFEX secondary defect test									
BIP			X		X				
MARVIKEN FSCB	X	X	X		X	X		X	
THAI			X		X			X	
ARTIST									
STEM	X	X	X		X			X	X
Rivne NPP SG collector cover lift-up									
Rivne NPP non-closure of the pressurizer safety valve	X	X	X	X	X	X	X	X	X

SCTR at Doel NPP																		
OECD-IAEA Paks Fuel Project												X		X				

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Table 6-4: Characterisation of experimental programmes (part I.) [Z. Hozer et al., 2023].

TEST CHARACTERISATION	Type			Scale		Atmosphere					Tested Sample							
	Separate Effect Test	Integral Test	NPP measurement	Scaled Down	Full Scale	Steam	Inert Gas	Hydrogen	Oxidising	Reducing	Small Cladding Sample	Small Pellet Sample	Single Rod	Bundle	Non-Core Material	Irradiated	Non-Irradiated	
Edgar tests	X			X		X			X		X		X					X
COCAGNE tests	X			X			X				X		X	X				X
REBEKA tests	X			X		X	X		X		X		X	X				X
AEKI/MTA EK burst tests	X			X		X	X		X		X		X	X				X
JAERI and JAEA burst tests	X			X		X			X		X		X			X		X
UK burst tests	X			X		X			X		X		X	X		X		X
MRBT (ORNL) burst tests	X			X		X			X		X		X	X				X
Russian burst tests	X			X		X			X		X		X			X		X
ANL burst tests	X			X		X	X		X		X	X	X			X		X
EDF burst tests	X			X		X			X		X		X					X
PBF tests		X		X		X			X					X		X		X
FR-2 tests		X		X		X			X		X	X	X			X		X
PHEBUS-LOCA test		X		X		X			X		X	X		X				X
Halden LOCA tests		X		X		X			X				X			X		X
ACRR (SNL) tests		X		X		X	X	X		X					X	X		X
NRU MT-4 test		X		X		X			X					X				X
LOFT LP-FP tests		X		X		X			X					X		X		
FLASH tests (Grenoble, Siloe)		X		X		X	X		X				X			X		
GASPARD-MERARG tests	X			X		X	X						X			X		
HEVA tests	X			X		X		X					X			X		
VERCORS tests	X			X		X	X	X	X	X			X			X		
VERDON tests	X			X		X	X	X	X	X			X			X		
ITU FP tests	X			X			X		X		X					X		
ORNL FP tests	X			X			X	X	X				X			X		

LOSS OF COOLANT ACCIDENT, LOCA

CRL FP tests	X			X			X	X	X			X	X			X	
Studsvik LOCA test		X		X		X			X				X			X	
CORA tests	X	X		X		X	X		X				X	X			X
QUENCH-LOCA integral tests		X		X		X	X		X					X			X
CODEX-LOCA integral tests		X		X		X			X					X			X
PARAMETER tests		X		X		X	X		X					X			X
Halden FGR tests	X			X										X			X
FIRST-Nuclides leaching tests	X			X									X				
MTA EK H uptake test	X			X				X		X	X						
CRL defective fuel tests		X		X		X								X			X
DEFECT tests with defective fuel		X		X		X			X					X			X
DEFEX secondary defect test		X		X		X								X			X
Halden IFA-631 secondary degradation test	X			X		X	X	X						X			X
BIP	X	X		X					X							X	
MARVIKEN FSCB		X			X												
THAI		X		X												X	
ARTIST	X			X			X									X	
STEM	X															X	
VVER NPP iodine spiking			X		X									X		X	
PWR NPP iodine spiking			X		X									X		X	
Rivne NPP SG collector cover lift-up			X		X									X		X	
SGTR at Doel NPP			X		X									X		X	
Rivne NPP non-closure of the pressurizer safety valve			X		X									X		X	
OECD-IAEA Paks Fuel Project			X		X	X			X					X		X	

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Table 6-5: Characterisation of experimental programmes (part II.) [Z. Hozer et al., 2023].

TEST CHARACTERISATION	Cladding							Fuel			Heating Method					FP Release and Transport							
	Zircaloy-4	Zircaloy-2	Zirlo	M5	E110	Precharged with H	Pre-oxidation	UO <sub>2</sub>	MOX	High Burnup	Nuclear	Electric	Internal	Furnace	Induction	pH	Dose Rate	Temperature	Transport by Natural Circulation/gravity	Transport by Forced Flow	Transport in Gas Atmosphere	Transport in Liquids	Interfacial Mass Transfer
Edgar tests	X			X		X	X					X											
COCAGNE tests	X					X	X					X											
REBEKA tests	X				X							X											
AEKI/MTA EK burst tests	X				X		X					X		X									
JAERI and JAEA burst tests	X		X	X		X	X							X									
UK burst tests	X											X											
MRBT (ORNL) burst tests	X												X										
Russian burst tests					X							X		X									
ANL burst tests			X			X	X							X									
EDF burst tests	X					X								X									
PBF tests	X						X			X													
FR-2 tests	X						X			X													
PHEBUS-LOCA test	X						X			X													
Halden LOCA tests	X	X		X	X		X		X	X	X	X											
ACRR (SNL) tests	X						X		X	X													
NRU MT-4 test	X						X			X													
LOFT LP-FP tests	X						X			X							X	X		X	X	X	X
FLASH tests (Grenoble, Siloe)	X						X			X								X					
GASPARD-MERARG tests	X						X	X	X					X	X			X					
HEVA tests	X						X							X				X					
VERCORS tests	X			X			X	X	X					X	X			X					
VERDON tests	X			X			X	X	X					X	X			X					
ITU FP tests							X		X					X	X			X					
ORNL FP tests	X						X							X				X					
CRL FP tests	X						X							X				X					
Studsvik LOCA test			X				X		X					X									

## 7 Computer codes and methods for LOCA,

### 7.1 Calculation tools and validation process

Modelling Loss of Coolant Accidents (LOCAs) in nuclear power plants (NPPs) necessitates distinct yet comprehensive approaches tailored for both experimental testing and accident analysis. To simulate LOCA accidents, predictive computer codes are employed to effectively capture a number of phenomena associated in general and in particular with simulated scenarios. Essential to this process are reactor physics codes, fuel behaviour codes, and thermal hydraulics system codes. Among them, multi-rod transient codes hold particular significance due to their ability to assess interactions among fuel rods, which may have a significant impact on the possibility of verifying acceptance criteria designed to confirm the core coolability during and after the LOCA accident. It is essential for these codes to be well-documented, verified, and validated. Additionally, they should undergo uncertainty qualification and remain at the cutting edge of predictive capability as our understanding of the phenomena evolves.

For LOCA tests simulation and data elaboration, single-rod transient fuel behavior codes are mostly used. This is because such tests typically focus on the response of individual fuel rods (or specimens) to conditions like high temperature, internal pressure, fission gas transport or cladding ballooning and burst. Single-rod codes can model these phenomena in greater detail, aligning closely with experimental data. However, multi-rod codes are important for more comprehensive modelling, especially for fuel assembly-level phenomena such as rod interaction, coolant flow blockage, and collective ballooning effects. Still, it is important that the codes used for LOCA tests simulation are equally well documented, accurately verified and validated, and continuously updated to reflect advancements in our understanding of modelled phenomena.

To ensure accurate predictions for LOCAs, modelling codes must quantitatively capture key phenomena related to fuel behaviour during accidents. Knowledge of fuel behaviour under conditions corresponding to LOCA accident is crucial for the possibility of verifying the acceptance criteria for LOCA, which are generally associated with nuclear fuel. Thus, the main objectives include assessing fuel rod deformation, rupture risk, cladding temperatures, oxidation processes, and interactions with neighbouring rods. Codes for simulating the behaviour of nuclear fuel are being developed and used in almost all countries using nuclear power, however, there are countries that play a leading role in the development of such codes and models describing the state of the art in this field.

The computer codes used for such analyses are usually transient codes, but steady-state codes are also sometimes used, in particular when a functional connection through data between a steady-state code and a transient code exists. The connection and numerical integration of codes is a separate issue.

In the following sections, examples of the computer codes used for LOCA modelling and analysis will be listed and in brief described.

The development of specific computer codes for LOCA, given their detail, complexity and accuracy of the models, is determined by their application. The codes used to elaborate laboratory data from out-of-pile LOCA tests and other codes used for in-pile integral LOCA tests or these for a reactor safety analysis will have a different level of complexity as well as a slightly different level of validation. A code typically draws input information on coolant condition from a thermal-hydraulic code, and data on fuel from a steady-state code, while sometimes other auxiliary codes generate additional input for more specialized models.

It is generally recognized that in order to effectively model or simulate an integral LOCA test or a selected LOCA scenario, three primary types of code are required [cf. OECD, 2024]:

- Reactor physics codes.
- Fuel behaviour codes.
- Thermal hydraulics codes.

These codes to varying degrees can be integrated with each other or only communicate with each other (e.g. create a completed input with initial or boundary conditions to models in the other code).

These codes can operate either as stand-alone or as systems merged with each other. Such connection with internal data transfer may be permanent (hard-wired codes) or communicate only via data transfer controlled by suitable software, allowing initial or boundary conditions to be efficiently transferred between the codes. In the latter case, code execution can be simultaneous or sequential. This flexibility of integration increases the robustness and accuracy of simulations, facilitating a comprehensive understanding of simulated LOCA scenarios.

### 7.1.1 Reactor physics codes

The first group of codes belonging to these three distinguished types are reactor physics codes which usually provide the starting condition for transient analyses including LOCA events. These codes perform several key functions beyond just establishing starting conditions:

- **Neutron Flux Distribution**: They calculate the neutron flux distribution within the reactor core, which is essential for determining the power profile and heat generation rates.
- **Isotopic Composition**: They provide the isotopic composition of the nuclear fuel at various burnup levels. This data is crucial for understanding fuel behaviour during a LOCA, including reactivity changes, decay heat, and fission product release.
- **Temperature and Reactivity Feedback**: Reactor physics codes model feedback mechanisms such as Doppler broadening and moderator temperature effects, which influence the reactor's behaviour during transient conditions like a LOCA.
- **Burnup Calculations**: These codes calculate fuel burnup, which affects the mechanical properties of the fuel rods, the fission gas release, and the potential for fuel fragmentation. This data is used to assess the structural integrity of the fuel and cladding during LOCA conditions.
- **Decay Heat Generation**: Accurate modelling of decay heat generation over time is provided, which is critical for evaluating the heat removal capability of the ECCS and predicting the peak cladding temperature (PCT).

One of the most essential components of modern reactor physics codes are libraries of isotopes and nuclear parameters necessary for simulations of the state of a nuclear reactor core. These libraries contain extensive data on various isotopes, including:

- **Neutron Cross-Sections**: Essential for determining reaction probabilities, such as fission, capture, and scattering, which directly affect reactor core behaviour and fuel cycle calculations.
- **Decay Data**: Provides information on the half-lives and decay modes of radioactive isotopes, crucial for decay heat calculations and waste management.
- **Fission Yields**: Gives the distribution of fission products resulting from nuclear reactions, impacting fuel composition and safety assessments.
- **Thermal Properties**: Includes data on thermal conductivity, specific heat, and density for both fuel and cladding materials, necessary for heat transfer and temperature distribution modelling.

Modern nuclear data libraries were repeatedly verified on the most recent experimental and theoretical data. Libraries such as JEFF, ENDF, JENDL, and CENDL, ensures that reactor physics codes can accurately predict nuclear reactor behaviour under various conditions, including normal operation, transient scenarios, and accident conditions like a LOCA. Recent developments have included a more advanced graphics user-interface and the merger of reactor physics codes with for instance system thermal hydraulics codes.

The most well-known and widely used reactor physics codes are listed in the condensed table below along with their country of origin or institution responsible for their development. Additionally, a brief outline of the basic methods used by each reactor physics code is given. Selected and for the code representative references are listed beneath the table.

Table 7-1: Reactor physics codes

Reactor Physics Code/ Reference	Country of Origin / Developing Institution	Description and Basic Methods
MCNP (Monte Carlo N-Particle) [MCNP]	USA / Los Alamos National Lab. (LANL)	MCNP uses the Monte Carlo method for neutron, photon, electron, or coupled neutron/photon/electron transport. The code simulates the interaction of radiation with matter using a probability-based approach to solve the neutron transport equation. This involves random sampling of collision events and particle paths to statistically determine system behaviour. MCNP supports continuous-energy cross sections for high-fidelity reactor simulations.
SERPENT [Leppänen et al, 2015]	Finland / VTT Technical Research Centre of Finland	SERPENT is a multi-purpose 3D continuous-energy Monte Carlo reactor physics burnup calculation code. It combines high-fidelity neutron transport calculations with advanced burnup modelling to simulate the behaviour of nuclear fuel over time. It uses a unionized energy grid for neutron cross-sections and supports multi-physics coupling, which allows integration with thermal-hydraulics and fuel performance codes.
KENO Monte Carlo Transport [SCALE]	USA / Oak Ridge National Laboratory (ORNL)	KENO is part of the SCALE code system and uses a Monte Carlo-based method for criticality safety and reactor physics analysis. It focuses on neutron multiplication and reactivity calculations, employing a stochastic approach to estimate eigenvalues and flux distributions. KENO supports both continuous-energy and multigroup neutron cross sections, making it suitable for various nuclear applications, including criticality safety.
TRITON Reactor Physics [SCALE]	USA / Oak Ridge National Laboratory (ORNL)	TRITON, part of the SCALE code system, integrates 3D deterministic transport calculations with isotopic depletion and decay heat analysis. It uses the Monte Carlo method or deterministic methods like discrete ordinates for transport solutions. TRITON is used for high-fidelity depletion analysis, decay heat calculations, and source term estimation in nuclear reactor cores.
CASMO [Knott et al, 2002]	Sweden / Studsvik Scandpower	CASMO is a multi-group, two-dimensional transport theory code that uses the method of characteristics (MOC) or discrete ordinates for neutron transport calculations in fuel assemblies. It performs detailed isotopic depletion analysis and provides accurate cross-section libraries for core simulators. CASMO is widely used in lattice physics for reactor design, fuel management, and safety analysis, specifically for light-water reactors (LWRs).
WIMS [Halls et al, 2000]	UK / National Nuclear Laboratory (NNL)	WIMS (Winfrith Improved Multigroup Scheme) is a deterministic lattice code that uses multigroup neutron diffusion or transport methods to perform reactor physics calculations. It provides a broad range of multigroup cross-section libraries and solves neutron transport equations for different geometries. WIMS is used for fuel cycle studies, criticality safety, and core analysis in various reactor types, including LWRs and gas-cooled reactors.
PARCS	USA / Purdue University and US NRC	PARCS (Purdue Advanced Reactor Core Simulator) is a three-dimensional nodal reactor core simulator that solves the two-group neutron diffusion or simplified transport equation. It is often

[Downar et al, 2009]		coupled with thermal-hydraulic codes like RELAP5 for transient and safety analysis, allowing for detailed multi-physics simulations. PARCS is used for core monitoring, operational support, and regulatory analysis for both PWRs and BWRs.
APOLLO [Sanchez et al, 2010]	France / CEA (French Alternative Energies and Atomic Energy Commission)	APOLLO is a deterministic neutron transport code that uses the method of characteristics and collision probability methods to solve the Boltzmann transport equation. It is designed for high-fidelity reactor core calculations, including fuel cycle studies, isotopic depletion, and 3D core modelling. APOLLO is widely used for pressurized water reactors (PWRs) and supports coupling with thermal-hydraulic codes.
POLCA [Lipiec et al. (2015)]	Sweden / Westinghouse Electric Sweden AB	POLCA is a nodal core simulator specifically developed for boiling water reactor (BWR) applications. It uses nodal diffusion theory to solve neutron transport equations and models both steady-state and transient conditions. POLCA is used for core design, safety analysis, and operational support, focusing on BWRs' unique characteristics and operational requirements.
PHISICS [PHISICS]	USA / Idaho National Laboratory (INL)	PHISICS (Parallel High-fidelity Innovative Simulation of Core Systems) is a flexible and modular reactor physics code system designed for high-performance computing. It uses advanced neutronics methods, including discrete ordinates and Monte Carlo simulations, to provide detailed core modelling. PHISICS is used for multi-physics applications and is capable of coupling with thermal-hydraulics and fuel performance codes.
BIPR-7 [Kravtsov et al. 2004] KASKAD [Lazarenko et al. 2007]	Russia / Kurchatov Institute	BIPR-7 is a reactor physics code developed specifically for the analysis of VVER reactors, which use a hexagonal lattice. The code uses nodal diffusion methods to model neutron flux distribution and reactivity. BIPR-7 is designed for core design, fuel management, and safety analysis of VVER reactors, providing solutions for steady-state and transient calculations, and integrating with thermal-hydraulic and safety analysis tools. [11]. BIPR-7 is included in the Kurchatov's software package KASKAD [12].
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**Remark:** The publications or website links given for each code in the first column do not exhaust the available publications, but are only a collection of relevant starting points for your own research.

## 7.1.2 Fuel performance codes

Fuel behaviour codes represent a most numerous and critical category of tools employed in LOCA analyses, encompassing a wide variety of simulations (conditions, scenarios). These codes are typically categorized into two primary types: steady-state codes and transient codes, each serving distinct purposes in understanding the performance of nuclear fuel under normal and accident conditions.

Steady-state codes focus on characterizing the normal operational behaviours of nuclear fuel over extended periods. They calculate core parameters such as temperature distribution, thermal conductivity, and the physical and chemical states of fuel. These codes simulate the steady-state conditions that are typically found during normal reactor operations, thereby establishing the baseline characteristics of the fuel assembly. They often examine the interactions between different materials, such as fuel pellets, cladding, and coolant, ensuring that all components perform adequately under expected normal operating conditions. The output from steady-state codes frequently provides the essential starting conditions and initial parameters for the transient codes that will simulate potential accident scenarios.

In contrast, transient codes are designed to analyse the reactor's behaviour during short-term events (transients), including accidents such as a LOCA, and some are numerically prepared to simulate longer transients. Transient codes simulate the dynamic processes occurring after a disturbance, providing insights into how the system evolves over time. They are capable of modelling a wide range of phenomena, including heat transfer, fluid dynamics, and the thermal-hydraulic responses of the reactor. Transient codes may also implement various modelling options, allowing users to conduct both conservative analyses - where conditions are intentionally exaggerated to ensure safety margins - and best estimate calculations, which aim to represent the actual behaviour of the system as accurately as possible. By capturing the transient response of the fuel and overall coolant system during an accident, these codes help to assess potential safety risks and the likelihood of fuel damage.

Both steady-state and transient codes are often tailored to specific reactor or/and fuel designs and regulatory requirements, making them particularly useful tools for both industry professionals and regulatory bodies.

Steady-state fuel behaviour codes model the thermal, mechanical, and sometimes chemical behaviour of nuclear fuel under normal operating conditions over time. These codes perform several key functions beyond just establishing starting conditions:

- **Fuel Temperature Distribution**: Predicts the radial and axial temperature profiles within the fuel pellet and cladding, crucial for understanding heat transfer during steady-state operations.
- **Fission Gas Release (FGR)**: Estimates the amount of fission gas released into the fuel rod plenum. The buildup of fission gas affects internal rod pressure, which can impact the cladding's integrity during a LOCA.
- **Cladding Oxidation and Hydriding**: Calculates the thickness of the oxide layer and the extent of hydrogen pick-up and sometimes its structure, which influence cladding embrittlement and its behaviour under high-temperature conditions in a LOCA.
- **Fuel Pellet Swelling and Creep**: Models fuel pellet swelling and the creep behaviour of both the fuel and cladding. This is important to assess the gap closure between the pellet and cladding, which affects heat conduction.
- **Thermal Conductivity Degradation**: Accounts for changes in thermal conductivity of both the fuel and cladding materials as a function of burnup, which impacts the ability of the fuel to conduct heat away from the rod centerline to the coolant.
- **Burnup-Dependent Material Properties**: Provides properties such as Young's modulus, thermal expansion, and specific heat, which change with irradiation and affect fuel performance during a transient.

Steady-state fuel behaviour codes provide several parameters that constitute essential inputs for transient analysis codes used in LOCA events modelling. Among them are:

- **Initial Temperature Profiles**: Accurate temperature distributions within the fuel rod are necessary to model the thermal response during the transient.
- **Internal Pressure**: The pre-existing pressure inside the fuel rod, influenced by fission gas release, is crucial for modelling cladding deformation and burst potential.
- **Cladding Condition**: Information on the oxide layer thickness and hydriding levels helps predict the cladding's mechanical response during a LOCA.
- **Fuel Fragmentation Data**: Burnup-dependent data on potential fuel fragmentation provide insights into how the fuel might behave under LOCA conditions, influencing the potential for fine fragmentation and dispersal.

Some of the more advanced steady-state fuel behaviour codes have implemented modules to verify acceptance criteria for LOCA such as those specified in the US NRC’s 10 CFR 50.46c and other international standards, i.e. how steady-state temperature profiles impact the calculation of PCT during LOCA, the extent of oxidation calculated by steady-state codes contributes to assessing the ECR or information on cladding deformation and burst risk, along with fuel fragmentation, which helps in verifying that the coolability of the reactor core is maintained during and after a LOCA.

A limited number of widely spread steady-state fuel behaviour codes useful for LOCA analysis is listed in the table below. In addition to information about the country or institution where the code was originated, the third column contains summary information about basic methodology, type of calculated geometry, and LOCA criteria verification possibilities and finally column four tells whether the code has an interface to a transient code.

Table 7-2: Steady-State Fuel Behaviour Codes

Code Name/ References	Country/ Institution of Origin	Basic Methodology, Description, Geometry Type, and LOCA Criteria Verification	Interface to Transient Codes
<p><b>FRAPCON</b> [Code overview] [Lanning et al, 1997] [Porter, 2017]</p>	<p>USA / Pacific Northwest National Laboratory (PNNL)</p>	<p>FRAPCON is a steady-state fuel performance code designed for light water reactors (LWRs). It uses finite difference methods to solve the heat conduction equation in the fuel rod, and empirical models for cladding oxidation, creep, and hydriding, among others. FRAPCON supports single-rod geometry, modelling the thermal, mechanical, and fission gas behaviour over long periods. It includes modules to verify LOCA-specific acceptance criteria such as peak cladding temperature (PCT), cladding oxidation (ECR), and hydrogen pickup in the cladding. These capabilities are crucial for assessing whether a fuel rod will maintain integrity under both normal and accident conditions, ensuring compliance with safety standards like 10 CFR 50.46c. Additional features include models for fuel densification, swelling, and fission gas release behaviour.</p>	<p>Yes: Interfaces with FRAPTRAN, which is used to simulate transient fuel rod behaviour during accidents such as LOCA and RIA. FRAPTRAN extends FRAPCON's steady-state predictions into transient scenarios, using the steady-state outputs as initial conditions.</p>
<p><b>TESPA-ROD</b> [GRS, 2024] [Sonnenburg, Boldt, 2018] [Sonnenburg, 2018]</p>	<p>Germany / Gesellschaft für Anlagen- und Reaktorsicherheit (GRS)</p>	<p>TESPA-ROD is a fuel performance code that provides detailed modelling of steady-state and transient behaviour of fuel rods in LWRs. It combines empirical and mechanistic models to simulate thermal conductivity degradation, cladding creep, oxidation, and hydriding. TESPA-ROD focuses on single-rod analysis but can be extended to multi-rod geometries under certain conditions. The code allows analysing the fuel rod behaviour under various accident conditions, normal operation as well as long-term storage condition. In particular the accident conditions refer to both LOCA and RIA. The code includes modules to evaluate LOCA-specific acceptance criteria such as PCT, maximum allowable cladding strain, and stress, which are essential for ensuring the integrity of the fuel rod during LOCA events. It also includes models for evaluating fuel fragmentation and relocation, contributing to a comprehensive safety assessment. Cladding failure can be determined from stress/strain relations who are specific to</p>	<p>Yes: TESPA-ROD can interface with various thermal-hydraulic codes, like ATHLET (GRS) providing a detailed evaluation of fuel rod behaviour during transients. These interfaces enhance the accuracy of predictions for accident conditions by allowing for integrated thermal-hydraulic and mechanical assessments.</p>

		the cladding materials Duplex, Zry-2, Zry-4, E110, M5 or Zirlo. Various fuels types (UO <sub>2</sub> , MOX, and Gadolinium doped fuel) at all burn-up levels can be analysed.	
<p><b>START-3</b> [START-3A software] [Medvedev et al, 1999] [Bibilashvili et al, 2000] [Khvostov et al, 2005] <b>[Khvostov et al, 2004]</b></p>	<p>Russia/ All-Russian Scientific Research Institute for Inorganic Materials (VNIINM)</p>	<p>START-3 is a Russian fuel performance code specifically designed for hexagonal fuel assemblies used in VVER reactors. The code employs finite difference and finite element methods to model thermal and mechanical behaviour under steady-state conditions. It focuses on single-rod geometry within hexagonal fuel assemblies. START-3 includes specific modules for LOCA analysis, such as the calculation of PCT, cladding oxidation (ECR), and prediction of cladding ballooning and burst risk, particularly under high-burnup conditions. The code is validated against Russian experimental data, making it highly relevant for VVER safety assessments, including LOCA scenarios where high-burnup fuel behaviour needs to be accurately predicted.</p>	<p>Yes: START-3 interfaces with Russian thermal-hydraulic codes such as RAPTA5 or DINAMIKA used for VVER reactor safety analysis, particularly under LOCA conditions.</p>
<p><b>ENIGMA</b> [Jackson et al, 1988] [Jackson et al, 1990] [Worrall, 2009] [Rossiter, 2011] [Rossiter, 2013] [Peakman et al, 2022] [Rossiter, Peakman, 2024]</p>	<p>United Kingdom / National Nuclear Laboratory (NNL), initially developed by BNFL and British Energy, to perform fuel safety analyses in support of the Sizewell-B PWR.</p>	<p>ENIGMA is a fuel performance code that models both steady-state and transient behaviour in LWRs and AGR systems. The code utilizes finite element methods to solve for thermal and mechanical behavior of fuel rods, including effects like pellet-clad interaction, cladding deformation, and fission gas release. ENIGMA primarily performs calculations in single-rod geometry, providing flexibility in modelling various reactor configurations, but can also analyse multiple rods independently and the results can be integrated into broader system-level analyses where necessary. It can also verify LOCA-specific acceptance criteria such as PCT, ECR, and cladding strain. The ENIGMA code primarily models pellet fragmentation, cladding creep, and deformation under operational and accident conditions, but it does not explicitly model FFRD in its full scope.</p>	<p>Yes: ENIGMA can interface with NNL's NEXUS code and Studsvik's CMS codes (Core Management System) allowing for enhanced assessment of reactor safety margins by combining detailed fuel performance with system-level thermal-hydraulic behavior.</p>
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**Remark:** The publications or website links given for each code in the first column do not exhaust the available publications but are only a collection of relevant starting points for your own research.

The transient fuel behaviour codes are used to simulate the response of nuclear fuel to rapid changes in operating conditions, such as during a LOCA. These codes perform several critical calculations to assess the safety and integrity of the fuel under transient conditions:

- **Cladding Temperature and Stress:** Modelling the temperature distribution and mechanical stress in the cladding during rapid temperature changes to predict the risk of cladding failure or burst.
- **Fuel Temperature and Heat Transfer:** Computing the changes in fuel pellet temperature and heat transfer between the fuel and the cladding, which is vital for evaluating the PCT during a LOCA.

- **Cladding Oxidation and Embrittlement:** Estimating the extent of oxidation (formation of oxide layers) and embrittlement of the cladding under high-temperature steam conditions, which can impact the equivalent cladding reacted (ECR) metric.
- **Transient Fission Gas Release and Rod Internal Pressure:** Predicting the release of fission gases from the fuel pellets during simulated transient and the resulting internal rod pressure, influencing the risk of cladding deformation and rupture.
- **Fuel Fragmentation, Relocation, and Dispersal (FFRD):** Evaluating the extent of fuel fragmentation and potential axial relocation as susceptibility for dispersal of fuel fragments (i.e., fine fragments) in the event of a cladding rupture, which can affect cooling channel blockage and heat removal.

The outcomes from transient fuel behaviour codes, including PCT, ECR, and cladding burst probability, are used to verify compliance with regulatory acceptance criteria (such as the US NRC’s 10 CFR 50.46c or equivalent criteria in other countries). This verification process ensures that the reactor core remains coolable, and the fuel integrity is maintained under LOCA conditions. Crucial in such verification process is treatment of uncertainties in transient fuel behaviour codes for expected conservative verification of acceptance criteria for several reasons:

- **Ensures Safety Margins:** Uncertainties in input parameters (like material properties, thermal-hydraulic conditions, and reactor state) must be considered to ensure that safety margins are maintained to protect the reactor core and prevent fuel damage during a LOCA.
- **Regulatory Compliance:** Regulatory frameworks (like the US NRC's 10 CFR 50.46c) require that safety analyses include uncertainty assessments to demonstrate compliance with acceptance criteria such as PCT and ECR.
- **Enhances Reliability of Predictions:** Proper treatment of uncertainties helps improve the reliability of model predictions, ensuring that calculated outcomes, such as cladding temperature or pressure, remain within safe limits even under worst-case scenarios.
- **Supports Decision-Making:** By understanding the range of possible outcomes, operators and regulators can make more informed decisions about reactor safety and necessary design or operational adjustments.

Again, a limited number of widely spread transient fuel behaviour codes useful for LOCA analysis are listed in the table below. In addition to information about the country or institution where the code was originated, a summary information about basic methodology, type of calculated geometry, and LOCA criteria verification possibilities as well as information concerning interface is given.

Table 7-3: Transient Fuel Behaviour Codes

Code Name	Country/ Institution of Origin	Basic Methodology, Description, Geometry Type, and LOCA Criteria Verification	Interface to Transient Codes
<b>FRAPTRAN</b> [Cunningham et al] [Geelhood et al, 2016]	USA / Pacific Northwest National Laboratory (PNNL)	FRAPTRAN is a transient fuel performance code developed for analysing the behavior of nuclear fuel rods during accidents, including LOCA and RIA. The code uses finite difference methods in the 1.5D rod model to solve the heat conduction, structural, and mechanical equations for the fuel rod. It models transient phenomena such as heat transfer and rod temperature transient, fission gas release and rod gas pressure, elastic-plastic fuel and cladding deformation, burst, cladding oxidation, and hydriding. FRAPTRAN is primarily used for single-rod geometry but can handle complex transients including power ramps and LOCA conditions. It can be used to verify LOCA-specific criteria, including PCT, oxidation levels (ECR), cladding rupture, and embrittlement. FRAPTRAN includes detailed material properties and empirical models validated with	Yes: Directly interfaces with FRAPCON (steady-state code) for burnup parameters and TRACE (thermal-hydraulic code) for system-level transient analysis. This integration enhances the ability to predict fuel behavior under LOCA conditions by combining detailed rod-level and system-level models.

## 8 LOCA SMR relevant information

### 8.1 Introduction

Near-term WC-SMRs represent a potential evolution in nuclear power generation, offering a range of advantages over traditional large-scale reactors, including enhanced safety, improved economics, and increased flexibility. SMRs are characterized by modularity, with major components manufactured in a factory and transported to the site for assembly. Many of the designs are still in the basic or conceptual design phase with limited reliable technical details about their performance. Basic information about SMR designs is summarized in Section 3.2.4, more details about leading western SMR concepts is summarized below.

### 8.2 VOYGR (NuScale)

The NuScale VOYGR power plant is built on a foundation of inherent and passive safety features, minimizing reliance on active systems and operator intervention. This philosophy is central to its LOCA response strategy. It utilizes a multi-module approach, with each module containing a 77 MWe PWR. A key feature is the integration of the reactor core, steam generators, and pressurizer within a single vessel i.e. so called integral SMR. The reactor vessel is submerged in a large pool of water within a containment structure. This configuration provides a significant heat sink.

In the event of a LOCA, the reactor automatically shuts down, and decay heat is removed primarily through natural circulation. The density difference between hot and cold water drives the coolant flow, requiring no pumps or external power. The large pool of water surrounding the reactor vessel acts as a massive heat sink, absorbing decay heat and preventing core damage. This passive cooling capability allows for long-term heat removal without operator intervention. Steam generators integrated in the vessel transfer heat from the reactor coolant to a secondary loop, producing steam for electricity generation in nominal operation and in LOCA, they continue to function passively, removing heat from the primary system. In addition Passive Residual Heat Removal System removes decay heat from the containment pool, transferring it to the atmosphere through natural convection and condensation. NuScale has conducted extensive analyses and simulations to evaluate VOYGR's response to a wide range of LOCA scenarios. Analyses cover various break sizes and locations, including breaks in the reactor coolant system piping, steam generator tubes, and the reactor vessel itself. System codes, such as RELAP5-3D and GOTHIC, are used to simulate LOCA events and predict plant response. NuScale has also conducted experiments to validate the performance of its passive safety systems.

Analyses and experiments have demonstrated the effectiveness of NuScale's passive safety systems in mitigating LOCA consequences. The plant can passively cool the reactor core for an indefinite period without operator intervention, even in the event of a large-break LOCA. The containment structure is designed to withstand pressure and temperature increases associated with a LOCA, preventing the release of radioactive materials.

### 8.3 BWRX-300 (GE Hitachi)

The BWRX-300 is a 300 MWe BWR that leverages GE Hitachi's experience with different variants of BWR technologies. It features a simplified design that reduces the number of components and systems compared to traditional BWRs. It utilizes natural circulation for core cooling. The BWRX-300 builds upon the design of the ESBWR and two older BWR designs built and operated in Switzerland and Netherlands.

The system employs a combination of passive and active safety systems to mitigate the consequences of a LOCA. The reactor core is positioned below the water level, ensuring that it remains submerged even with a loss of pressure. This prevents core uncover and damage. In the event of a LOCA, the reactor automatically shuts down, and natural circulation removes decay heat from the core. Isolation condensers are passive safety systems that condense steam and return condensate to the reactor vessel, maintaining water level and cooling capacity. The isolation condensers operate by natural circulation, requiring no external power.

While emphasizing passive safety, the BWRX-300 also incorporates other safety systems to provide backup cooling capabilities. The ECCS is an active system that injects water into the reactor vessel in the event of a LOCA. It provides backup cooling in case of passive system failure. Reactor Core Isolation Cooling provides cooling to the reactor core during shutdown conditions, including LOCA events. It utilizes a steam-driven turbine to power a pump that circulates water through the core. And finally, Standby Liquid Control System is used to inject neutron-absorbing borated water into the reactor core to shut down the nuclear reaction in emergency situations.

The BWRX-300 features a dry containment structure designed to withstand the pressure and temperature increases associated with a LOCA. This prevents the release of radioactive materials to the environment. In addition to the containment, systems like the suppression pool and the standby gas treatment system help to filter and remove radioactive fission products that may be released from the fuel during a LOCA.

## 8.4 Rolls-Royce SMR

The Rolls-Royce SMR is a 470 MWe PWR designed with a strong emphasis on modularity and factory fabrication. It is built from a series of factory-made modules that are transported to the site and assembled. The design incorporates passive safety systems that rely on natural phenomena for operation, minimizing the need for active components and operator intervention.

The Rolls-Royce SMR is currently undergoing the Generic Design Assessment process in the UK. This rigorous assessment evaluates the safety, security, and environmental aspects of the design. Rolls-Royce is also working with international regulators to ensure the SMR meets global safety standards.

By combining inherent safety features, passive safety systems, and engineered safety systems, the Rolls-Royce SMR is designed to effectively mitigate the consequences of a LOCA and ensure public safety. Continued research, development, and engagement with regulators will further validate its performance and contribute to the advancement of nuclear safety.

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## Appendix A - Reactor characteristics (Charles Patterson)

There are essentially five different types of commercial water-cooled reactors. By design, the water-cooled reactors in operation can be separated under thermodynamic aspects into Closed Cycle Systems (PWR, VVER, and PHWR) and Open Cycle Systems (BWR and RBMK) [Riess & Millet, 1994].

The separation into “Closed” and “Open Systems” has consequences when it comes to the interaction between radiation and the coolant. In a “Closed” system, the net decomposition of the coolant can be suppressed by the addition of hydrogen or ammonia, whereas in “Open Systems” volatile radiolysis species are produced and removed from the coolant while non-volatile species remain in the coolant.

The primary coolant in Closed Cycle System reactors operates under alkaline and reducing conditions with LiOH or KOH as the agents of pH control. This basic principle has been used for almost 50 years and has only been modified within the established framework of specified values for pH control. Such modifications are called “coordinated”, “modified”, “elevated lithium”, or “coordinated at elevated pH” Li/B-chemistries.

The primary coolant in Open Cycle System reactors has historically operated under neutral and oxygenated conditions while keeping the coolant water as clean as possible. However, based on material integrity concerns, the coolant chemistry conditions have been modified in most reactors to reducing conditions by injecting hydrogen in to feedwater (Hydrogen Water Chemistry (HWC), Noble Metal Chemical Addition (NMCA) and Online Noble Chemistry (OLNC). The latter two technologies involve injecting a solution containing noble metals (Pt +Rh) in the case of NMCA and Pt in the case of OLNC.

The fuel in water reactors operates in a wide range of environments. Typical operating conditions for varying reactor types are shown in Table A-1. Additional information on the thermal-hydraulic conditions in BWR and PWR cores is given in Figure A-1. For reference, the number and relative generating capacities of each type of operating plant in Figure A-2. The figure shows similar data for reactors under construction, Figure A-3.

Table A-1: Design and operating conditions for water cooled reactors.

Parameter	Western type PWR	VVER <sup>4</sup> (440/1000) MW	CANDU/ PHWR <sup>1</sup>	BWR	RBMK <sup>2</sup>
Coolant	Pressurized H <sub>2</sub> O	Pressurized H <sub>2</sub> O	Pressurized D <sub>2</sub> O	Boiling H <sub>2</sub> O	Boiling H <sub>2</sub> O
Fuel Materials (Pressure tube materials)	Zry-4, Optimized ZIRLO <sup>5</sup> , DUPLEX <sup>6</sup> , M5, Inconel, SS <sup>3</sup>	E110, E635	Zry-4 (Zr2.5Nb)	Zry-2, Zry-4, Inconel, SS	Zr-alloy E110, (Zr2.5Nb)
Average power rating, (MW/m <sup>3</sup> )	80–125	83/108	9–19	40–57	5
Fast Neutron Flux, Average, n/m <sup>2</sup> •s (E>1MeV)	6–9E17	5-7E17	1.6-4.3E17	4–7E17	1–2E17
Temperatures, °C					
Average Coolant inlet	279–294	267/290	249–257	200–235 (FW)	270
Average Coolant outlet	313–329	298/320	293–305	280–288	284
Max Cladding (outside surface)	320–350	335/352	330	285–305	290
Steam mass content, %				7–14	14
System pressure, MPa	15.5–15.8	12.5/16.5	10-11	7.0	6.7
Coolant Velocity, m/s	3–6*	3.5/6	3–5	2–5*	3.7
Coolant Chemistry**					
Oxygen, ppb	<5	<10	<10	200-300 (NWC <sup>7</sup> )	<20
Hydrogen, ppm	1.5–4	2.6–5.3	0.3–1 <sup>8</sup>	FW (H <sub>2</sub> ) 1-2ppm (HWC without noble metals)	-
STP cc/kg	17–50	30–60	3 to 10 <sup>8</sup>	FW (H <sub>2</sub> ) 0.25-0.35ppm (HWC with noble metals)	
Boron (as H <sub>3</sub> BO <sub>3</sub> ), ppm	0–2200	0–1500	–***	–	–
Li (as LiOH), ppm	0.2–6.0	0.05–0.5	0.35-1.4	–	–
K (as KOH), ppm	–	2–20	–	–	–
NH <sub>3</sub> , ppm	–	5–25	–	–	–
NaOH, ppm	–	0.03–0.35	–	–	–
* Variation from lower to upper part of the core and from plant to plant.					
** Zn in ppb quantities may be added to BWRs and PWRs; Pt in ppb quantities may be added to NMCA BWRs and ppt quantities to OLNK BWRs					
*** Not in coolant but in moderator					
1. Canadian Deuterium Uranium [CANDU 6 Program Team, 2005]; Pressurised Heavy Water Reactor (PHWR), 2. Reaktor Bolshoi Mozhnosti Kanalov (RBMK), 3. Stainless Steel (SS), 4. Voda Voda Energo Reactor (VVER),			5. Optimized Zirconium Low Oxidation (Optimized ZIRLO), 6. Cladding tube consisting of an outer soft layer and inner layer with high strength - normally high Sn Zry-4, 7. Normal Water Chemistry, 8. Deuterium		

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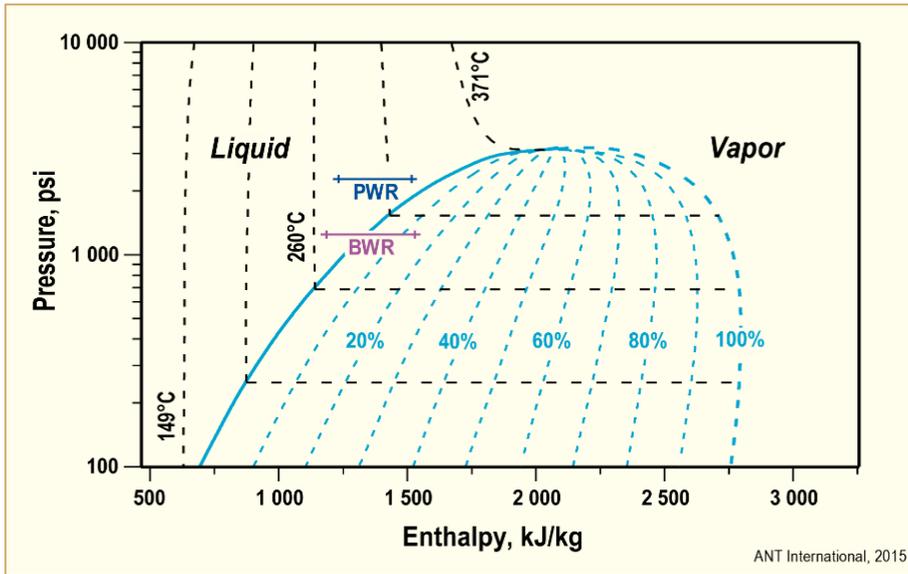


Figure A-1: Typical thermal-hydraulic conditions in BWRs and PWRs.

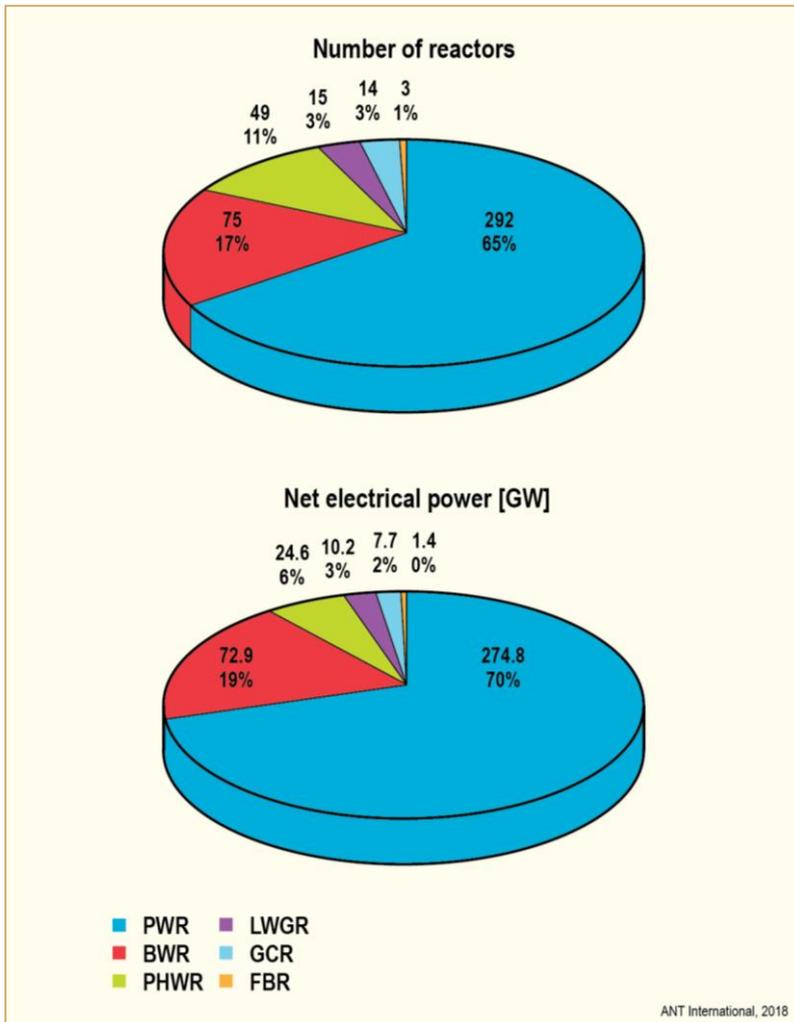


Figure A-2: Operating power plants. PHWR – Pressurized Heavy Water Reactor -same as CANDU; LWGR – Light Water Graphite Moderated Reactor – identical to RBMK; GCR – Gas Cooled Reactor; FBR – Fast Breeder Reactor; after [IAEA, 2018a].

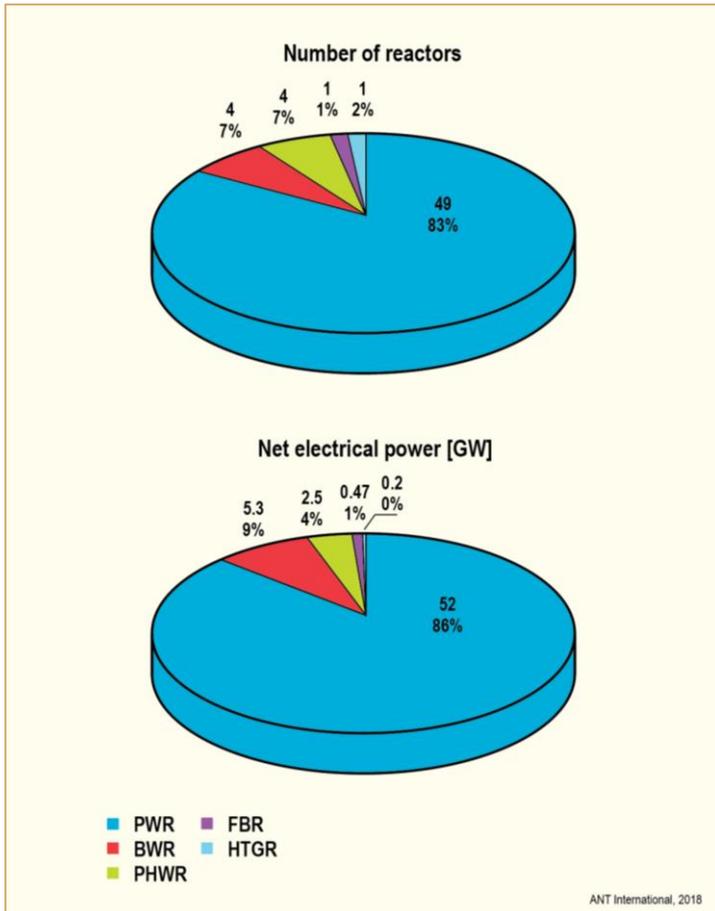


Figure A-3: Power plants under construction. PHWR – Pressurized Heavy Water Reactor -same as CANDU; FBR – Fast Breeder Reactor; HTGR- High Temperature Gas-cooled Reactor<sup>50</sup> after [IAEA, 2018a].

<sup>50</sup> High-temperature gas-cooled reactor (HTGR), is a Generation IV reactor concept that uses a graphite-moderated nuclear reactor with a once-through uranium fuel cycle [Wikipedia, 2018].

## Appendix B - Fuel assembly designs (Peter Rudling)

### B.1 BWR fuel design

The fuel rod array for BWRs was initially 6×6 but the 7×7 lattice configuration was adopted relatively early and there has been a trend over the subsequent years to increase the number of rods so that now most fuel assemblies are either of 9×9, 10×10 or 11×11 square configuration designs. An example of a BWR fuel assembly based on a 10×10 lattice is shown in Figure B-1. The driving force in the trend toward larger number of fuel rods in each fuel assembly was the reduction of Linear Heat Generation Rate (LHGR). Such changes produced a number of fuel performance benefits, such as lower Fission Gas Release (FGR) and increased margin to the Pellet Cladding Interaction (PCI) failure threshold, which is discussed later. However, to decrease fuel cycle costs, the LHGRs of 9×9, 10×10 and 11×11 fuel assemblies have successively been increased so that peak heat generation rates are today almost comparable to those of the older 7×7 and 8×8 designs.

In all BWRs, fuel bundles are enclosed in open-ended, square tubes or “channels”. The combination of a fuel bundle (rods, spacers, tie plates and related hardware) and a channel is commonly identified as a fuel bundle or, more generically, as a fuel assembly. The channels are ducts for coolant flow that prevent steam induced lateral flow of coolant among the assemblies operating at different power levels. They also form the inter-assembly space into which control blades are inserted.

Fuel assemblies are positioned in a reactor core support structures at the bottom and top of each assembly. Lower tie or bottom support plates, Figure B-1, are supported by a pedestal that carries the assembly weight and directs coolant into the channel. The upper ends of fuel assemblies are supported laterally by a rectangular lattice structure which bears against wear pads on the outer surface of the fuel channels. Cruciform-shaped control blades are located in the region between the channels of selected sets of four fuel assemblies. Part of the incoming coolant is diverted into the region among fuel assemblies to transfer heat from the control blades and prevent boiling.

Irrespective of the many possible different shapes, sizes and configurations, the common fuel assembly design requirements are:

- Maintain proper positioning of the fuel rods under normal operating conditions and in Design Basis Accidents (DBAs); i.e., maintain a controllable and coolable (safe) geometry during normal operation, anticipated operating occurrences and during postulated seismic events as well as LOCA, RIA and (more recently) beyond design-bases accidents.
- Contain or at least confine fuel materials and fission products.
- Facilitate handling before and after irradiation.

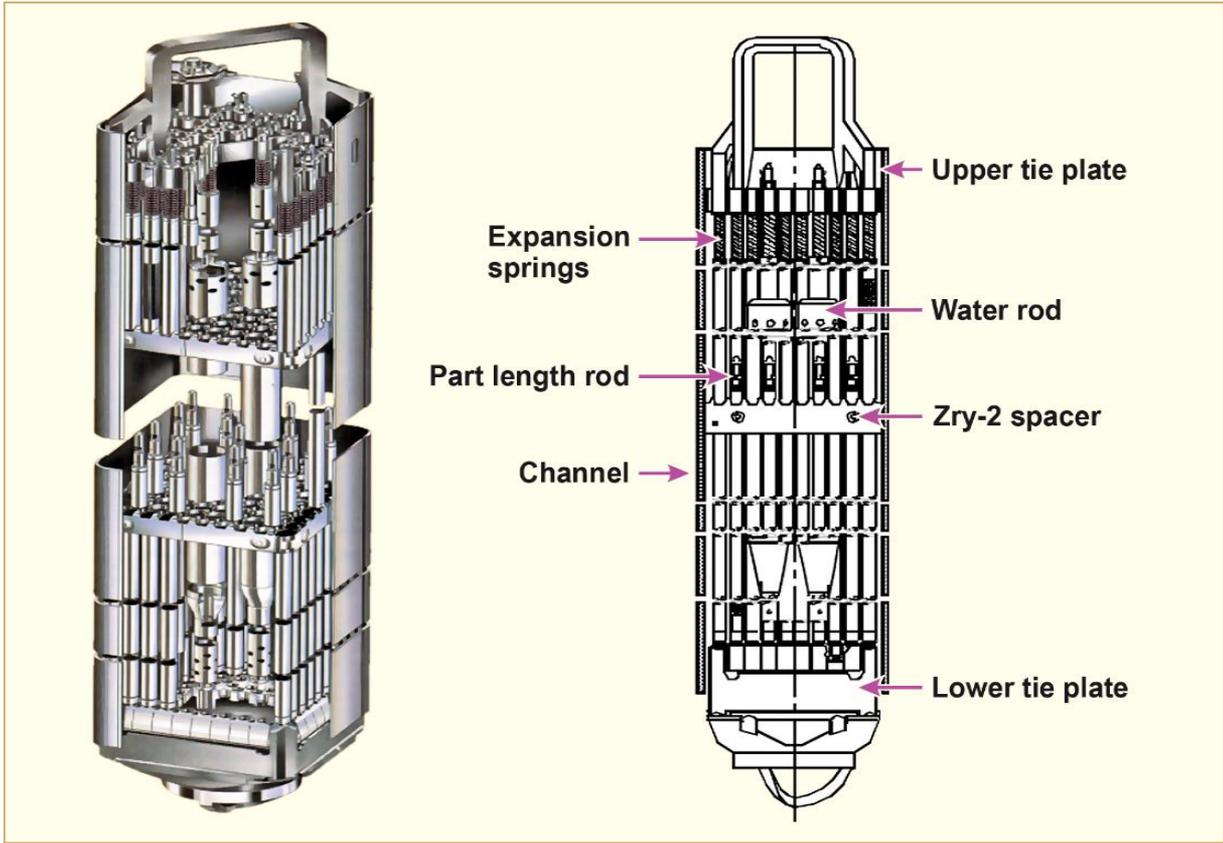


Figure B-1: GE 14 Fuel Design, provided courtesy of GE.

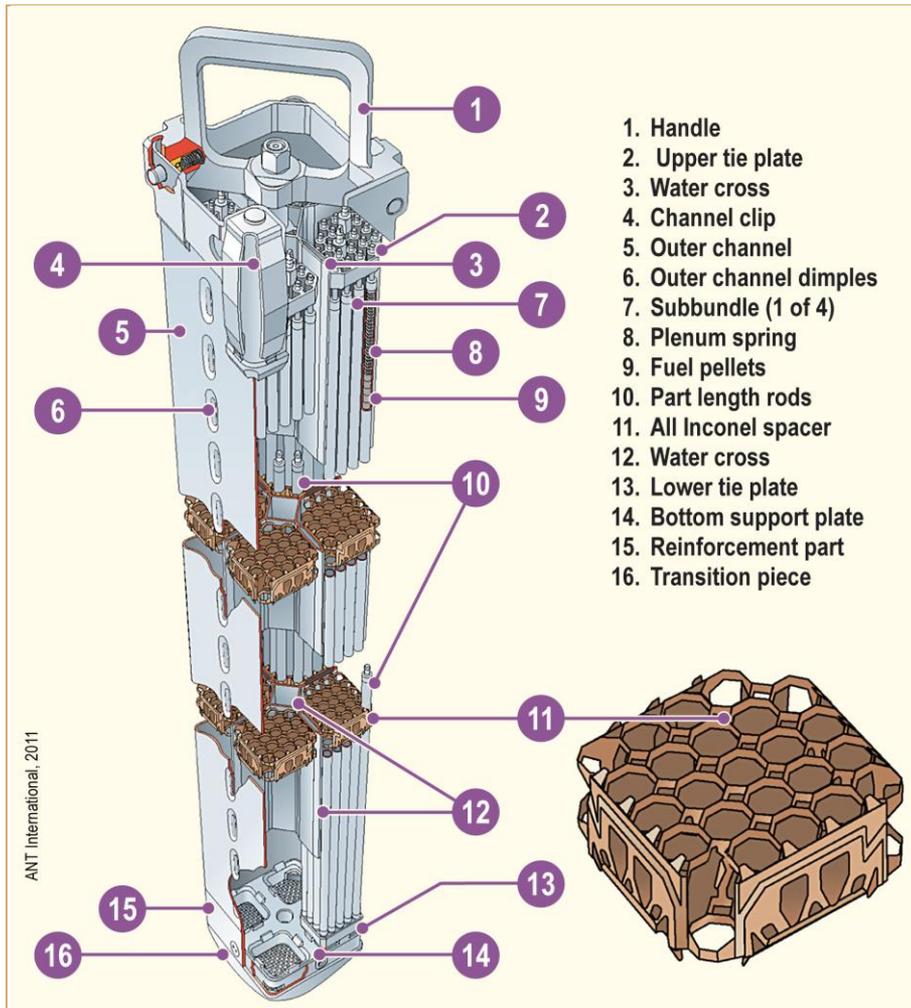


Figure B-2: The BWR Fuel SVEA-96 Optima2, provided by the courtesy of Westinghouse Electric.

## B.2 PWR fuel design

For PWRs there has also been a trend to greater subdivision of fuel rods, e.g. from 15×15 to 17×17 fuel assemblies in PWRs of the Westinghouse design. However, since the control rods and control rod drives in PWRs are designed for a specific lattice configuration, they do not have the same flexibility with respect to changing fuel designs as BWRs. There is however, one exception namely DC Cook 1 which was switching to 17×17 by changing the reactor internals. Figure B-3 shows the current PWR fuel rod arrays.

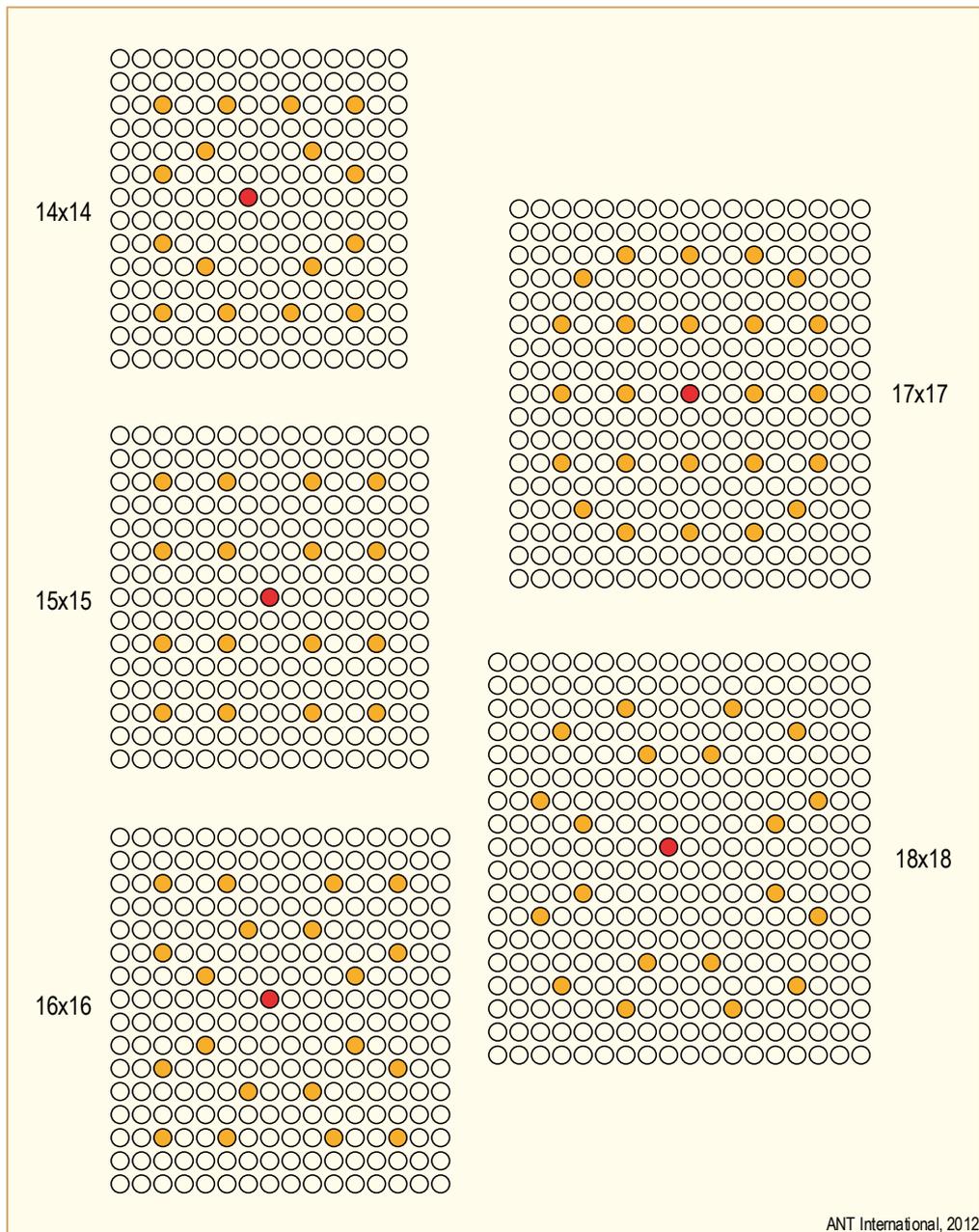


Figure B-3: Layouts of different PWR fuel assembly arrays. Rods marked with yellow colour are guide tubes into which the control rod cluster is inserted. The position marked by a red filled circle is the instrument tube position.

In most PWRs, the assemblies are positioned in the core by bottom and top fittings, and the lateral clearances are restricted by the assembly-to-assembly contacts at the spacer-grid levels. PWR control rods consist of Rod Cluster Control Assemblies, RCCAs, the poison part of which moves into guide thimbles or tubes (GTs). These guide thimbles are an integral part of the assembly structure. Figure B-4 shows an example of the Westinghouse PWR fuel design.

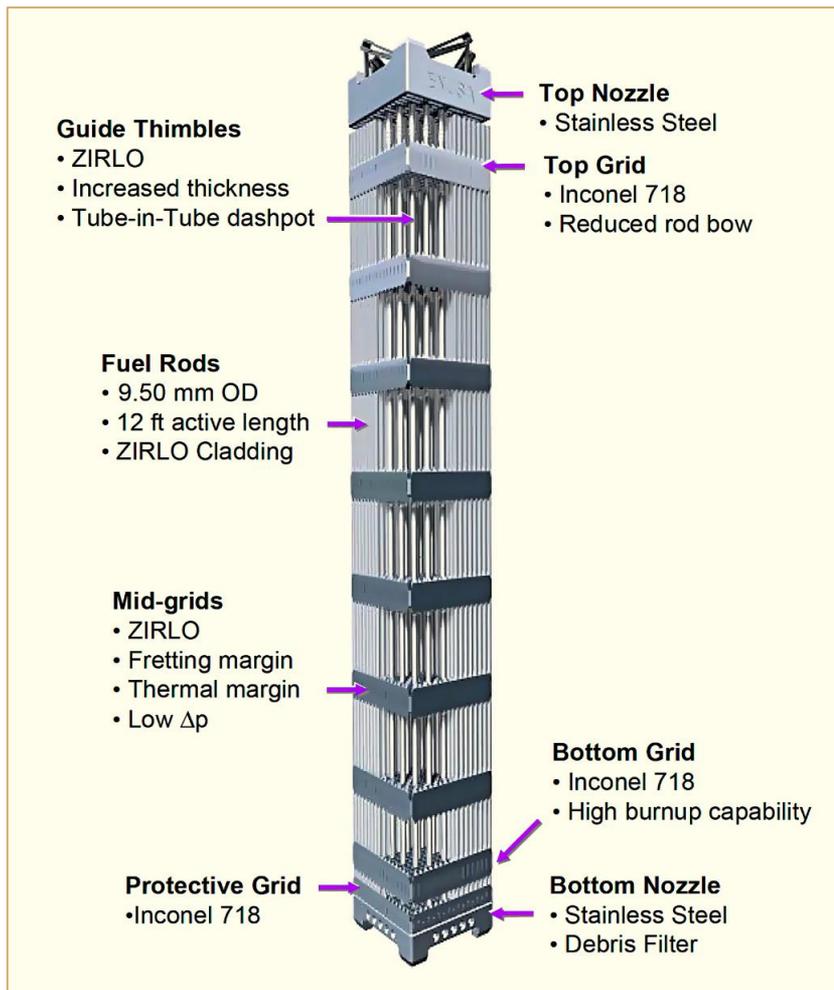


Figure B-4: RFA<sup>51</sup> fuel [Aulló et al., 2012].

### B.3 VVER fuel design

The design and construction of VVER fuel differs slightly between the VVER-440 and VVER-1000 reactors. All VVERs cores are based on a hexagonal geometry, with the individual fuel assemblies and control rods having hexagonal symmetry. VVER-440 fuel assemblies have a hexagonal cross section that contain fuel rods arranged on a triangular pitch. Early VVER-440 fuel assemblies incorporate an outer solid sheath that provides out-of-plane structural rigidity and limits cross flows. A later (third generation) design introduced in 2010 utilizes structural members at each corner of its hexagonal lattice without a solid shroud.

A VVER-440 core contains 349 fuel assemblies, each having 126 fuel rods. In V-213 and V-230 cores there are 37 control assemblies. The control assemblies are twice as long as the standard assemblies, with the upper half made from a hexagonal boron-steel (20%Cr, 16%Ni, 2%B) absorber segments in a Zr-2.5%Nb sheathed tube.

Until 1998–2000, VVER fuel cycles were typically 12 months. The fuel loaded had an enrichment of up to 3.6% <sup>235</sup>U in VVER-440 units and 4.4% in VVER-1000 units. The burn-up is up to 40 GWd/tU in a typical lifetime of four fuel cycles. Higher enrichment next generation fuel ('profiled' fuel) with an enrichment of 3.82% <sup>235</sup>U has been loaded from 1998 and fuel assemblies with enrichments of up to 4.25% <sup>235</sup>U

<sup>51</sup> Robust Fuel Assembly

containing gadolinium burnable poison and a lifetime of up to five fuel cycles and a burn-up of up to 57 GWd/tU were being loaded by 2005–2006.

In the early VVER-440 reactors (440 MWe), the rods were clad with E110 (Zr1Nb) tubes. The assemblies contained 10 or 11 stainless steel grid spacers and were enclosed in an E125 (Zr2.5Nb) hexagonal wrapper (which performs the same function as a BWR channel) [Smirnov et al., 1994]. An example of such fuel is shown in Figure B-5. As noted above, this design was modified in 2010 to replace the flow channel with corner structural elements.

In some early VVER-440 fuel assemblies that were tested in reactors such as the MR or MIR, boron was added to the E125 wrappers for reactivity control. In commercial power reactors, boric acid is usually added to the coolant for reactivity control along with movable control assemblies.

The fuel is uranium-enriched up to 4.95% and is arranged in a hexagonal grid. Fuel pellets can have a central coaxial hole.

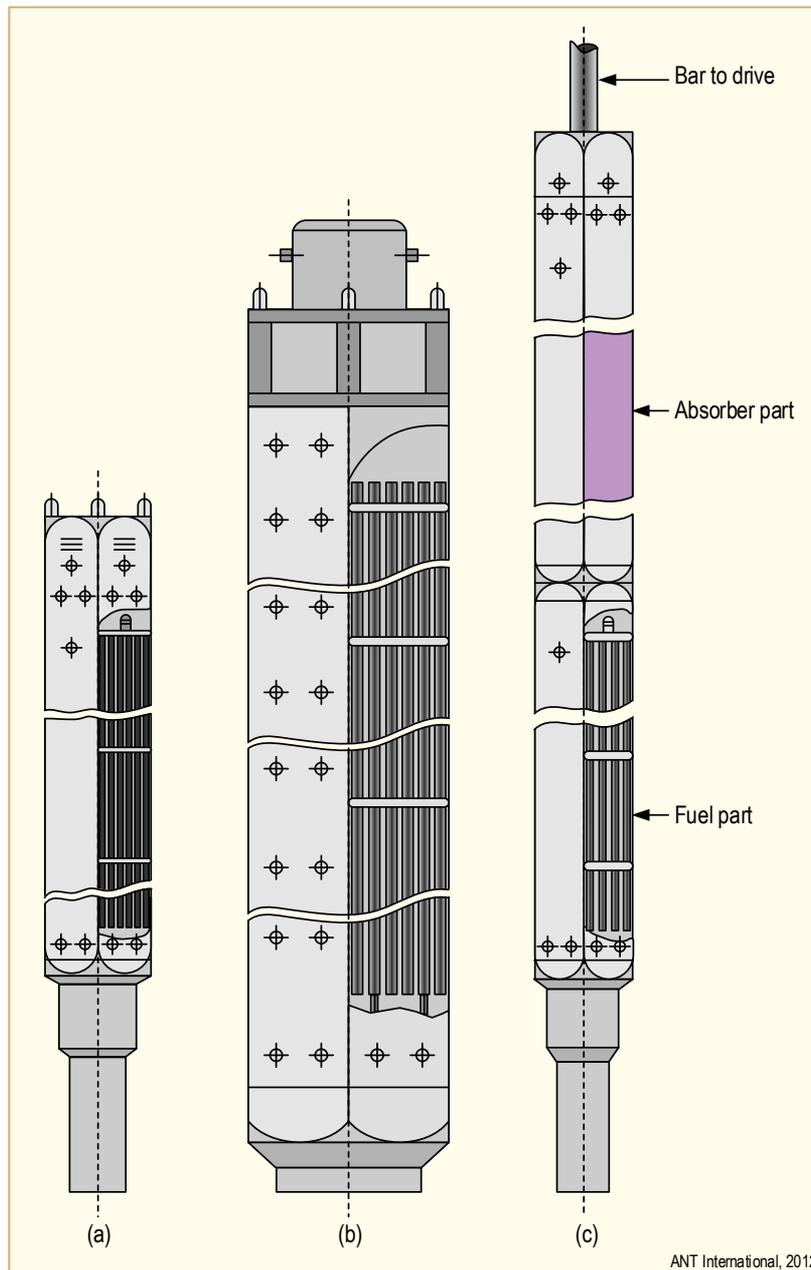


Figure B-5: Schematic diagram of VVER fuel: (a) VVER-440 operating fuel assembly, (b) VVER-1000 fuel assembly, (c) VVER-440 control rod and fuel assembly.

All VVER-1000 cores have 163 fuel assemblies, each containing 312 fuel rods in a hexagonal lattice, a central E125 alloy support tube and 18 stainless steel guide tubes for control and shut-down rods. VVER-1000 fuel assemblies consist of the following components: skeleton, bundle of fuel and gadolinia rods, top nozzle and bottom nozzle. The general views of two base designs VVER-1000 FA are illustrated in Figure B-6 and Figure B-7.

In VVER-1000 units the control assemblies used in the VVER-440 design was replaced by sixty-one PWR-type reactor control rod clusters (RCCAs) containing either boron carbide (Russian fuel) or silver-indium-cadmium absorber (Westinghouse fuel). Standard Russian fuel supplied by TVEL has uranium dioxide pellets with which are either solid or annular; i.e., contain a central hole which fills with increasing burn-up. The fuel has a zirconium-1% niobium alloy (Zr-1%Nb) clad and when outer sheaths are used they are made from zirconium-2.5% niobium alloy. Spacer grids and end plates were made from the same 08X18H10T SS as used in other parts of the 12 primary circuit up to about 1998, but later Russian fuel has Zirconium-1% niobium alloy spacer grids. Fuel supplied by Westinghouse and BNFL has either Zry-4 or ZIRLO clad and Zry-4 spacer grids. Standard VVER-1000 Russian TVEL fuel also has zirconium-1% niobium clad and 08X18H10T spacer grids and end plates.

Except for the first VVER-1000 (Novo-Voronezh-5) these VVER-1000 fuel assemblies do not have wrappers; i.e., they are essentially open like PWR fuel assemblies rather than encased in a flow channels like the early VVER-440 and BWR fuel assemblies. Zirconium alloy E-635 (Zr1.3Sn1Nb0.4Fe) is used as the structural material for the skeleton angle pieces and guide channels (GC). The alloy E-635 is used because of better mechanical and radiation resistance properties relative to alloy E110 in the VVER-1000 environment.

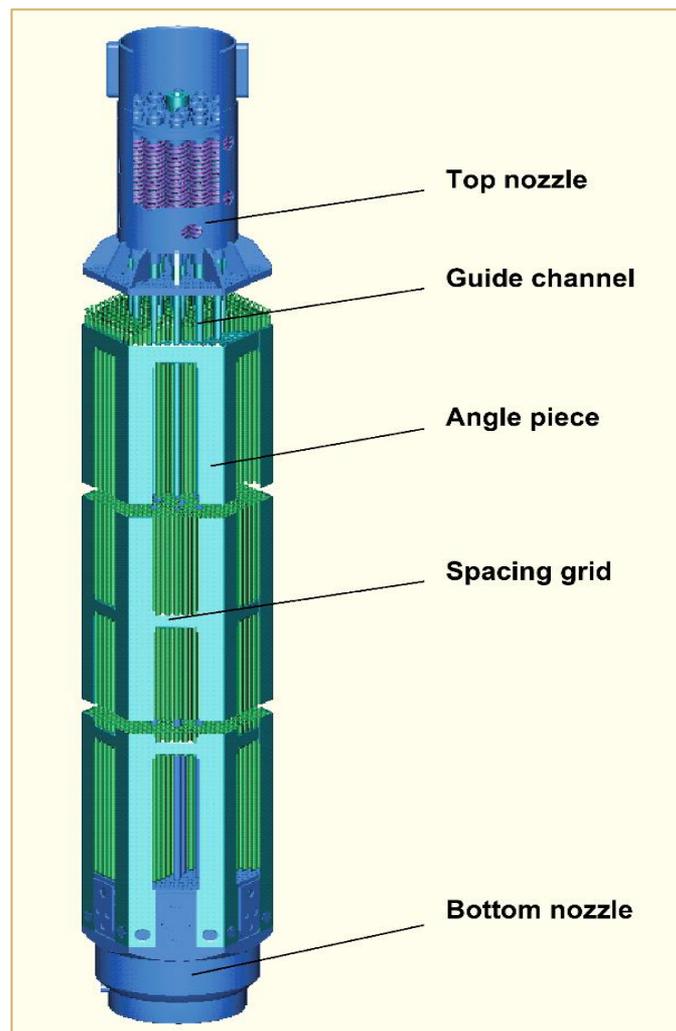


Figure B-6: VVER-1000 AFA fuel design [Dragunov et al., 2004].

## Appendix C - The materials used in fuel assemblies (Alfred Strasser)

### C.1 Introduction

The materials used for the fuel assembly components are zirconium alloys, nickel-based materials such as alloy X-750, alloy 718, alloy 625 and low-carbon, low-cobalt austenitic stainless steels (SS) [Strasser & Rudling, 2004]. Materials are selected based on their nuclear, physical and mechanical properties along with their compatibility with conditions in the core of water reactors.

Zirconium alloys are now used exclusively for fuel cladding and are also widely used in the grids (spacers) of modern fuel because of their low capture cross-section for the thermal neutrons. Stainless steel was used in some early fuel cladding and grids (1960–1970 era), but was replaced with zirconium alloys as commercial capabilities and the related technology emerged.

Other materials such as stainless steel and nickel-based alloys are used in structural components, but have significantly larger cross-sections and can adversely affect reactivity and increase fuel-cycle costs depending on their mass and location relative to the active fuel. Upper and lower nozzles (tie plates) are usually constructed of cast, austenitic stainless steel. Springs are typically fabricated from materials with high strength and low rates of stress relaxation, such as alloy X-750 or alloy 718. These Ni-based alloys are solution heat treated and then precipitation hardened by means of thermal cycles that have been developed to achieve high strength while maximize their resistance to stress corrosion cracking. Low cobalt content is needed in Ni-based alloys and SS to minimize radiological exposure of plant personnel due to transmutation, corrosion and transport of activation products through the primary coolant system.

To lower the parasitic neutron absorption by grids, their structural components have historically been made of zirconium alloys such as Zry-2 or Zry-4. In most cases, zirconium alloy spacers utilize springs made of either alloy X-750 or alloy 718 to ensure adequate fuel rod support during irradiation. Some PWR designs utilize grids constructed entirely of a zirconium alloy. Alloy X-750 and alloy 718 is being used in an increasing number of fuel designs for the entire spacer; e.g., the top and bottom grids in some PWR designs and all spacers in some high-burnup BWR designs. In PWR fuel, Ni-base alloys can be used for the top and bottom grids because the neutron flux is much lower in these locations, resulting in a very small loss of thermal neutrons due to parasitic material absorption. In BWR fuel, Ni-based alloys are used in minimum-mass configurations (thin strips with all non-load bearing material removed) to reduce parasitic loss of neutrons. Such spacers are being used to minimize the pressure drop associated with thicker Zr-alloy spacers and to avoid hydrogen pickup and embrittlement of Zr-based spacers at fuel intended for high burnup, long residence applications.

It also should be noted that BWR zirconium alloys continue to be primarily Zry-2 or slight variants of Zry-2. PWR zirconium alloys no longer tend to be Zry-4, for reasons of insufficient corrosion resistance (and hydriding resistance) at high burnup, but have moved toward zirconium alloys with Nb additions.

### C.2 Zirconium alloys

A number of different zirconium alloys have been developed for use in light and heavy water reactors. The principal alloys are described in Table C-1. It is noteworthy that there are so many different Zr-alloys for PWR applications. Originally, Zr-4 was the standard material in PWRs. But increased corrosion rates and hydrogen pickup at extended burnups resulted in a need to develop more corrosion resistant alloys either as monotubes or as DUPLEX tubes (only used in Europe).

Both Zry-2 and Zry-4 have been used in BWRs; i.e., Zry-2 for fuel cladding and either Zry-2 or Zry-4 for fuel channels. With the composition and microstructural controls of current fuel, Zry-2 exhibits good corrosion resistance through high burnups as BWR fuel cladding. The pickup of hydrogen due to corrosion at the surface of fuel cladding is observed to increase with burnup. Such increases in hydrogen pickup have shown that alloys with better corrosion and hydriding resistance are needed. Improved versions of Zry-2 and alternate alloys are being developed. A similar condition exists for fuel channels. That is, Zry-2 is resistant to corrosion in BWR environments, but undergoes increases in hydrogen

pickup with increasing burnup and exposure to dissimilar metals (shadow corrosion – discussed later). At higher burnups, the Zry-2 fuel channel sides facing the control rod may absorb more hydrogen (from the corrosion reaction) than the channel side not facing the control resulting in a larger tendency for channel bowing<sup>52</sup> that may interfere with the control rod, see [Garzarolli et al., 2011]. However, for Zry-4 channel material the hydrogen pickup is similar to the sides facing and not facing the control rod, thus reducing the tendency for excessive fuel channel bowing. This condition has led to the replacement of Zry-2 with Zry-4 as a channel material in some core designs as a short turn solution to the channel bowing issue and has also led to the development of hydriding and growth resistant materials (ZrNbX materials), see [Garzarolli et al., 2011].

Table C-1 lists the commercial alloys today while Table C-2 lists not yet licensed Zr alloys.

The zirconium liners included in Table C-1 are thin layers of Zr-alloys metallurgically bonded to the inner surface of fuel cladding and are used to reduce the risk of fuel failure due to the Pellet-Cladding Interaction (PCI) mechanism in BWR fuel. Initially, pure Zr-sponge liners were used to maximize PCI resistance. It was later found that the pure zirconium sponge material results in a large tendency for secondary degradation of failed fuel; i.e., the formation of long axial cracks after cladding perforation due to other damage mechanisms. Therefore, all fuel vendors now add some alloying elements to increase the resistance towards secondary degradation of failed rods. The most beneficial and widely used alloying element is iron, which increases the resistance of zirconium to corrosion while only slightly degrading PCI performance.

For PWRs Duplex cladding was developed by Siemens in the 1980s, consisting of a thin outer layer (~100 µm) corrosion resistant alloy (but with low creep strength) while the major part of the cladding, providing high creep strength was high Sn Zry-4. Duplex cladding is still provided by Framatome to Swiss and Spanish utilities for their high duty plants.

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<sup>52</sup> Today two major parameters may result in fuel channel bowing:

- 1) Shadow induced bow (only for Zry-2 and perhaps other nickel containing Zr alloys),
  - a) Can be eliminated by replacing Zry-2 with an alloy not containing Ni
- 2) Fast fluence induced bow which can be reduced by
  - a) Using a Nb containing alloy such as e.g. Low Tin ZIRLO or NSF Nb-Sn-Fe alloy used by GNF, see Table C-2, which show a smaller slope of the irradiation growth curve vs. fast fluence. A smaller slope results in a smaller difference in irradiation growth rate of two facing channel surfaces and thereby a lower bowing rate
  - b) Beta-quenching any Zr alloy which will result in a random texture which means that there will be no irradiation growth of the facing channel surfaces. Thus, there will be no contribution from fluence induced growth to channel bowing. Some channel bowing may however occur for beta-quenched material due to differences in hydrogen content on the facing sides of the fuel channel.

Table C-1: Commercial zirconium base materials currently used for zirconium alloy fuel components in PWRs, BWRs, CANDU, Voda Voda Energo Reactor (VVER), RBMK.

Alloy	Sn%	Nb%	Fe%	Cr%	Ni%	O%	Fuel Vendor
<b>BWRs</b>							
Zry-2 (SRA <sup>A</sup> /RXA <sup>B</sup> ) <sup>1)</sup>	1.2–1.7	–	0.07–0.2	0.05–0.15	0.03–0.08	0.1–0.14	All fuel vendors
<b>Zr-Liner<sup>2)</sup></b>							
Sponge	–	–	0.015–0.06	–	–	0.05–0.1	Only used in Japan and Russia
ZrSn	0.25	–	0.03–0.06	–	–	0.05–0.1	W
ZrFe	–	–	0.4	–	–	0.05–0.1	AREVA
ZrFe	–	–	0.10	–	–	0.05–0.1	GNF <sup>C</sup> P8 <sup>3)</sup>
<b>PWRs</b>							
Zry-4 (SRA)	1.2–1.7	–	0.18–0.24	0.07–0.13	–	0.1–0.14	
PCA-2b (PR)	1.3	–	0.3	0.2	–	0.12	Siemens
PCAm	1.3	–	0.3	0.2	–	0.12	Siemens
ZIRLO (SRA)	1	1	0.1	–	–	0.12	W
Optimized ZIRLO (SRA/pRXA <sup>D</sup> )	0.7	1	0.1	–	–	0.12	W
M5 (RXA)	–	0.8–1.2	0.015–0.06	–	–	0.09–0.12	AREVA
HPA-4 <sup>E</sup> (SRA/RXA)	0.6	–	Fe+V	–	–	0.12	AREVA
NDA <sup>F</sup> (SRA)	1	0.1	0.3	0.2	–	0.12	NFI <sup>G</sup>
MDA <sup>H</sup> (SRA)	0.8	0.5	0.2	0.1	–	0.12	MHI <sup>I</sup>
<b>Duplex<sup>4)</sup></b>							
ELS (SRA) <sup>J</sup> <sup>5)</sup>	0.5/0.8	–	0.3/0.5	0.2	–	0.12	Siemens
D4 (SRA)	0.5	–	0.7	–	–	0.12	Siemens
3b (SRA) <sup>6)</sup>	0.8	–	0.5	–	–	0.12	W
3b+ (SRA) <sup>7)</sup>	<1.0	–	<0.6	–	–	0.12	W
D4 (SRA) <sup>8)</sup>	0.5	–	0.5	0.2	0.03	0.12	W
<b>VVER, RBMK</b>							
E-110 (RXA)	–	0.9–1.1	0.014	<0.003	0.0035	0.06	Fuel cladding
E-635 (RXA)	1.3	1	0.4	–	–	0.09	Fuel structural components
Alloy E125 (SRA)	–	2.5	–	–	–	0.06	Pressure tube in RBMK
<b>CANDU</b>							
Zry-4 (SRA)	1.2–1.7	–	0.18–0.24	0.07–0.13	–	0.1–0.14	Fuel cladding
Zr2.5Nb (SRA)	–	2.4–2.8	<0.15	–	–	0.09–0.13	Pressure tube
<p><sup>A)</sup> Stress Relieved Annealed <sup>B)</sup> Recrystallised Annealed <sup>C)</sup> Global Nuclear Fuel <sup>D)</sup> Partially Recrystallised Condition <sup>E)</sup> High Performance Alloy <sup>F)</sup> New Developed Alloy <sup>G)</sup> Nuclear Fuel Industries <sup>H)</sup> Mitsubishi Developed Alloy <sup>I)</sup> Mitsubishi Heavy Industries <sup>J)</sup> Extra-Low Sn</p> <p>1) All but one of the BWR fuel vendor use RXA, fuel cladding. The exception uses SRA, fuel cladding.                  2) In all BWR cladding with liners, about 90% of the thickness-the outer part of the tube consists of Zry-2.                  3) Starting with Process 8 (P8), GNF adopted a liner that contains somewhat higher Fe content than that the earlier liners (P7 and earlier). The current process is P9.                  4) All DUPLEX claddings consist of an outer corrosion resistant layer with a thickness &lt;100 microns and the rest of the thickness is Zry-4 to provide the mechanical strength.                  5) All Siemens duplex claddings contains Zry-4 with 1.5 wt%Sn                  6) Zry-4 with 1.3 wt %Sn                  7) Zry-4 with 1.5%Sn                  8) Zry-4 with 1.5%Sn</p>							
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Table C-2: New zirconium base materials for zirconium alloy fuel components (not yet licensed) in PWRs, BWRs, CANDU, Voda Voda Energo Reactor (VVER).

Alloy	Sn %	Nb%	Fe%	Cr%	Ni%	Other Elements	Fuel Vendor
<b>BWRs</b>							
Zry-2 B ( $\alpha+\beta$ ) quenched							GNF - cladding
Zry-2C ( $\beta$ ) quenched							GNF - cladding
ZIRON	1.45		0.25	0.1	0.1		GNF- claddings
NSF	1	1	0.4				GNF- channels
LowSn ZIRLO	0.67	1	0.1				WES - channels
HiFi	1.5	-	0.4	0.1	0.08		NFI - claddings
Z4B	1.20-1.70	-	>0.24	>0.13			AREVA-channels
<b>PWRs</b>							
AXIOM (80%PR <sup>A</sup> )	0.3	0.8	0.05			0.12%Cu, 0.2%V	W
HANA-4	0.4	1.5	0.2	0.1			KNF
HANA-6		1				0.05%Cu	KNF
M5-Fe (RXA)	-	0.8-1.2	0.015-0.06	-	-		AREVA
Q12 (RXA)	0.5	1	0.1				AREVA-guide tubes
J1 (RXA)	-	1.8	-	-			MNF
J2 (RXA)	-	1.6	-	0.1			MNF
J3 (RXA)	-	2.5					MNF
S2 (RXA)	0.8	0.1	0.3	0.1			NFI
M-MDA (SRA)	0.5	0.5	0.4	0.3			MNF
<b>VVER</b>							
E-110M (RXA)	-	0.9-1.1	0.12			0.12	TVEL
E-110G (RXA)	-	0.9-1.1	0.04			0.08	TVEL
E-635M (RXA)	0.8	0.9Nb	0.35Fe			0.08	TVEL
A) Partially Recrystallised Condition							
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## Appendix D - Fuel behaviour and Fuel Fragmentation, Relocation and Dispersal (FFRD)

### D.1 Basic information on UO<sub>2</sub> fuel

During base irradiation and during a LOCA event, the fuel pellet stack is subjected to various physical phenomena which can have a significant impact on the LOCA fuel performance. These are:

- **The Fission Gas Release (FGR)** occurring during base irradiation increases the rod inner pressure (RIP), which can affect the ballooning behaviour of the cladding and the probability and time of the cladding burst during LOCA (if good axial gas communication is available within the fuel rod).
- **The rod internal pressure** and the amount of moles of gas available locally to balloon and burst the cladding (the FGR-prior to the LOCA- and TFGR-during the LOCA- constitutes the parts of the driving force and of the volatile radiological source term).
- **The degradation of the thermal conductivity** of the fuel pellet, which occurs during base irradiation by (1) irradiation damage, (2) fission product formation and (3) the formation of the porous outer high burnup rim, causes a relative increase of the pre-transient fuel temperature and the corresponding stored energy, which in turn play a primary role in the course of the transient (i.e., decay heat correlation used in the safety analysis).
- **The high burnup structure (HBS)** which is formed at the pellet periphery during base irradiation exhibits a high inventory of fission gases, contained mainly in large over-pressurized inter- and intra-granular pores. During a LOCA transient the outer rim experiences a temperature increase, e.g. from 400 to 1100 °C, see Fig. 0-8, which may lead to pronounced transient fission gas releases, TFGR, which in turn can favour localized ballooning of the cladding (possibly without the help of the pre-transient gases if the axial gas transport is impaired as it is expected in high burnup fuel with strong pellet-clad bonding).
- **Fuel fragments relocation** may occur after ballooning and burst if the driving force is sufficient to slump the fuel fragments from upper location in the ballooned section. Such fuel relocation increases locally the fuel power density and subsequently the cladding temperature as well as the Equivalent Cladding Reacted (ECR). This temperature rise is the result of 2 antagonist phenomena: (i) improved heat transfer coefficient thanks to a larger exchange surface of the balloon and (ii) degraded cooling due to the ballooning of the neighbouring fuel rods (see section 3.2.4, in the later phase of *LOCA*).

Figure D-1 shows the changes that occur during base irradiation for a typical 17×17 fuel rod (with a length of 3848 mm, an active length of 3658 mm, a 9.5 mm outer cladding diameter, 0.57 mm cladding wall thickness, a pellet with a diameter of 8.19 mm, a pellet density of 10.43g/cm<sup>3</sup>, and a pre-pressurization of 20 bars) and different Linear Heating Generation Rates, LHGRs. The data were taken or calculated from information from the ZIRAT8/IZNA3 Topical report on High Burnup Fuel Issues [Adamson et al, 2003/2004]. The figure reveals that:

(1<sup>st</sup>) The HBS (or high burnup rim) is formed at pellet average burnups in excess of about 40-50 MWd/kgU and extends radially inward at higher burnup,

(2<sup>nd</sup>) the inner gas pressure increases as consequence of the FGR and pellet-clad gap closure under normal operating conditions, especially at burnups in excess of 40-60MWd/kgU, depending on LHGR, and

(3<sup>rd</sup>) fuel temperature increases (due to a decrease of the thermal conductivity of UO<sub>2</sub> as consequence of the chemical changes and lattice damage) almost linearly up to 50 MWd/kgU and even more at higher burnups (due to the formation of the porous high burnup rim).

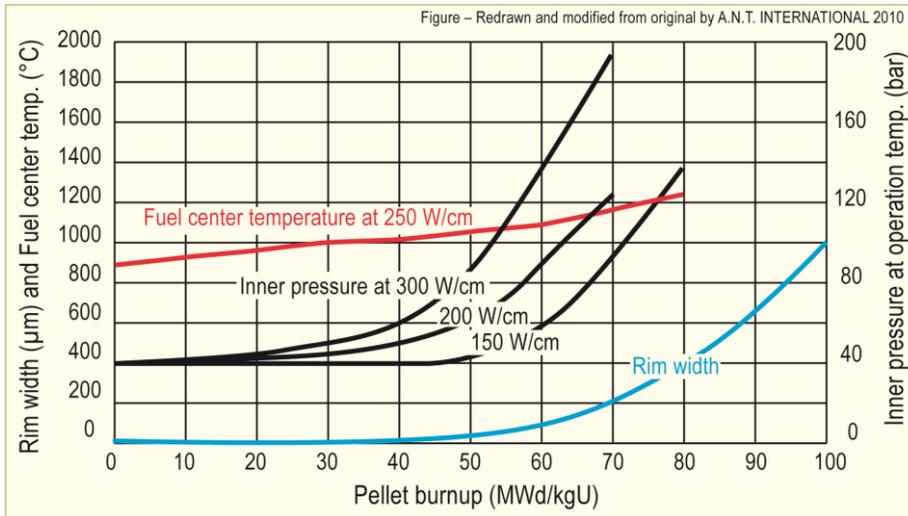


Figure D-1: Changes within the fuel rod for a typical  $17 \times 17$  fuel rod and different LHGRs. The data were taken or calculated from information in [Adamson et al, 2003/2004].

It is crucial that all fuel microstructural changes that are relevant for *LOCA* fuel performance can be correctly modelled in the fuel vendor licensing codes.

### D.1.1 Fission Gas Releases (FGR) during base irradiation

The FGR mechanism during normal operation includes several processes. The fission gases, formed during irradiation, remain initially in the  $\text{UO}_2$  lattice in form of supersaturated solution or fine pores. Once the effective solubility, which decreases with increasing temperature  $>1000$ - $1200$  °C, is exceeded larger pores are formed at the grain boundaries. These pores at the grain boundaries grow till they interlink to tunnels and release a fraction of the fission gases to the space inside the fuel rod. Besides this thermal release mechanism, gas is also released by an athermal mechanisms, i.e., release by recoils from open surfaces, including those formed by irradiation induced diffusion and restructuring.

In-pile-experience has shown that FGR is also strongly related to micro-structural changes in the fuel pellets, such as grain growth and HBS formation. As pointed out in the ZIRAT8 Topical report on High Burnup Fuel Issues [Adamson et al, 2003/2004], FGR during steady state irradiation increases with increasing LHGR and burnup. Furthermore, additional amounts of fission gasses can be released during power ramps after control rod manoeuvres or fast power changes during base irradiation period. Within a *LOCA* licensing process, the licensee must calculate with validated fuel performance code FGR at the onset of the anticipated *LOCA* transient, accounting for:

- The manufacturing pre-pressurization.
- Total amount of fission gases produced in a fuel rod during base irradiation (FGR).
- The TFGR during the considered *LOCA* event.
- The size of the free volume (plenum, gap and void volume within the pellet column) within a fuel rod.
- The hydraulic section within the fuel rod (to assess axial gas communication and the number of moles of gas effectively available at PCT, where the balloon is going to occur).

The rod internal pressure can be calculated during the considered LOCA event for any assumed LOCA temperature history. Figure D-1 gives an indication for the potential inner pressure at operation temperature versus burnup. From the rod internal pressure value and the predicted cladding temperature history during an anticipated LOCA, the fuel rod ballooning behaviour, as well as the burst failure time can be calculated. As indicated in Figure D-1, the rod internal pressure build up during base irradiation can be expected to increase sharply at burnups in excess of 40-60 MWd/kgU depending on LHGR. Most of the codes used for licensing are capable of conservatively predicting the potential FGR and rod internal pressure at the onset of the LOCA under consideration. Depending on the particular licensing demands, realistic predictions with statistical methods may be more appropriate in the future than deterministic analyses based on bounding conditions, e.g. [Heins, 2004].

### D.1.2 Degradation of thermal conductivity and increase of stored energy

Thermal conductivity of  $UO_2$  is a key parameter affecting the thermal behaviour of the fuel during reactor operation and thus an important value in code calculations. The parameters affected by the fuel thermal conductivity are, in particular, the fuel temperature, thermal expansion, fission gas release and gaseous swelling. Thermal conduction in  $UO_2$  is a complex physical process and depends on many parameters such as the temperature, the density (porosity), the stoichiometry (the U/O ratio), impurities and additives, and the duration of irradiation (burnup). Irradiation damage, fission product formation and the formation of a porous outer rim leads to a reduction of the thermal conductivity of the  $UO_2$ , e.g. Ronchi, et al., 2004. This causes a relative increase of fuel temperature, *FGR*, and the stored energy to be considered for safety analysis. In Figure D-2 the predicted fuel center temperature is given for a hypothetical fuel rod irradiated at a constant *LHGR* of 250 W/cm, as reported by Sontheimer & Landskron, 2000. The figure indicates a nearly linear increase of the center temperature up to a burnup of about 60 MWd/kgU. At higher burnups the porous high burnup rim forms and degrades the thermal conductivity even more. As consequence the centreline temperature increases even more pronounced at higher burnups.

For *LOCA* analysis calculations of the stored energy and the cladding and fuel temperature considering the temperature profile immediately before *LOCA* have to be performed by appropriate codes. The decreasing thermal conductivity of  $UO_2$  causes a relative increase of fuel temperature and the stored energy during base irradiation. The release of the stored energy induces higher temperatures during the early stage of blow-down phase of the anticipated *LOCA* event. As consequence of the higher temperatures the rod bursts will occur earlier during the *LOCA* transient leading to a larger degree of clad embrittlement since double sided oxidation starts earlier, see section 4.2. Formation of HBS (high burnup structure) during base irradiation and TFGR during LOCAs

The high burnup rim zone, that forms at the rim of the  $\text{UO}_2$  pellet at pellet average burnups in excess of about 40-50 MWd/kgU, has a particular structure. The high burnup rim structure is characterized by a simultaneous reduction in grain size, increase in porosity and depletion of fission gas from the  $\text{UO}_2$  matrix. It starts at a local burnup 50-70 MWd/kgU (the local burnup at the pellet periphery is due to the fissile  $\text{Pu}^{239}$  formed by epithermal neutrons increased by a factor of 1.6 to 2.4 in comparison with the pellet center). Tangled dislocation, formed by fission damage, acts as nucleus for  $\text{UO}_2$  recrystallization, e.g. [Une et al, 1997]. The fully developed rim structure has a grain size of 0.2-0.3  $\mu\text{m}$ , which is much smaller than that of the original  $\text{UO}_2$  grain size ( $\approx 10 \mu\text{m}$ ). The recrystallized grains are depleted of fission gas and contain only a small fraction of the fission gas formed. Analysis has further shown that only a minor fraction of the fission gases formed at the  $\text{UO}_2$  rim are released to the plenum during the restructuring of the grains. A major part of the fission gas is trapped in large size pores. The size of the gas pores and their total volume (the porosity) increases with increasing burnup. At the pellet periphery (at  $r/r_0 > 0.9$ ) the porosity increases from about 7% at a pellet averaged burnup of 50 MWd/kgU to almost 15% at an average burnup of 102 MWd/kgU, whereby the porosity at the outmost pellet rim can be as high as 24%. The increase in porosity is associated with an increase in bubble size. Mean diameters of 1.5 – 2  $\mu\text{m}$  at a burnup of 67 MWd/kgU increase to a mean value of 4  $\mu\text{m}$  and a maximum value of 10  $\mu\text{m}$  at a burnup of 102 MWd/kgU [Manzel & Walker, 2000]. With increasing average pellet burnup, the width of the high burnup rim zone increases, as indicated in Figure D-1. Some effect of the initial grain size of the  $\text{UO}_2$  pellets exists on the onset and advance of the high burnup rim zone. The inward propagation of the rim zone, however, will be also affected by the temperature. Above about 1000 °C dislocation annealing will cut down  $\text{UO}_2$  grain restructuring. The burnup and temperature threshold of rim structure formation is shown in Figure D-2.

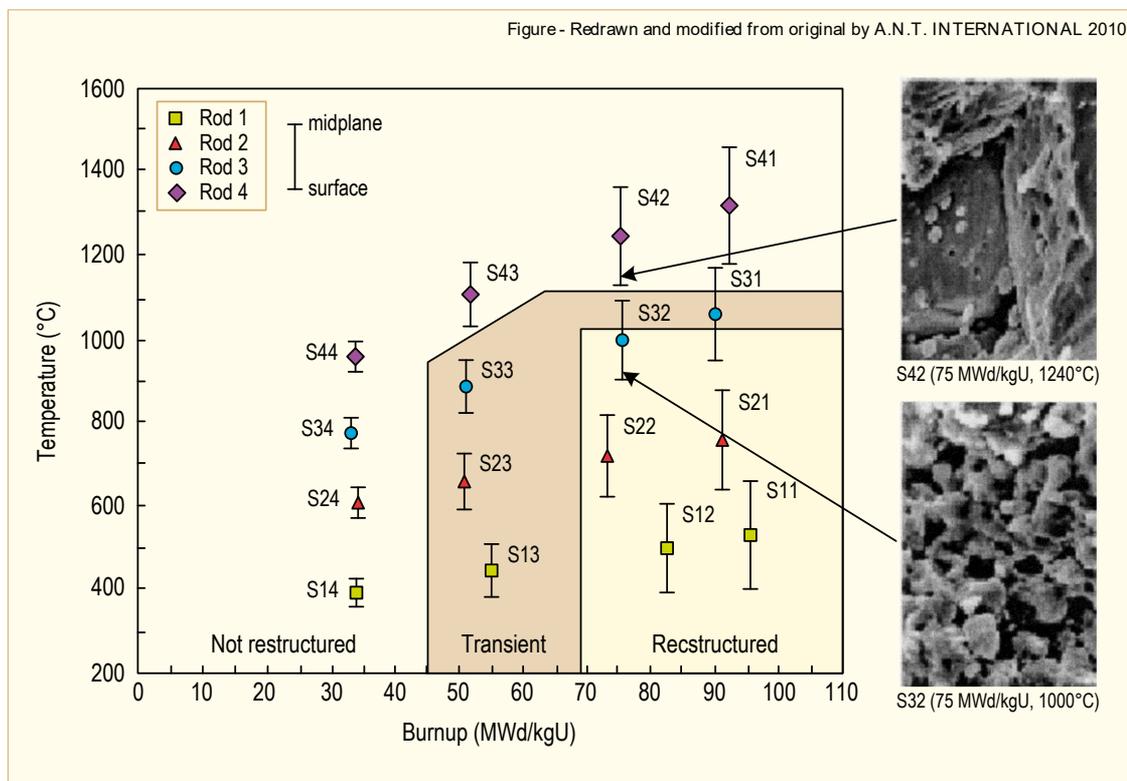


Figure D-2: Burnup and temperature threshold of rim structure formation, modified figure according to [Kinoshita et al, 2004].

The rather large fission gas bubbles within the high burnup rim zone contain fission gases at very high pressures, exceeding the equilibrium values, due to the surface tension forces and the expected arrival rates of gas atoms and vacancies. Comparison of the volume of the pores with the amount of fission gases formed indicates that these bubbles are over-pressurized by a factor of 3 and more, e.g. [Une et al, 1997] and [Guerin, 2002]. The over-pressurization occurs because irradiation diffusion of vacancies and gas atoms is not much different in the athermal region ( $< 1000 \text{ °C}$ ) but much more vacancies are needed per fission gas atom arriving a large bubble to keep the equilibrium.

The rim structure can affect the high burnup behaviour significantly. It will reduce the thermal conductivity (that increases the stored energy of importance for LOCA) as already discussed, and it may lead to a strong increase of FGR during a LOCA.

This can affect the burst probability and time of burst. During a not too fast transient, e.g., 20 K/s, as in the case of a prototypical LOCA transient, when the pellet rim temperature exceeds 1000 °C to 1200 °C, vacancy diffusion will become significant, and a rapid growth of over-pressurized bubbles localized at the grain boundary towards the equilibrium may occur. Additional bubble growth can also come from gas atoms located in the vicinity of grain boundary (particularly the gas having suffered irradiation induced re-solution at the end of base irradiation). This bubble growth may result in a rapid inter-linkage of intergranular bubbles and release of fission gases to the open space within the fuel rod.

Annealing tests have shown, that if the transient occurs without any external constraint applied by the cladding to on the fuel, the bubble growth will be much larger, and inter-linkage will occur much sooner than in a case of transient with an external constraint [Kashibe & Une, 2000]. In a LOCA transient, the mechanical properties of the cladding degrade during the high temperature phase, thus the constraint brought by the cladding is progressively lost enabling FGR and TFGR. As such, heating tests without external pressure are too conservative to be representative of the real case. It has been shown that a hydrostatic pressure of 17 MPa is enough to inhibit TFGR. (Jean Noiroit FSRM proceedings at JAEA Mito 2019). The impact of constraint on TFGR, then on ballooning and burst, is paramount when simulating a LOCA transient. Unfortunately, most if not all semi-integral LOCA tests are performed in conditions minimizing the constraint brought by the cladding (heating from outside, beginning of the transient at zero power, etc.) See Appendix F -for more details.

A great number of out-of-pile heating tests on irradiated fuel samples has been performed in several laboratories with UO<sub>2</sub> samples irradiated up to 50 MWd/kgU. These heating tests have shown that FGR increases strongly (>1%) at a certain temperature, which is quite high in this medium burnup range. Figure D-3 shows that this threshold temperature where 1% FGR will be exceeded decreases with increasing burnup but remains in the burnup range up to 50 MWd/kgU above 1204 °C.

The low FGR during a LOCA, which is deduced from Figure D-1 was confirmed by LOCA experiments with pre-irradiated fuel rods (2.5-35 MWd/kgU), where temperatures up to 1300 K were hold for up to 60 seconds [Karb et al, 1982].

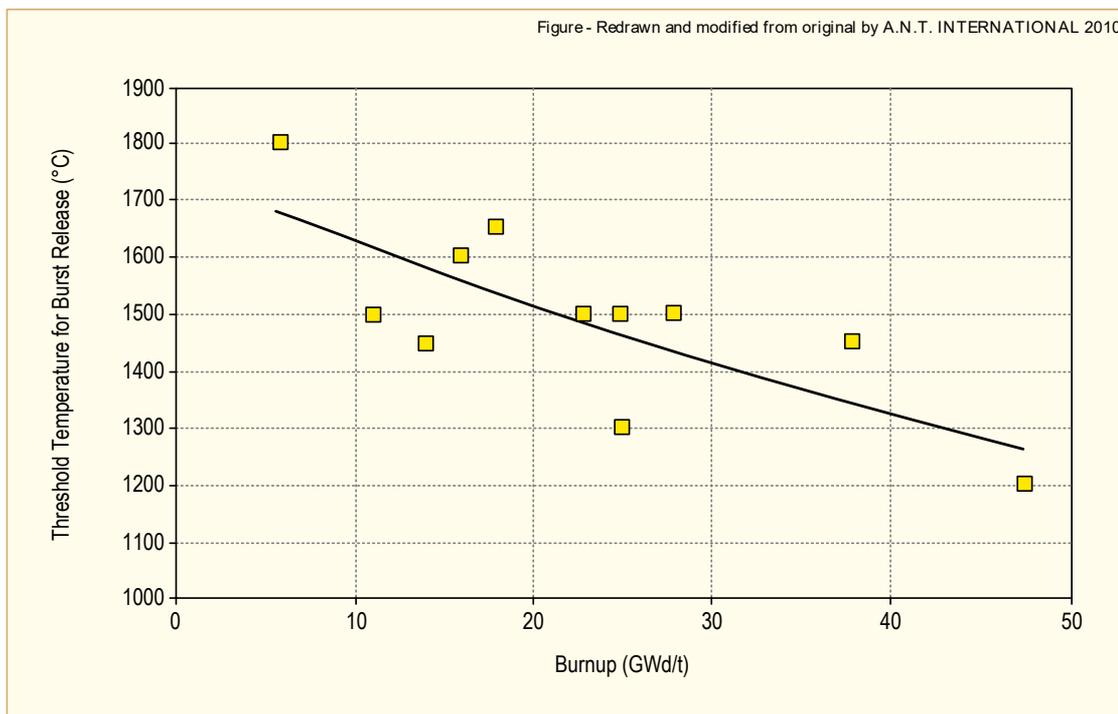


Figure D-3: Threshold temperature for onset of burst release (> 1%) during heating ramp, modified figure according to [Guerin, 2002].

TFGR was studied by [Pontillon et al, 2004] and [Marcet et al, 2009]. Analysis showed that the FGR is driven by two main mechanisms:

- The interconnections between bubbles (diffusion of atoms and vacancies at HT) at higher temperatures. A strong release burst is Figure D-4). The gas fraction released during the transient being about 12% of the initial inventory and correspond largely to the intergranular gases. Note that release fractions are based on measurements of  $^{85}\text{Kr}$  (10.76 yr half-life) due to decay between the end of irradiation and the time of measurement.
- Grain boundary fractures induced by thermal shock at about 300 °C, which lead to the release of fission gases accumulated at the grain boundaries in over pressurized bubbles (about 0.3% of the initial inventory). Comparison of the images of the pellet periphery in Figure D-4, Figure D-5 and Figure D-6 reveals severe cracking in the HBS zone, and an opening of the grain boundaries during the thermal transient. The grain boundaries are now clearly visible in the form of dark lines.

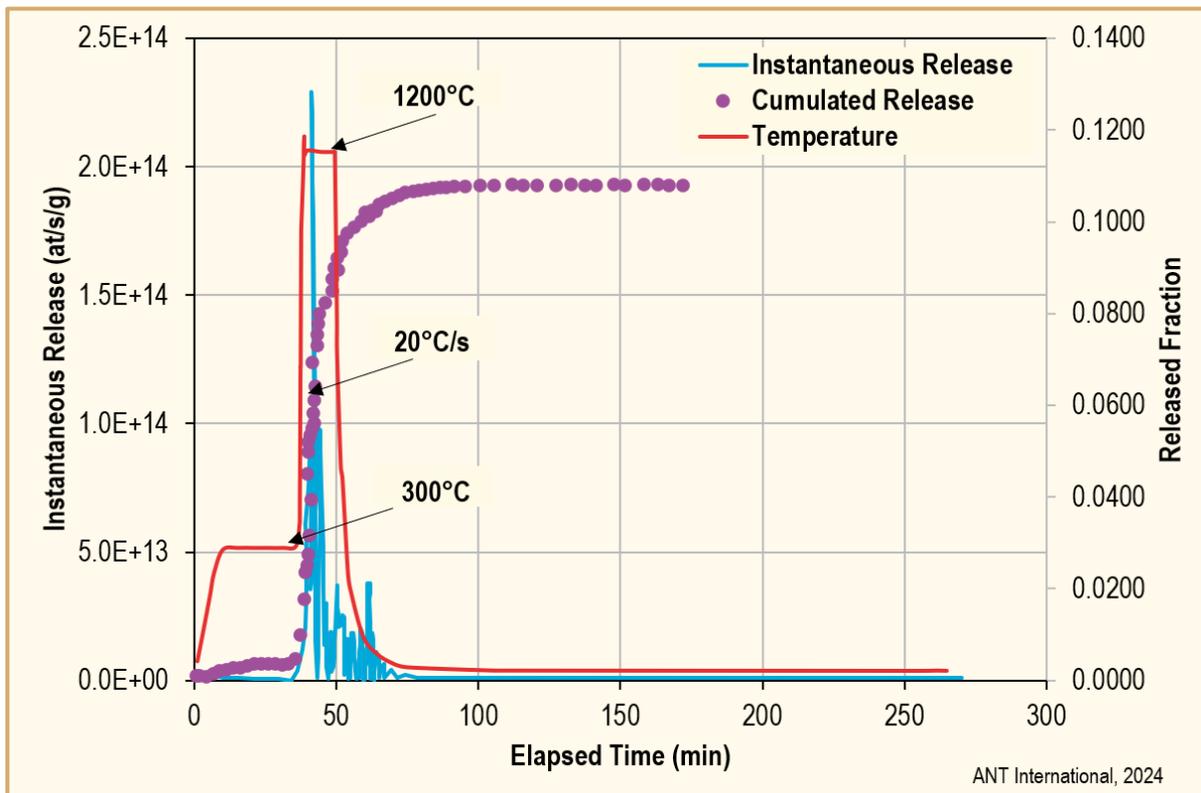


Figure D-4:  $^{85}\text{Kr}$  Instantaneous fission measured during the so-called LOCA test (1200 °C, 10 minutes, 20 °C/s). During the plateau at 300 °C, a weak  $^{85}\text{Kr}$  burst lasting about 2–3 minutes is measured, representing about 0.3% of the initial inventory,  $\text{UO}_2$  fuel burnup was 62.7 MWd/t, modified figure according to [Marcet et al, 2009].

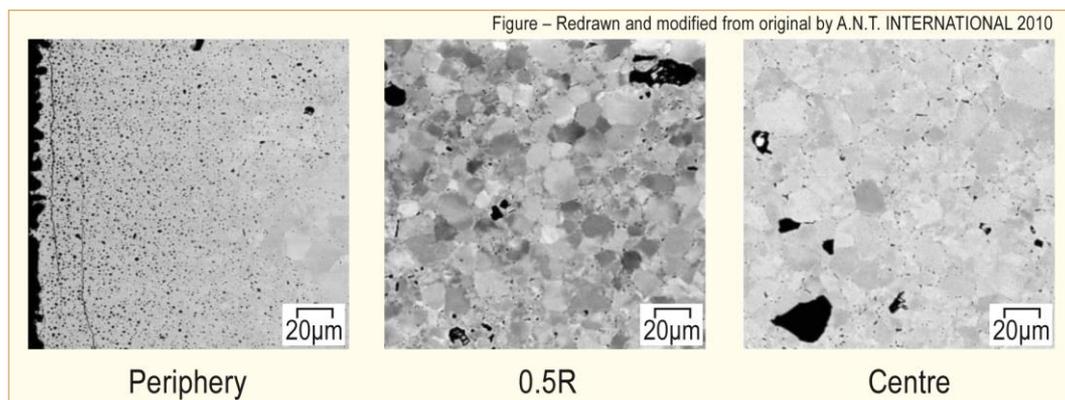


Figure D-5: SEM images at periphery, at mid-radius, and at the centre of the sample before the test. UO<sub>2</sub> fuel burnup was 62.7 MD/t [Marcet et al, 2009].

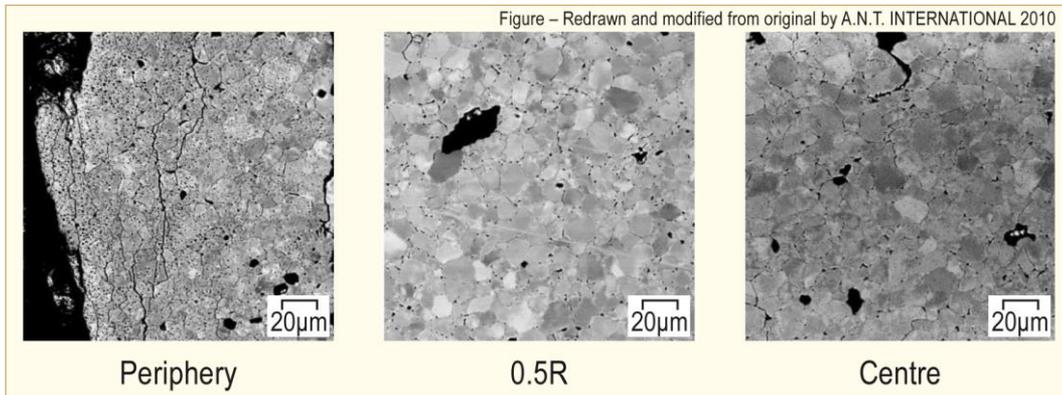


Figure D-6: SEM images at periphery, at mid-radius, and at the centre of the sample shown in Figure D-5 after the annealing test [Marcet et al, 2009].

The post-irradiation heating test and subsequent microstructural analyses identified two main release zones. First is the central gas precipitation zone which is located in  $0 < r/R < 0.45$  and released 22% of its gas inventory. The second is the HBS zone which is located in  $0.96 < r/R < 1$  and released between 32% and 45% of its initial gas inventory Figure D-7.

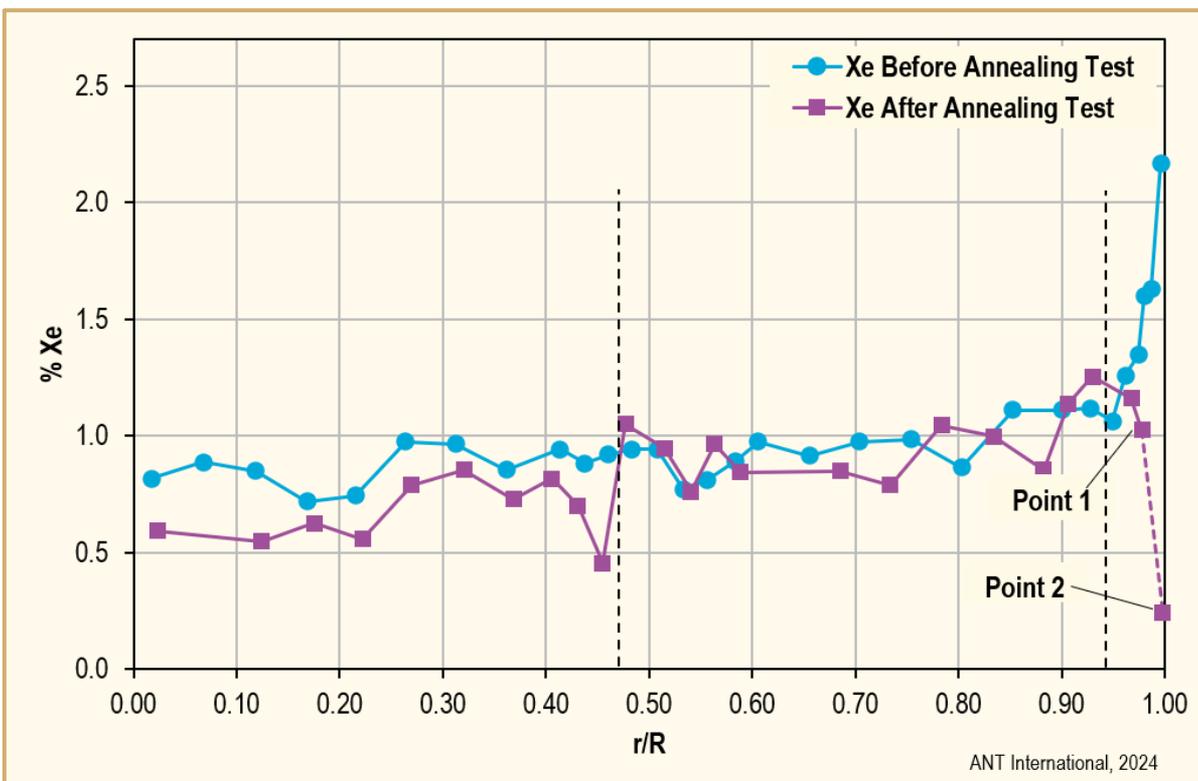


Figure D-7: Comparison of Secondary Ion Mass Spectroscopy (SIMS) xenon measurements before and after the test, modified figure according to [Marcet et al, 2009].

## Appendix E - LOCA safety analysis acceptance criteria regarding FFRD issues

### E.1 US-NRC recommendations regarding FFRD on the current fuel management schemes.

Chapter 5 provided above a current status of regulations for the Emergency Core Cooling Systems (ECCS) and several specific safety criteria (synonymous often with acceptance criteria) for the LOCA safety analyses in the United States and several other countries. Ever since the ECCS regulations were introduced in the USA, they have historically had a strong influence on the development and formulation of corresponding safety regulations and acceptance criteria regarding LOCA analysis in other countries operating nuclear power plants with LWRs.

The current regulatory situation in the US is reflected in the 2016 final draft of the 10 CFR 50.46(c) [USNRC 2016] rules and associated Regulatory Guides 1.222, 1.223, and 1.224.

Since more than two decades the most discussed and researched issue connected with these regulations are experimental evidence of the notably different behavior of high burnup fuel during LOCA conditions. Phenomena confirming this kind of behavior have been observed in numerous in-pile and out-of-pile LOCA tests with prototypical LOCA conditions on highly burned nuclear fuel specimens and were given a common abbreviated name the fuel fragmentation, relocation, and dispersal (FFRD).

As a brief reminder from Appendix D -, the FFRD can be simply described as a fine fragmentation of highly burned fuel pellets (very small fragments) and their relocation inside the fuel element, mainly axially toward the location of an outward deformation of the fuel cladding (balloon), with a tendency to accumulate in the balloon region. Clad ballooning is amplified by the transient fission gas release, a consequence of fuel pellet cracking, and a sharp temperature increase during a LOCA. These phenomena, as a consequence, create considerable risk that the rupture of the deformed fuel cladding (balloon) with sufficiently large break size, which can happen during LOCA, may lead to a dispersion of fine fragments into the steam-water mixture of the cooling channel.

Three main mechanisms for fuel pellets fragmentation observed in numerous experiments are:

1. Cracking - thermal stress, large fragments, typically observed in normal operation.
2. Fragmentation - transient-induced stress factors, transient thermal and fission gases (inter- and intragranular), small fragments.
3. Fine fragmentation (pulverizing) - specific to highly burnt fuel, likely the same mechanisms as fragmentation, resulting in very small fragments.

Both nuclear fuel vendors tests, and the international programs like in Halden (HRP) and in Studsvik (SCIP) established grounds for understanding of the fuel fragmentation phenomena, although both labs used non-prototypical test conditions, promoting FFRD. Worth to mention is that there is no direct evidence from operating power reactors available at this time. This means that experimental findings serve as a basis for knowledge on fragmentation and relocation mechanisms, as well as provide a foundation for developing thresholds for fragmentation dependent on burnup based on Figure D-16 and others in Appendix D - one can identify three types of fragmentation connected to burnup levels:

Coarse fragmentation occurs between 60 and 70 MWd/kgU (fuel rod average), with some fine debris present alongside.

1. The fraction of fine (pulverized) debris increases, likely in a non-linear manner, between 70 and 80 MWd/kgU (fuel rod average).
2. From 80 to 90 MWd/kgU (fuel rod average), only a small portion of larger fragments with almost all debris being pulverized.

Understanding the relationship between burnup and the type of debris generated is important for evaluating the potential consequences and risks associated with fuel failure during LOCA which may lead to dispersal into the coolant.

However, the results of the research programs are mostly included in reports with limited availability because such programs are conducted by either nuclear fuel vendors or as international projects with paid membership (HRP, SCIP, other). One of the best open sources of information on FFRD research results as well as critical comments and conclusions is the US NRC Research Information Letter 2021-13 [Bales et al, 2021]. Useful definitions for FFRD subject can be found there as well. The obvious risk of consequences for reactor safety prompted the US NRC, as early as 2011, to open a Generic Issue Proposal [USNRC 2011] for separate consideration of emerging FFRD issues as safety issue.

In 2012, the NRC staff completed the comprehensive review [USNRC 2012] of past research programs related to the phenomena of fuel fragmentation, relocation, and dispersal. The goal of this investigation was to revisit the conclusions of Research Information Letter 0801 [USNRC 2008] in two areas: axial fuel relocation and the loss of fuel particles through a rupture opening, and assess whether they should be incorporated into the criteria used to evaluate the acceptability of emergency core cooling systems. The review report also presents a preliminary assessment of the consequences of fuel fragmentation, relocation, and dispersal. The topics discussed are core damage distribution, fuel-coolant interaction, hydraulic and mechanical effects with relation to downstream effects, and radiological consequences. The preliminary conclusion was that the consequences of fuel fragmentation and dispersal are not likely to result in an imminent safety hazard. This conclusion was made in consideration of the anticipated small number of fuel rods expected to burst and the conservative manner in which radiological consequences for a postulated LOCA are calculated.

Performed in 2013 screening of the issue [USNRC 2013] confirmed that fuel fragmentation and relocation during a LOCA will not be treated separately and be addressed in another, already ongoing, legislative process for updating 10 CFR 50.46(c). It seems that NRC's emphasis has been on using burnup restraint to counteract the risk of fuel dispersal with the attendant risk of flow blockage and on elevated radiological consequences. Finally, US-NRC closed the Generic Issue [USNRC 2011] with following conclusion:

“The experimental results continued to support the hypothesis that FFRD phenomena are primarily a high burnup fuel issue and that the current licensing limits in the U.S. (62 MWd/kgU rod average) are adequate to prevent dispersal of large quantities of fine fuel fragments. Therefore, the rulemaking staff concluded that the 10 CFR 50.46c final rule may proceed without incorporation of regulatory requirements to address FFRD. Nonetheless, the NRC staff will continue multilateral research activities and interactions with stakeholders with the goal of developing a regulatory framework to address FFRD, if needed, in the next few years.”

On the other hand, the U.S. NRC staff have in parallel performed research on fuel fragmentation, relocation and dispersal during LOCA. The purpose of such research is twofold: to develop and make credible NRCs calculation methods and to quantify better a magnitude of potential consequences in order to determine the safety significance of these phenomena in support of the 10CFR50.46c rulemaking. An improved methodology to predict the amount of fuel dispersed during a LOCA was presented at [WRFPM 2014]. The U.S. NRC computer codes FRAPCON/FRAPTRAN and TRACE together with realistic operating conditions were used for calculations. Detailed analyses included a Westinghouse 4-loop PWR (LBLOCA), a Combustion Engineering PWR (LBLOCA and SBLOCA), and a General Electric BWR/4 (LBLOCA and SBLOCA). The results of study<sup>61</sup> provided detailed core-wide maps of fuel dispersal, where the time and location of fuel rod cladding ruptures is calculated, as well as the mass of fuel dispersed and the particle size distribution of the dispersed fuel, to be used in future consequence analyses. The calculated amounts of the dispersed fuel and its distribution over cores has specifically been concluded in the context of the ongoing rule making: “The deleterious effects of increased burnup on predicted fuel dispersal constitute a possible reason to limit burnup to current levels until the consequences of fuel dispersal are fully assessed from a safety perspective”.

The draft final update of the 10 CFR 50.46(c) [USNRC 2016] included revisions to the ECCS acceptance criteria, focusing on performance-based requirements for fuel cladding performance and explicitly addressing concerns related to high-burnup fuels and their behavior during LOCA scenarios.

- **Cladding Embrittlement Criteria:** The update refined the criteria for cladding embrittlement, emphasizing limits on cladding temperature and oxidation to ensure the cladding remains ductile during LOCA.
- **High-Burnup Fuel Considerations:** The rule changes included enhanced requirements for high-burnup fuel, which is more susceptible to fragmentation and dispersal under high-temperature conditions.
- **Fuel Fragmentation, Relocation, and Dispersal (FFRD):** Although FFRD is not explicitly named as a separate requirement, the updated regulation implicitly covers these phenomena through its focus on cladding performance and integrity during accident conditions.

The U.S. NRC does not specifically mention FFRD in 10 CFR 50.46c as a separate requirement, yet the regulation encompasses aspects of fuel behavior under LOCA that inherently address concerns. Retaining the previous general formulation for specific limits on Peak Cladding Temperature (PCT) and Equivalent Cladding Reacted (ECR) applied to high burnup fuel as well as lower burnup levels.

1. **PCT Limit:** The maximum allowed PCT is 1204°C (2200°F).
2. **ECR Limit:** The ECR, which measures the extent of oxidation of the cladding, must not exceed 17%.

In Regulatory Guides (1.222, 1.223 and 1.224) connected to 10 CFR 50.46c returns NRC to the cladding embrittlement issue for high burnup fuel (expressed in hydrogen pickup measures) but not exactly to challenges due to its increased susceptibility to fuel fragmentation and relocation at high burnup.

Fuel fragmentation and relocation can significantly impact the cladding PCT during LOCA when fine fuel fragments relocate with potential to change the heat distribution within the fuel rod. This can lead to localized increases in temperature and potentially higher PCT. The presence of fragmented fuel can also alter the heat transfer dynamics between the fuel and cladding, which might exacerbate the thermal response of the cladding, potentially leading to higher PCT and affecting overall cladding integrity. This can also lead to higher oxidation rates, increasing the ECR, and exacerbate cladding embrittlement. As embrittlement is closely related to temperature and oxidation, fuel fragmentation and relocation can thus worsen the mechanical properties of the cladding, making it more susceptible to failure. The U.S. NRC staff is well aware of this, as evidenced lately in RIL 2021-13 [Bales et al, 2021] and earlier in NUREG-2121 [USNRC 2012], but this is not translated into either the formulation of 10CFR50.46c or the associated Regulatory Guidance.

Taking into account the fuel fragmentation and relocation aspect of a LOCA with cores containing highly burned fuel turns out to be an important part in modern methodologies for analyzing this accident. Quantification of the increase in PCT, sometimes expressed as a PCT penalty, in the case of clad balloon and relocated fragments of fuel pellets has been functionally linked to the packing fraction - a metric that defines the ratio of the volume occupied by fuel fragments to the total volume in a given space. Referring to situation in high burnup fuel rods under LOCA conditions indeed axial relocation of pulverized fuel in a “packed” crumbled configuration in a balloon region will have an increased void fraction when compared to its undamaged state, impacting the overall heat removal from the fuel rod. This will, in turn, affect temperatures in the fuel and cladding, potentially driving microstructural changes, FGR, differences in cladding ductility, ballooning and burst behavior, and cladding oxidation.

Figure E-1 from [Bales et al, 2021] illustrates axial fuel relocation and packing.

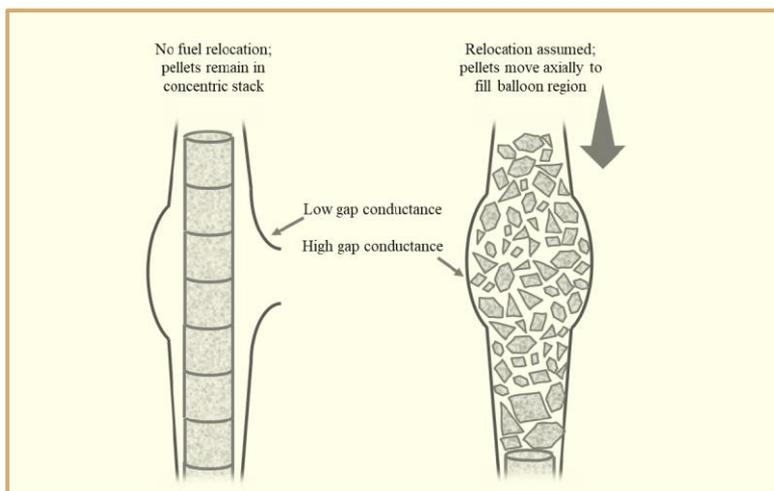


Figure E-1: Illustration of fuel relocation and packing in the ballooned region.

This phenomenon has been observed in multiple programs and facilities such as Halden, SCIP, and even earlier in the Power Burst Facility. Generally speaking, the PCT during a LOCA is expected to be inversely related to the packing fraction of fuel fragments in the ballooned region of the cladding.

**In summary**, while the U.S. NRC acknowledges the influence of fuel fragmentation and relocation during a LOCA on PCT and ECR in its publications, this recognition is not explicitly incorporated into the requirements for ECCS or the associated Regulatory Guides. Notably, there are no defined requirements mandating the consideration of PCT increases resulting from the relocation of fuel fragments to ballooned regions.

## E.2 US-NRC recommendations regarding FFRD considering the anticipated high burnup, high enrichment fuel management schemes.

The U.S. NRC employs a performance-based approach to address the safety evaluations of ECCS, implicitly incorporating fuel fragmentation and relocation phenomena. This is informed by guidance and research on high-burnup fuel's effects on cladding embrittlement under LOCA conditions [USNRC 2008;2]. The NRC actively participates in multilateral research efforts related to FFRD, funding experiments and measurement development at facilities like Halden and Studsvik. Their ongoing support for experimental research and the advancement of analytical methods underscores the NRC's commitment to understanding FFRD risks. For instance, the NRC has published a methodology for estimating fuel dispersal during LOCA conditions, which poses challenges to core coolability [USNRC 2024]. This methodology is encapsulated in the Workflow for Holistic Accident Multi-physics (WHAM), designed to assess FFRD potential within various reactor core designs and utilizing several NRC-sponsored computer codes such as SCALE/Polaris for lattice physics, PARCS/PATHS for full-core neutronics, TRACE for core and RCS thermal hydraulics, and FAST for steady-state and transient fuel performance calculations. The methodology was validated against a theoretical Westinghouse four-loop core design with high burnup and extended enrichment developed through the DOE-funded Nuclear Energy Advanced Modeling and Simulation (NEAMS) program [ORNL 2020]. Results indicate that the NRC's approach can yield valuable estimates of fuel dispersal, essential for evaluating FFRD consequences. However, several critical areas for improvement have been identified in both the codes and modeling approaches, necessitating further development before the methodology achieves full reliability.

Shifting the perspective from regulators to utilities and fuel vendors a key dilemma arises from the need to maintain burn-up limitations (in the US and several other countries of 62 MWd/kgU rod average), particularly in the absence of explicit requirements related to fuel fragmentation and relocation phenomena. This concern requires balancing safety assurances linked to implicit FFRD considerations

with the economic pressures faced by utilities and fuel vendors. Many nuclear power plants, both in the United States and globally, encounter financial challenges stemming from fluctuating market conditions of alternative energy sources. Consequently, these facilities have pursued strategies to enhance economic viability, often through increased discharge fuel burnup or the adoption of extended fuel cycles with improved fuel management practices. The latter typically leads to elevated burnup levels, strategically distributed throughout the reactor core.

Increasing the fuel rod burnup limit from 62 MWd/kgU to 75MWd/kgU is substantiated by research from the EPRI [EPRI 2019] and DOE indicating potential fuel enrichment savings and a reduction in the number of fuel assemblies. This study, relevant for the US nuclear industry, demonstrated potential benefits of increasing the burnup limit. Additionally, theoretical investigations have suggested that extending PWR cycle lengths from 18 to 24 months could further enhance economic competitiveness by minimizing reactor downtime [ORNL 2023]. Such an extension would necessitate higher average burnup levels, risking surpassing the established regulatory limit of 62 MWd/kgU, potentially approaching the upper limit of 75 MWd/kgU as assessed by EPRI [EPRI 2019]. This elevation raises concerns regarding reactor safety and performance, notably regarding the likelihood of fuel fragmentation and relocation during LOCA. Research has identified specific reactor operating conditions that increase core susceptibility to fuel fragmentation and relocation, revealing a strong correlation with both rod burnup and linear heat rate (LHR) history.

In the ORNL study [ORNL 2023], two distinct fuel loading patterns were developed using parallel simulated annealing (PSA) coupled with the PARCS computer code. The first core optimization aimed to maximize reactor cycle length while adhering to regulatory limits on radial peaking factor and soluble boron concentration, achieving a peak rod average burnup of 75 MWd/kgU. The second objective focused on minimizing susceptibility to fuel fragmentation and relocation during a LOCA, targeting a 24-month cycle length while complying with regulatory constraints. Validation of predictions generated by the PARCS model was performed using the high-accuracy Virtual Environment for Reactor Applications (VERA). By contrasting the two core designs, the study identified strategies to mitigate the risk of fuel fragmentation and enhance economic viability. The effectiveness of the PSA algorithm in rapidly evaluating thousands of core designs highlights its potential for optimizing reactor performance, safety, and feasibility.

In summary, US-NRC strongly recommend to pursue research addressing various optimization aspects of future US fuel management schemes.

### E.3 FFRD related acceptance criteria used in various countries.

Several countries operating LWRs, similar to the United States, have acknowledged the issue of FFRD. Consequently, FFRD considerations have been incorporated into the safety review approval processes. Each nation has adopted distinct approaches to integrate FFRD into their safety evaluation frameworks, ensuring the integrity and reliability of nuclear fuel under accident conditions. A summary of the available information through 2022 regarding various countries is presented in the Table E-1 below:

Table E-1: FFRD consideration in safety review in several LWR operating countries.

Country	FFRD Consideration in Safety Review	Specific Burnup Limit Due to FFRD
United States	Issue not explicitly addressed in 10 CFR 50.46c; however, expected in a comprehensive safety assessment.	Informal limit around 62 MWd/kgU; considering increases up to 75-80 MWd/kgU.
Finland	STUK includes FFRD in safety assessments, especially for high burnup fuels.	Limits around 55-62 GWd/tU.
Sweden	FFRD considered in safety assessments; aligned with specific SSMs recommendations.	Firm limits not detailed in regulations; the 60 MWd/kgU constitutes a soft-limit for operation and design
France	No-burst approach in LOCA analysis; indirectly considers FFRD.	No specific FFRD-related burnup limit although NFIR-6 FFRD threshold has been

		endorsed by the safety authorities; general safety evaluations.
<b>Japan</b>	FFRD is under consideration; not yet a specific regulatory requirement.	No specific burnup limit solely due to FFRD.
<b>Russia</b>	FFRD addressed in LOCA analyses; ROSATOM oversees evaluations.	Specific limits vary by reactor type and fuel configuration.
<b>Spain</b>	Consideration of FFRD is implicit in safety reviews, particularly for newer fuel designs.	No specific burnup limit defined for FFRD; general safety margin requirements apply.
<b>United Kingdom</b>	FFRD is a consideration in safety assessments, particularly for advanced gas-cooled reactors (AGR) and light water reactors (LWR).	No specific FFRD-related burnup limit; comprehensive evaluations on a case-by-case basis.
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Countries such as France and Sweden have incorporated FFRD considerations into their nuclear safety evaluation frameworks to enhance the robustness and reliability of nuclear fuel under accident conditions. Their regulatory bodies, the Autorité de Sûreté Nucléaire (ASN) and the Swedish Radiation Safety Authority (SSM), have developed guidelines based on extensive research, thus integrating FFRD findings into safety assessments to support the sustained safe operation of nuclear power plants.

In general, evaluating the inclusion of FFRD in safety reviews, two primary strategies have emerged. One strategy involves establishing a stringent limit on fuel burnup to mitigate the dispersal of fine fuel fragments, with the U.S. NRC currently favoring this path by maintaining a fuel burnup limit of 62 MWd/kgU and refined cladding embrittlement criteria as outlined in 10 CFR 50.46. However, they allow certain fuel damage limited only by the requirements set by radioactive release analyses (dose analyses) of LOCA. Conversely, France and Sweden employ a strategy defined by specific operational conditions that aim to eliminate the risk of fragment dispersal, often referred to as the "no-burst" approach.

This methodology allows for reactor operation with licensed burnup levels and offers the potential for cycle extension without encountering FFRD-related challenges. The Swedish approach exemplifies this strategy and will be elaborated upon in subsequent sections.

## E.4 A no-burst approach: the Swedish example

The Swedish Radiation Safety Authority (SSM) has evaluated the safety implications of high-burnup nuclear fuel fragmentation during LOCA. The assessment concluded that while the safety significance of such fragmentation is substantial enough to warrant safety evaluations for facilities planning to operate with high burnup levels, the current risk does not justify implementing strict burnup restrictions. Instead, SSM provides recommendations rather than regulations to licensees regarding this issue, as detailed in a lower-level document ("Promemoria") [SSM2014-4208-7, SSM2014-4208-5]. Licensees aspiring to achieve high burnup in nuclear fuel rods are required to demonstrate that safety standards are maintained and that potential consequences from pellet fragmentation do not worsen conditions during LOCA events. A critical aspect of this evaluation involves ensuring that the core retains the prescribed coolable geometry, as outlined in Section 26 of SSMFS 2008:17 [SSM 2008]. The SSM designates a burnup threshold of 60 MWd/kgU for fuel rods, beyond which FFRD phenomena need to be considered by licensees. This level for average rod burnup serves as an operational and design soft limit (see Figure D-16 in Appendix D -). Exceeding this threshold necessitates further risk assessments related to fuel fragmentation and dispersal, in addition to the standard fuel licensing and safety analysis protocols. The SSM advises even starting the assessment with slightly lower burnups than those resulting in fine fragmentation.

To align with the SSM recommendations, the Swedish nuclear industry initially implemented reactive cycle management adjustments to remain below the 60 MWd/kgU burnup threshold, albeit at the cost of cycle efficiency. Over the long term, new LOCA analyses were performed for Ringhals units 3 and 4 utilizing the Full Spectrum LOCA (FSLOCA) methodology developed by Westinghouse. This advanced methodology addresses a broad spectrum of break sizes and employs a reference fuel design concept, (kind of "dummy fuel type" allowing to cover other fuel types falling within analysed parameter ranges) with pre-selected parameter ranges relevant to the statistical method used for the uncertainty analysis

applied by FSLOCA. Although the details of this innovative approach are restricted due to Westinghouse's proprietary rights, the FSLOCA analyses were further enhanced by an accompanying operator stand-alone methodology that effectively demonstrates reactor safety and coolability.

This new methodology has been developed in collaboration between Ringhals NPP and Vattenfall Nuclear Fuel AB [Holmström et al, 2023] to address the issue of cladding ballooning and burst during a LOCA when fuel pellets experience fine fragmentation due to high burnup. This methodology aims to demonstrate how these issues can be avoided, especially above a certain burnup threshold. For this purpose, a burnup threshold of 55 MWd/kgU has been selected based on the most current data compilation [Bales et al, 2021], which indicates the onset of fine fragmentation. As a pragmatic upper bound for Ringhals PWRs, a rod average burnup limit of 50 MWd/kgU has been adopted for the analysis.

The fuel licensing framework for Ringhals PWRs permits a maximum rod average burnup of 72 MWd/kgU; however, operational cycles typically do not exceed 65 MWd/kgU. The evaluation presented in [Holmström et al, 2023] was aimed at assessing reactor performance up to end-of-life conditions. To identify the most critical Loss-of-Coolant Accident (LOCA) scenarios, a comprehensive analysis was performed that considered the specific characteristics of each reactor, including limiting break opening areas and their respective positions. This analysis utilized the TRACE code [USNRC 2017], incorporating conservative assumptions for key parameters in line with Option 2 of the International Atomic Energy Agency (IAEA) safety standard for deterministic safety analysis, IAEA-SSG-2 [IAEA 2009].

Table E-2: Options for Combination of a Computer Code and Input Data

Option	Computer code	Availability of systems	Initial and boundary conditions
1. Conservative	Conservative	Conservative assumptions	Conservative input data
2. Combined	Best estimate	Conservative assumptions	Conservative input data
3. Best estimate	Best estimate	Conservative assumptions	Realistic plus uncertainty; partly most unfavourable condition?
4. Risk informed	Best estimate	Derived from probabilistic safety analysis	Realistic input data with uncertainties*
* Realistic input data are used only if the uncertainties or their probabilistic distributions are known. For those parameters whose uncertainties are not quantifiable with a high level of confidence, conservative values should be used.			
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The analysis and sensitivity studies confirmed that the intermediate sized break in the cold leg poses the greatest risk.

Further step in this “no-burst” approach was to determine a maximum cladding temperature where no rupture can occur. Commonly used correlations of rupture temperature as a function of hoop stress over the cladding wall can be found in NUREG-0630 [USNRC 1980] but in Ringhals analysis [Holmström et al, 2023], a more cladding specific variant was used in order to cover hydrated and irradiated material. The general trend was preserved and a negative bias was added to cover the absolute majority of the data cloud.

Existing calculations of pressure as a function of fuel rod burnup for Ringhals specific fuel designs were used and uncertainty was added in the direction providing a smaller margin to cladding burst. The pressure buildup in the rods for high burnup fuel i.e. potential transient fission gas release was considered using a simplistic model. Combining the curves of rupture temperature as function of hoop stress and hoop stress as function of burnup, a curve of rupture temperature as function of burnup is determined. The resulting curve is shown in Figure E-2 and can be used as an acceptance criterion for demonstration of avoidance of burst for high burnup fuel.

## Appendix F - LOCA testing methodologies

In the following sections the Separate effects tests, Tests with moderate integration and Integrated tests will be described.

### F.1 Separate effect tests

#### F.1.1 Introduction

Table F-1 describes some of the tests and their objectives further discussed in the following sections.

Table F-1: This table summarises the different separate effect tests and their main objectives described more in the subsequent sections.

Type of Test/Examination	Main Goal	Covered in Section:
Creep Test	The creep properties of cladding alloys are central to modelling the ballooning deformation and burst failure of cladding tubes during a LOCA	Section F.1.2
Oxygen Diffusion	Modelling of the growth of the oxide layer and the oxygen stabilised Zr $\alpha$ layer (impacts fuel clad ductility)	Section F.1.3
Metallographic evaluation, calorimetry, dilatometry, and image analysis on elemental X-ray maps	Determination of the Zr $\alpha$ to $\alpha+\beta$ and $\alpha+\beta$ to $\alpha$ transformation temperatures (impacts fuel clad ductility during ballooning)	Section F.1.4
Ductility (RCT, Ring Tensile Test, 3- and 4-point bending tests, axial tensile testing)	Assess PQD during different loading conditions	Section F.1.5
Hardness testing	Measurement of the macro-hardness of materials is a quick and simple method (compared to tensile and creep testing etc., and only if done correctly) of obtaining mechanical property data for the bulk material from a small sample	Section F.1.6
Impact Testing	Fuel cladding performance during rapid deformation rates such as e.g. fuel rod quenching at the end of the LOCA sequence, or seismic loads following a LOCA event	Section F.1.7

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#### F.1.2 Cladding Creep Testing

The creep properties of cladding alloys are central to modelling the ballooning deformation and burst failure of cladding tubes during a LOCA. As an example, Chung and Kassner reported tests in vacuum and steam of Zircaloy cladding heated with alternating current [Chung H.M. et al, 1976]. Their specimen design is shown in Figure F-1.

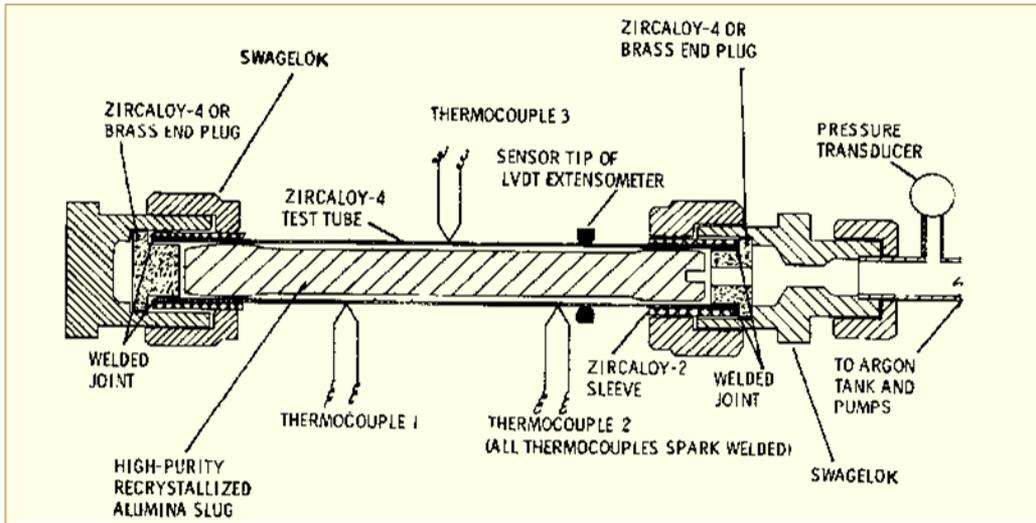


Figure F-1: The 15.2 cm long specimen used by Chung and Kassner. Modified figure according to [Chung H.M. et al, 1976]

The diametral expansion was monitored by an extensometer seen in the figure. The specimen was also photographed with a high-speed camera during the test for post-test evaluation of diametral expansion and axial deformation. The experiments were carried out in a vacuum chamber with or without steam flow. Several different heating rates were used, from 5 K/s to 115 K/s. Figure F-2 shows the rupture strain of unconstrained tubes tested in vacuum as a function of burst temperature, which provides baseline burst behaviour for unoxidized cladding. The maximum circumferential strain is partly dependent on the temperature, being larger in the  $\alpha$  Zr and  $\beta$  Zr phases than in the  $\alpha+\beta$  Zr phase, see also Figure F-6.

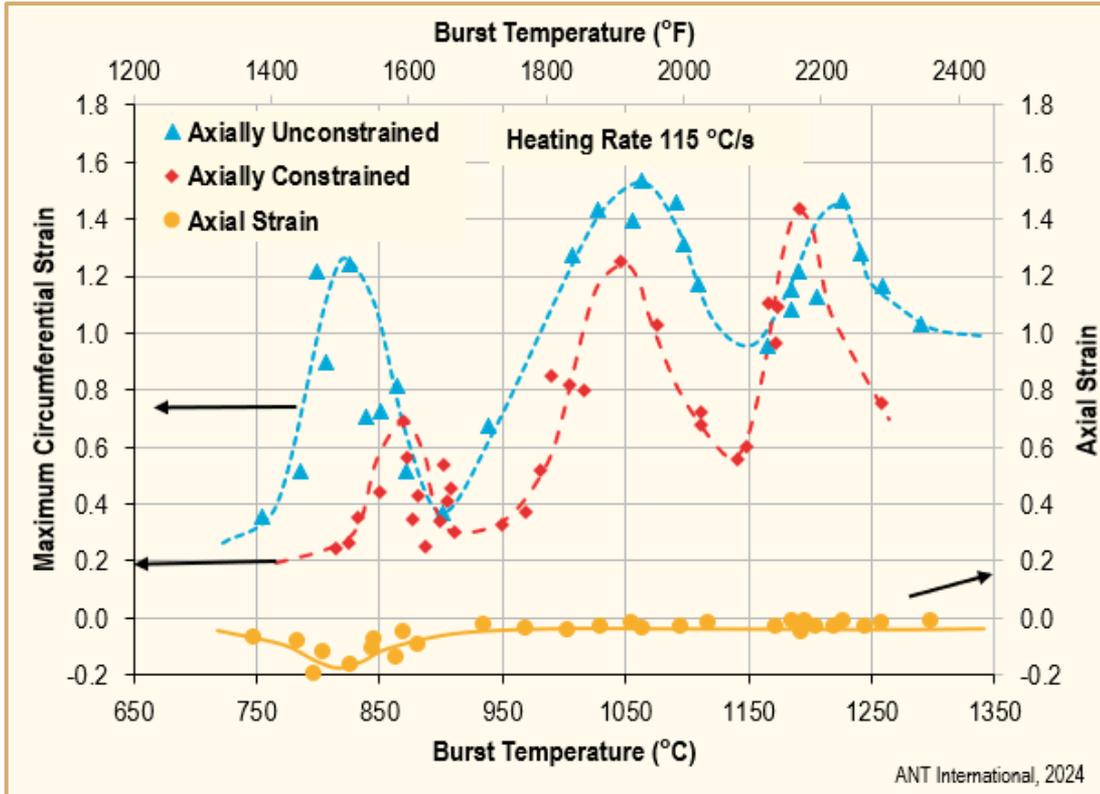


Figure F-2: Maximum circumferential strain at rupture vs burst temperature for axially constrained and unconstrained Zircaloy-4 cladding at a heating rate of 115 K/s. The axial strain of unconstrained tubes is also shown, modified figure according to [Chung H.M. et al, 1976].

Rizkalla *et al.* studied the effect of oxygen concentration on the high temperature creep of Zircaloy-2 fuel cladding [Rizkalla et al., 1978]. One of the conclusions of the work was that the effect of oxygen on strength is twofold in the  $\alpha + \beta$  Zr region: the oxygen contributes to solid solution strengthening and it also increases the volume fraction of  $\alpha$  Zr.

Creep tests showed that hydrogen ranging from 600 to 1000 ppm in the cladding in the temperature range of 700 to 850 C lowers the cladding creep ductility, Figure F-3. During a LOCA situation it would mean that hydrogen in the fuel cladding will lower the burst temperature. *This expected effect means that with increasing hydrogen cladding contents more fuel rods will burst during a hypothetical LOCA increasing the source term. The reason being that hydrogen decreases the  $\alpha/\alpha+\beta$  phase transformation temperature, which means that increasing the hydrogen content in the fuel cladding will lower its ductility (ductility is lower in the two phase than in the one phase region) and result in more Fuel Rods (FR) burst during a LOCA, Figure F-4.*

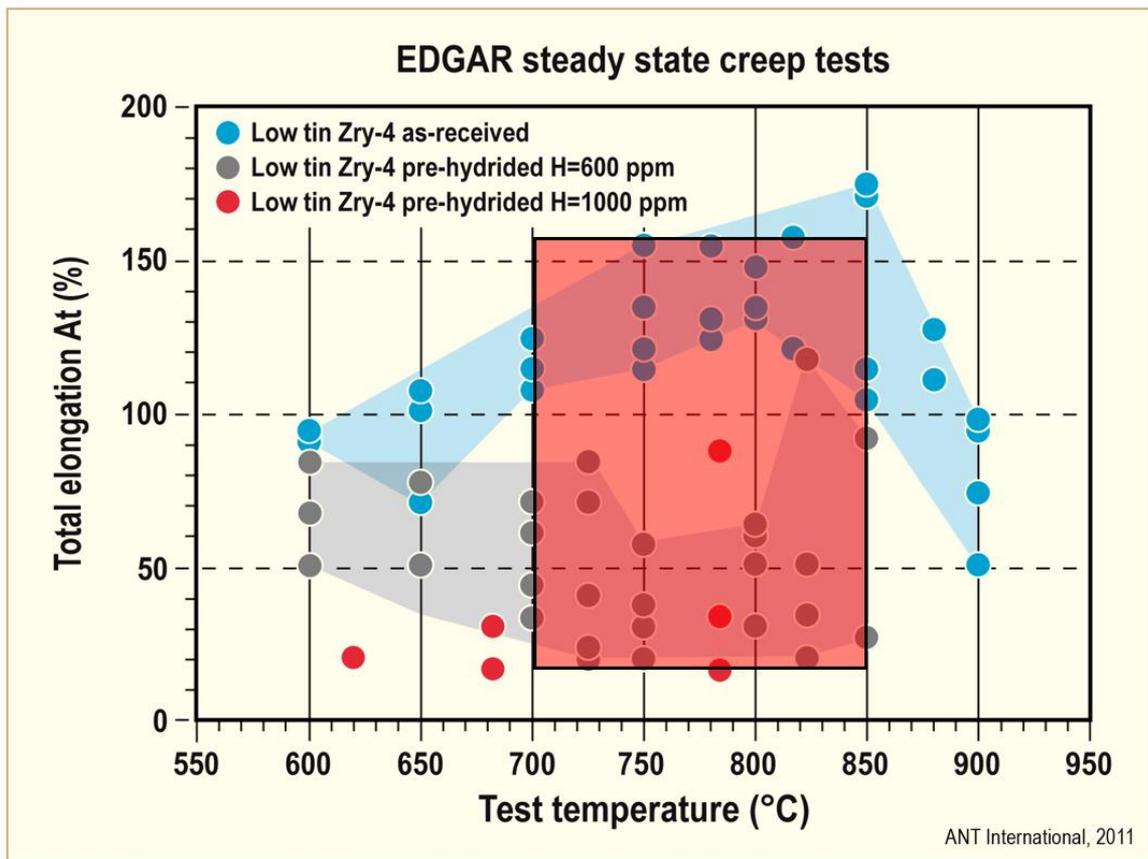


Figure F-3: Isothermal creep test results of unirradiated Zry-4. The red box shows the test temperature range where the hydrogen additions will reduce ductility, , modified figure according to [Waeckel et al., 2004].

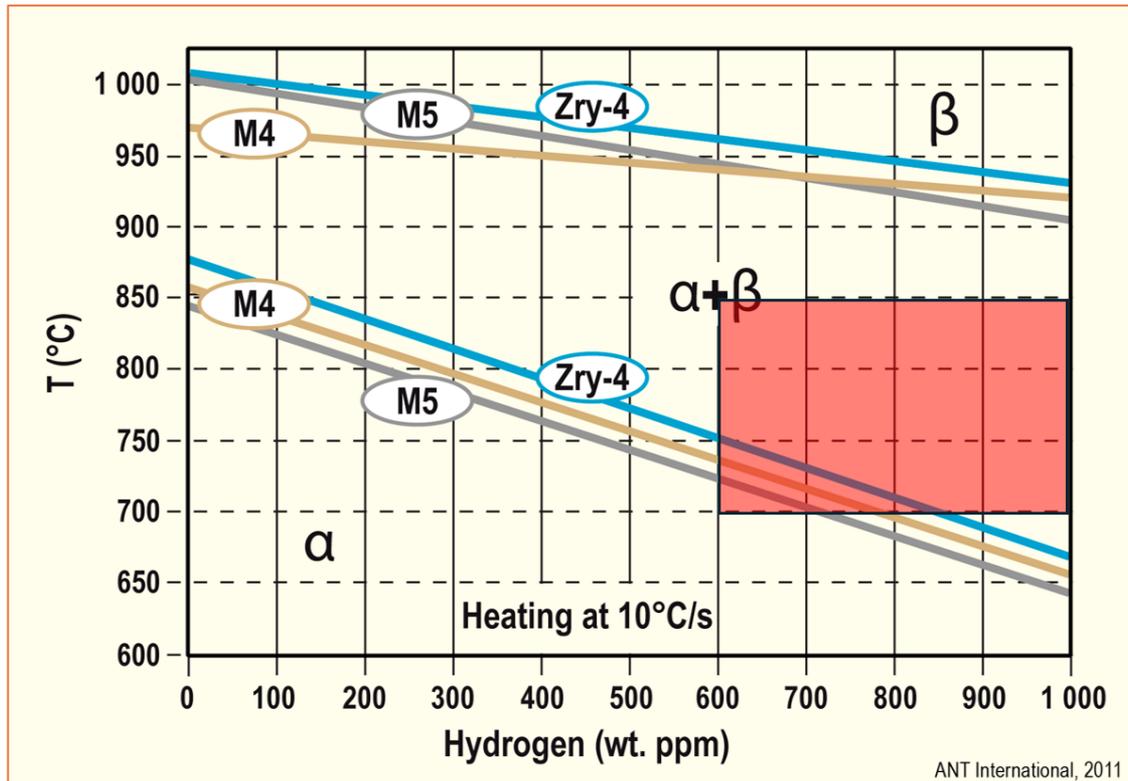


Figure F-4: Scheme of the  $\alpha/\beta$  phase transformation temperatures of Zry-4, M4 and M5 alloys as a function of the hydrogen content for a heating rate of 10 K/s. The red box indicates the Zr phases in the temperature range of 700 to 850°C, see also Figure F-3, modified figure according to [Brachet et al, 2002].

### F.1.3 Oxygen diffusion constants

The ductility of the fuel rod cladding depends on the thickness of  $ZrO_2$ , Zr  $\alpha$  and Zr  $\beta$  layers. Only the Zr  $\beta$  layer is ductile while both  $ZrO_2$  and Zr  $\alpha$  are brittle. Modelling of the growth of the oxide layer and the oxygen stabilised  $\alpha$  layer requires knowledge of the diffusion coefficients of oxygen in the two phases. In addition, it is necessary to know the diffusion coefficient in the  $\beta$  layer in order to calculate how fast the  $\alpha$  grows into the  $\beta$ . [Pawel et al., 1976] determined the diffusion coefficient in  $\beta$  phase by studying the diffusion of the  $^{18}O$  isotope.

### F.1.4 Determination of the $\alpha$ to $\alpha+\beta$ and $\alpha+\beta$ to $\alpha$ transformation temperatures

Assessment of the  $\alpha$ ,  $\alpha+\beta$  and  $\alpha+\beta$  to  $\alpha$  transformation temperatures are important to enable modelling of fuel cladding circumferential strain during the ballooning phase of the LOCA, see Figure F-5.

The temperatures, when  $\beta$  starts to transform to  $\beta$  and when the transformation to  $\beta$  is complete, depend on the alloying elements in zirconium. The pure Zr metal transforms at 862°C but the addition of an alloying element will change this transformation temperature. The addition of an alloying element either increases or decreases the transformation temperature. If it increases the temperature the element is called an  $\beta$  stabiliser. Examples are O which is a strong  $\beta$  stabiliser and Sn. Examples of  $\beta$  stabilisers are most other elements including common alloying elements like Nb, Fe, Cr and Ni and inadvertently added elements like H.

In several earlier works [Rizkalla, A et al. 1978], [Chung & Kassner, 1979] the transformation interval for Zircaloy has been charted by metallographic evaluation of the volume fraction of  $\beta$  after fast cooling from a

known temperature. Figure F-5 shows equilibrium phase boundaries reported by Chung and Kassner from examination of quenched metallographic specimens of Zircaloy-4 [Chung & Kassner, 1979].

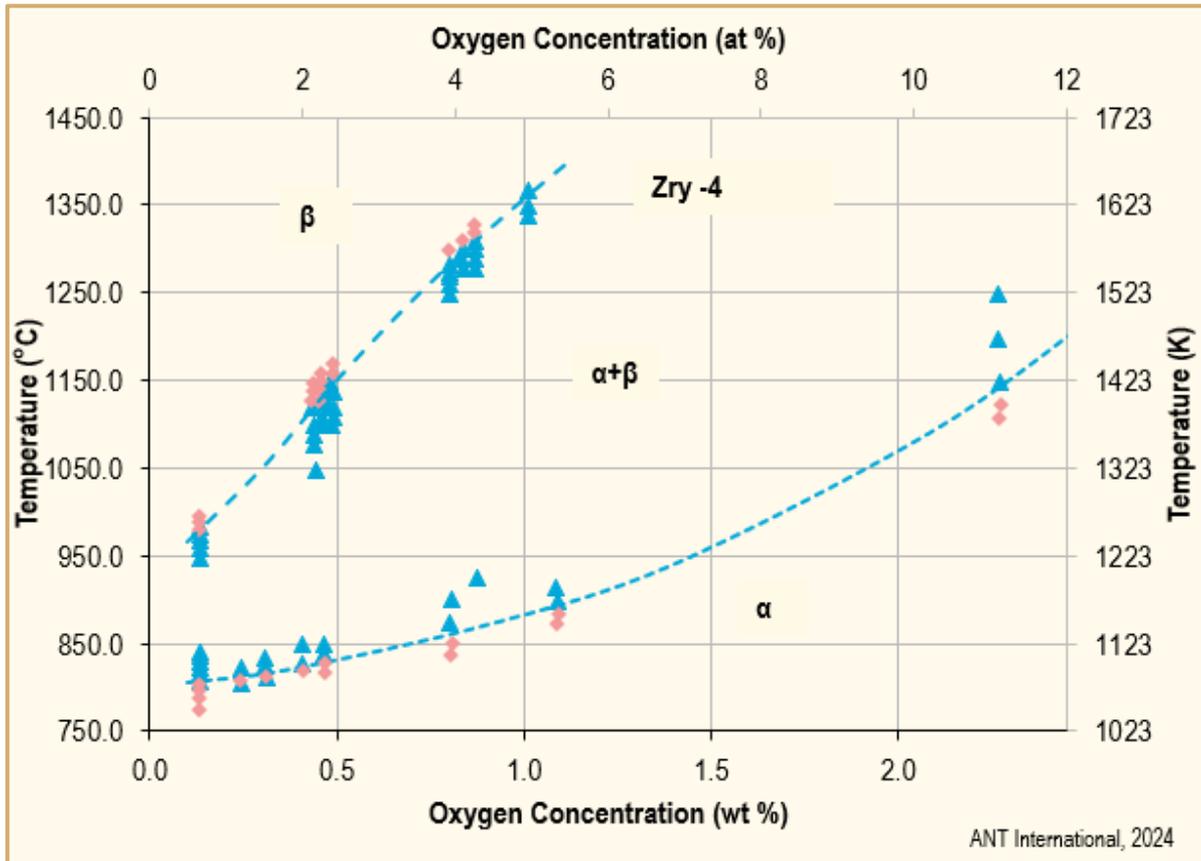


Figure F-5: Oxygen-rich-side Zircaloy-4-oxygen phase diagram determined from metallography of quenched samples, , modified figure according to [Chung & Kassner, 1979].

Forgeron *et al.* studied the transformation temperatures for Zircaloy-4, M4, and M5 alloys [Forgeron, T., et al., 1998]. Their methods were calorimetry, dilatometry, and image analysis on elemental X-ray maps. The same techniques were used in a study by Brachet *et al.* on the effect of hydrogen on the phase transformation of the same three alloys [Brachet, J.-C., et al., 2001]. Figure F-6 shows the effect of hydrogen on the transformation temperatures of Zircaloy-4.

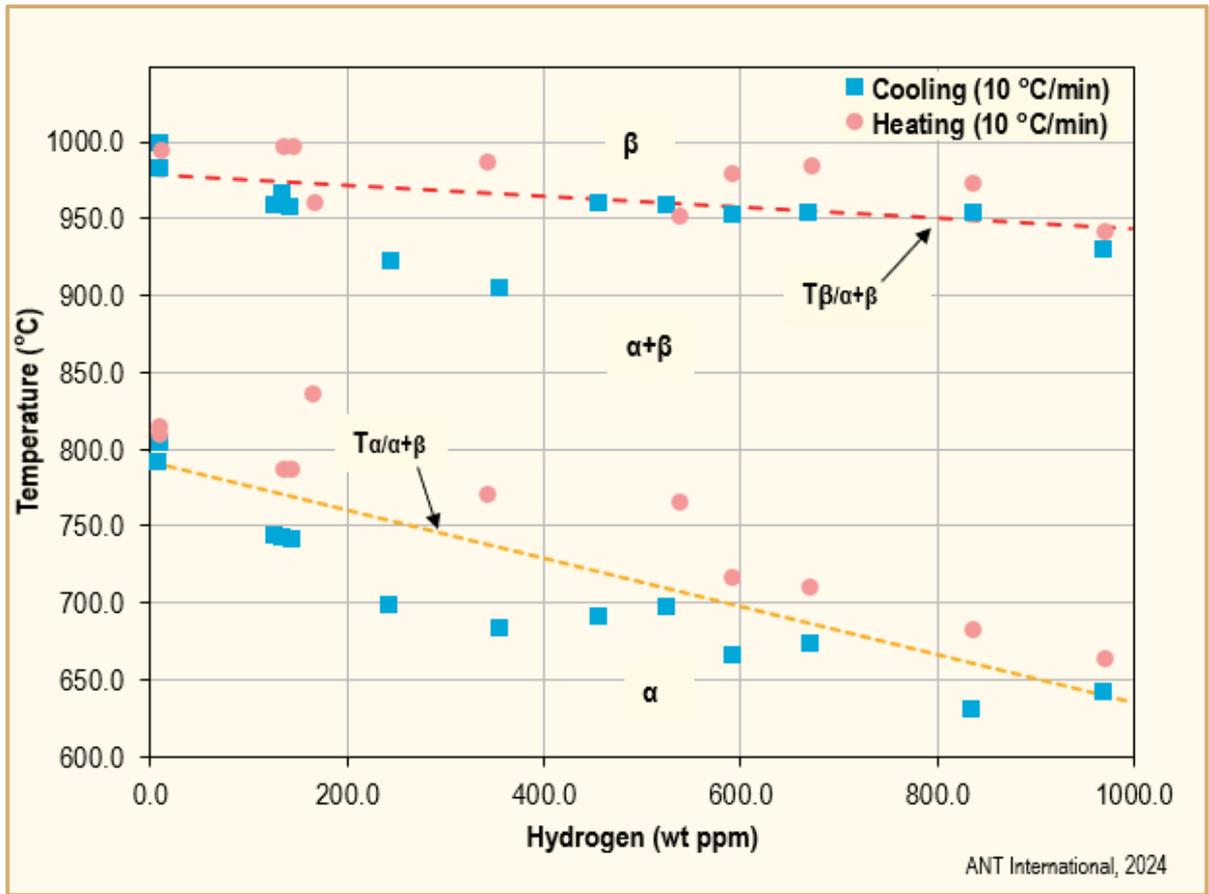


Figure F-6: Plot of  $T_{\alpha/\alpha+\beta}$  and  $T_{\beta/\alpha+\beta}$  temperatures measured upon heating and cooling at 10 °C/min as a function of hydrogen content. The straight lines (average between heating and cooling) are assumed to be representative of equilibrium temperatures, modified figure according to [Brachet, J.-C., et al., 2001].

### F.1.5 Post-LOCA or Post-Quench Ductility (PQD) Tests

Before we start to review different mechanical tests, it is important to identify the difference between ductility and toughness shown in Figure F-7.

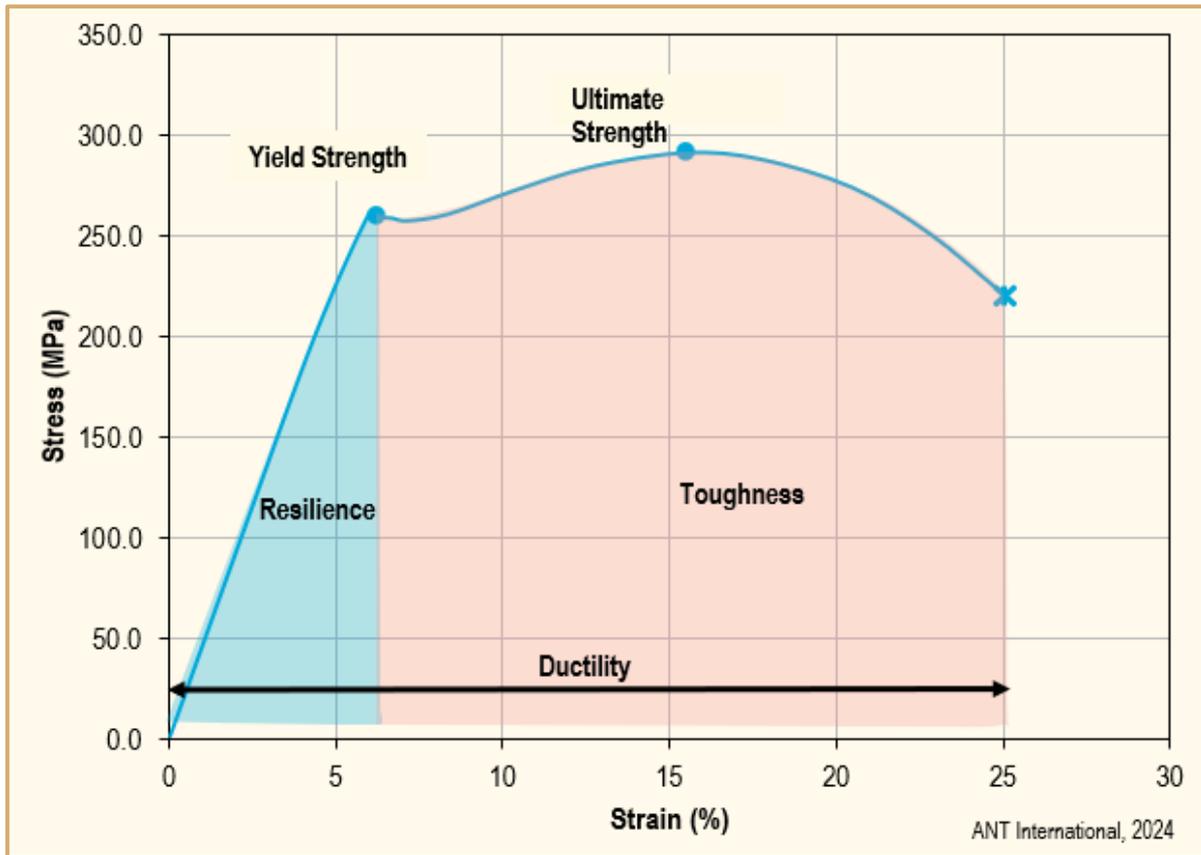


Figure F-7: Showing the difference between ductility and toughness measured in tensile test.

### F.1.5.1 Post-LOCA Ring Compression Test (RCT)

PQD is typically assessed in tests of tubular samples at RT and/or 135°C (275°F) that have been subjected to a HT oxidation in presence of steam. Mechanical tests used includes ring compression, ring tension, tube bending and impact [OECD, 2009] and [Brachet et al, 2007]. Tests based on compression, tension and bending involve relatively slow loading and a validated method to identify conditions for the loss of ductility through load-deformation data. Impact tests involve rapid loading and a protocol to relate ductility with impact energy, see Section F.1.7. In all tests of PQD, the criterion is brittle versus ductile fracture. RCT are commonly used and are discussed in this section. The RCTs and the related brittleness criterion have the advantage of being consistent with the experimental basis of current LOCA criteria.

The 17% ECR limit was derived almost 4 decades ago by a specific set of test conditions reported by [Hobson & Rittenhouse, 1972]. The simple RCT was used to produce the key data, and the test is still used today. Early on, efforts were made to see if some other test or modification was more representative [Meyer, 2003]. Figure F-8 gives the basic geometry and Figure F-9 indicates the general stresses. Most often 2 cracks form on the outside surface at the 3 and 9 o'clock positions and 2 others start on the inside surface at the 6 and 12 o'clock positions. Adding curved loading plates, Figure F-9 (lower) reduces the tendency for fracture at 6 and 12 o'clock. The load-displacement curve produced is complicated because cracks often do not propagate through way due to the reversal of stress at the mid-wall. The recommended analysis method is shown in Figure F-10 where the defined critical strain parameter is the plastic strain at the time of first crack formation. Another approach is to use the "energy at failure" (similar to strain energy density used in RIA analysis) as the indication of the ductile – brittle transition [Hózer & Gyori, 2004], (Figure F-11).

## Appendix G - Structured frameworks for developing, validating, and quantifying uncertainties in BE LOCA analysis methods

### G.1 CQD

The Code Qualification Document (CQD) methodology [Bajorek et al, 1998] is a conservative approach traditionally used in the nuclear industry for the safety analysis of nuclear power plants. This methodology employs conservative assumptions and safety margins to ensure that all potential risks associated with nuclear reactor operations and accidents are adequately covered.

CQD methodology was developed primarily in the United States during the 1970s and 1980s. It emerged in response to the need for rigorous safety standards in the nuclear power industry, particularly after significant incidents like the Three Mile Island accident in 1979, which underscored the importance of robust safety measures. The main goal of the CQD methodology was to establish a safety analysis framework that would be inherently conservative to ensure that nuclear reactors could withstand a range of postulated accident scenarios without leading to catastrophic failures. This approach was intended to provide a wide safety margin by assuming worst-case scenarios in system behaviours and failure modes. In CQD, parameters related to reactor operation and accident scenarios are set to their most conservative potential values. This includes high estimates of reactor power, conservative predictions of heat generation, and minimal effectiveness of cooling systems.

The regulation in 10 CFR 50.46 identified five following criteria:

- (1) Peak clad temperature (PCT) should be less than 1204 C.
- (2) Local maximum oxidation (LMO) should be less than 17%.
- (3) Core-wide oxidation (CWO) should be less than 1% (to limit the maximum hydrogen generated).
- (4) The core should maintain a coolable geometry.
- (5) Long-term cooling should be demonstrated.

In early criteria applications to ECCS assessment, the last two criteria—coolable geometry and long-term cooling—were generally deemed satisfied outside the LOCA analysis, provided that the LOCA calculations met the first three criteria. However, in more recent evaluations, even when conservative analysis methods are employed, this approach has proven insufficient due to the involvement of high-burnup fuel.

At that time, regulations were designed to address potentially unknown phenomena and acknowledged the limited understanding of fundamental physical processes. Appendix K to 10 CFR 50 required several very conservative features like: decay heat calculations based on the ANS model from 1971 [ANS 1971] with an additional 20% margin; the metal-water reaction calculated using the conservative Baker-Just model [USNRC 2023] and heat transfer considerations restricted to steam only for low-flooding rates. Additionally, safety margins in the CQD are intentionally large to accommodate uncertainties in data, modelling, and reactor physics. These margins ensure that the reactor core remains cool and intact, even if multiple safety systems fail.

CQD [Bajorek et al, 1998]] has been widely used for licensing and regulatory compliance in the nuclear industry, particularly in the United States and abroad. It forms the basis for many of the deterministic safety analyses that nuclear plants must perform to demonstrate their ability to safely shut down in the event of an emergency. CQD methodologies have been integrated into the regulatory frameworks of several countries operating nuclear power plants, influencing the way nuclear safety is approached.

Over time, the nuclear industry and regulatory bodies have been moving towards more risk-informed approaches that incorporate probabilistic risk assessment (PRA) techniques. These modern methodologies, such as the CSAU (Code Scaling, Applicability, and Uncertainty) approach, build upon the conservative foundations of CQD but aim to provide a more knowledge-based and realistic LOCA analysis method by quantifying uncertainties rather than just conserving against them.

## G.2 CSAU

CSAU stands for Code Scaling, Applicability, and Uncertainty, a methodology developed to enhance the reliability and accuracy of predictions made by thermal-hydraulic safety analysis codes used in nuclear reactor safety evaluations, specifically in ECCS performance assessment (i.e. LOCA analysis). The CSAU was developed by the USNRC during the late 1980s as a part of an effort to implement more realistic and scientifically-based techniques in regulatory decision-making regarding ECCS performance assessment. The primary goal of CSAU was to improve the understanding and treatment of uncertainties in the simulation results of nuclear reactor safety codes. CSAU can be viewed as an evolutionary step beyond the conservative design principles embodied in CQD [cf. Chapter G.1]. While CQD focuses on conservatism without detailed quantification of uncertainties, CSAU explicitly addresses these uncertainties, assessing how they impact the safety analysis.

Historically, the USNRCs worked in late 1980's with amendment of the 10 CFR 50.46 rule and in 1989 issued an important Regulatory Guide 1.157 [USNRC 1989] which together with the experience gained by the industry in a more realistic approach in the ECCS assessment collected in sort of LOCA-compendium [USNRC 1988] encouraged the use of realistic models to perform safety analyses by means of best-estimate computer codes (cf. Federal Register [FR-1988]).

The CSAU methodology [Boyack et al, 1989] after extensive review and many requests for additional information has been in 1996 approved by USNRC. Although the same 10 CFR 50.46 rules and acceptance criteria for ECCS design apply, this methodology initiated a shift from conservative, deterministic methods (like CQD). The methodology must still meet the five acceptance criteria outlined in Chapter G.1, but the CSAU approach allows flexibility in determining which attributes will be subjected to rigorous uncertainty assessment.

The first approved best-estimate LOCA evaluation model [Boyack et al, 1989], [Young et al, 1998] in its original version was applicable to Westinghouse's 3- and 4-loop plants with safety injection into the cold leg. Subsequently, the methodology applicability was extended to 2-loop plants with upper plenum injection (UPI) [Takeuchi & Nissley, 2000] and advanced passive plant such as the AP600 and AP1000 [Frepoli et al, 2003]. Since its approval, Westinghouse has applied the methodology to several dozen nuclear power plants both in the USA and abroad. These regulatory guidelines directly supported the principles underlying the CSAU methodology, which aims to provide a more realistic and strictly mathematically derived assessment of safety margins. This methodology systematically addresses the scaling, applicability, and uncertainty of code calculations, providing a structured approach to quantifying the confidence in safety assessments performed using these codes.

The CSAU methodology is designed with a general structure comprising three main elements and 14 distinct steps. This framework serves as a foundation for implementing LOCA analysis across various plant designs. Each user may adapt the methodology to fit their specific codes, models, documentation, and validation data, making it a versatile tool rather than a finalized or closed system. It's important to note that the methodology is protected by intellectual property rights held by both its creator and often the operator of the NPP involved. Consequently, the complete details of the CSAU steps cannot be provided here, and the following description will be simplified and generalized.

- E1. Requirements and Code Capabilities - provide the fundamental technical basis for decisions downstream in the methodology development process. Steps 1, 2, 4, and 5 identify the problem through specification of the event scenario, plant type, computer code and version, and computer code documentation, respectively. Step three of the first element is crucial to the whole methodology as in it potentially important phenomena/processes are identified for the selected plant type and scenario. A panel of experts performs the ranking and documents the basis for consensus. The results are compiled into a Phenomenon Identification and Ranking Table (PIRT).

The PIRT is a key component of the CSAU-based methodology. It is designed to prioritise code evaluation and facilitate decision-making regarding the development of the physical model and methodology.

- (1) Specify Scenario - selection of the transient to be analysed, determines processes that must be addressed (e.g. LB LOCA scenario).
  - (2) Select plant - selection of the plant type, determines many type-specific details regarding construction, phenomena, interactions (e.g. W 3-loop plant).
  - (3) Identify and Rank Phenomena – a group of experts (referred to as the technical program group or TPG) under the sponsorship of the USNRC has, by reaching a certain consensus, created a prototype of PIRT. Now experts knowledgeable in selected scenario (LBLOCA) prepare PIRT identifying and ranking the important phenomena (relative importance) for the specified scenario and plant type. This step is particularly important due to the fact that the PIRT provides the basis for determining used code applicability, determine assessment matrix and important phenomena to be qualified and ranged for evaluating uncertainties. For these reasons this step undergoes usually a thorough peer review.
  - (4) Select Frozen Codes – selected computer codes with frozen versions and releases are used throughout whole process for consistency (e.g. WCOBRA/TRAC with given MOD and Revision in ASTRUM or S-RELAP5 and RODEX3A with given code versions in RLBLOCA).
  - (5) Provide Complete Documentation – which must be consistent with the frozen code versions including models and correlations, programmers guide and user manuals.
  - (6) Code Applicability Determination – confirms that code models of phenomena identified as important in PIRT (step 3) are present and sufficiently documented.
- E2. Code Scaling and Applicability - the second phase involves determining the scaling laws that govern the physical phenomena identified in the PIRT phase. It assesses how well the computational models in safety codes can be applied to the full-scale reactor scenarios (applicability), ensuring the code's predictions are relevant to actual reactor conditions.
- (7) Establishment of Assessment Matrix – select assessment matrix of separate and integral effect tests (SET/IET) in support of code evaluation for identified in PIRT important phenomena (e.g. validation of selected plant nodalization, scaling and models uncertainty considerations).
  - (8) Define Nodalization - select consistent noding used for code validation against experimental data (SET/IET) and for selected plant analysis. Selected nodalization should preserve dominant phenomena, minimize code uncertainty, support selected plant characteristics and remain numerically efficient.
  - (9) Determine Code and Experiment Accuracy – determine individual parameter uncertainty or uncertainty distribution for identified important PIRT phenomena. Uncertainty and biases determined by comparison of code results to SET and IET experiments defined in Assessment matrix.
  - (10) Determine Effect of Scale – Potential code scaling effects must be quantified for bias and deviation. Scalability of tests used to demonstrate code models applicability should be verified. Scale effects addresses several issues depending on selected plant, two are crucial scalability of the tests relevant for selected plant and code models from the tests to the selected plant.

E3. Uncertainty Evaluation – the element is probably the most straight forward of all the element focusing on quantifying the uncertainties associated with dominant contributors and their probability distributions when properly identified and quantified, and if the computer code, through assessment and comparison with data, is shown to accurately predict the effect of variations in input variables on the output result, then several well-established methods are available to perform the uncertainty propagation step. The choice of method is basically a practical one, controlled by the expense incurred in performing computer calculations. The methods utilized evolved over the last few decades. A short information on the methods for combining the uncertainties is given in step 13.

- (11) Determine Effect of Reactor Input Parameters and State – plant specific variations in input and process parameters are determined based on plant data or analytical studies. Analysis must address the Technical Specification limits for those parameters impacting the selected scenario (e.g. LBLOCA).
- (12) Perform Plant Sensitivity Calculations – the code’s output sensitivity to the plant specific input and process parameter variations is determined from sensitivity studies using selected plant model. These plant specific input and process parameters that affect the code results should in an appropriate manner for the chosen statistical method be included in the final statistical estimation. For instance, in modern LOCA analysis methodologies like ASTRUM [Nissley et al, 2003 , Frepoli, Oriani, 2006] or RLBLOCA [EMF-2103 2001, EMF2103 2016]) the identified uncertainty contributors are randomly sampled for each case. Break type, split break size, and time in cycle are also sampled as uncertainty contributors.
- (13) Combine Biases and Uncertainties – the objective of this step is to combine the bias and uncertainty of the important individual contributors identified in Step 9 and 11 through the execution of a large set of the for plant selected accident scenario simulations. Statement for the limiting values of the primary safety criteria (LOCA) at the in statistical method selected (determined) confidence level is expressed as upper bound values of the 95th percentile established at 95-percent confidence using non-parametric statistics (e.g. in ASTRUM the 95/95 estimates regard PCT, LMO, and CWO)

E3 of the CSAU roadmap is dealing with how uncertainties are combined and propagated throughout the transient having probability distribution functions determined in E2 for all uncertainty parameters with a probability of occurrence associated to each value. The statistical methods historically selected in this phase of the CSAU have been briefly described in [Frepoli, 2008] and will only be listed here for information, and these are:

- response surface method,
- direct Monte Carlo method,
- data based code uncertainty method,
- non-parametric statistics method.

Choosing between parametric (three first methods) and non-parametric methods in CSAU depends on the available data, the level of understanding of the phenomena being modelled, and the specific requirements of the nuclear regulatory framework. Non-parametric methods, given their fewer assumptions, are generally more aligned with the conservative principles foundational to nuclear safety analysis, making them a frequent choice in handling uncertainties in critical safety parameters. These methods are derived from direct Monte Carlo approaches, but instead of trying to extract information from the underlying probability distribution function (PDF) of the measure (such as PCT), they adopt a distribution-free stance and utilize nonparametric statistics to establish a bounding value of the population at a specified confidence level. These alternative methods have in recent years begun to find applications in realistic calculations for LOCA analysis [Wickett et al,

1998] and a similar conceptually approach began in Europe in the late 1990s [Glaeser et al, 1998 and Glaeser, 2000]. More recently, in 2000s, both AREVA and Westinghouse have developed own best-estimate LOCA evaluation models that incorporated such methods. Both methodologies have been endorsed and approved by U.S. NRC. Even other industry applications of non-LOCA character appeared [Sauvage & Keldenich, 2005 and Bolger et al, 2002]. Although all these implementations share a similar technique for combining uncertainties, they differ in how the resultant calculations align with regulatory acceptance criteria.

The nonparametric statistical sampling technique is often categorized as “distribution-free.” This approach allows for the determination of tolerance limits from unknown distributions through random sampling of the variable in question. The concept of nonparametric tolerance limits was initially introduced by Wilks [Wilks, 1941], who demonstrated that the proportion of the population falling between two order statistics from a random sample is independent of the sampled population, relying solely on the specific order statistics selected. Utilizing the established Wilks formula, one can determine the necessary sample size for a particular population proportion within a defined tolerance interval. For instance, in seeking to determine a bounding value for peak clad temperature (PCT) at the 95th percentile ( $\gamma = 0.95$ ) with a 95% confidence level ( $\beta = 0.95$ ), the sample size (N) needed is derived from the equation:

$$\beta = (1 - \gamma) ^N$$

By substituting  $\gamma = 0.95$  and  $\beta = 0.95$ , the required number of computer simulations, N, is determined to be 59. In this method, all uncertainty parameters are sampled concurrently in each run, similar to the direct Monte Carlo method. Essentially, this constitutes a crude Monte Carlo simulation, utilizing the minimum number of trials necessary to stabilize the “estimator.” The results are then ranked from highest to lowest PCT, where the top rank serves as the bounding estimate for the 95th percentile PCT at the 95% confidence level.

A rigorous interpretation of the regulations would require the formulation of a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of the three attributes PCT, LMO, and CWO (for instance a 95% tolerance interval with a 95% confidence level for each). According to Wilks approach extension by Guba in [Guba et al, 2003], this requires extending the sample size beyond 59 runs, which is the minimum number sufficient when measuring only one outcome from the sample. A more general theory applicable to scenarios where more than one outcome is considered from the sample (for instance tree,  $p=3$ ) with preserved values of  $\gamma = 0.95$ ,  $\beta = 0.95$ , the computed number of computer simulations, N, is found to be 124.

This method has been implemented in the Westinghouse’s ASTRUM [Nissley et al, 2003] evaluation model and its uncertainty management approach received approval from the USNRC in November 2004. However, it’s important to note that the implementation and interpretation of order statistics in safety analysis remain inconsistent across the industry, resulting in extensive public debate among regulators and researchers, as evidenced in the open literature.

- (14) Calculate Total Uncertainty – the final step in the CSAU methodology has a provision to consider adding margin to the results of Step 13, if warranted due to limitations in the code or data base. This means if any gains or penalties were identified during the CSAU process, to the safety criteria parameters of consideration, they are to be applied in Step 14.

The flow chart of the CSAU methodology with its three main elements and 14 steps is presented graphically in Figure G-1, below.

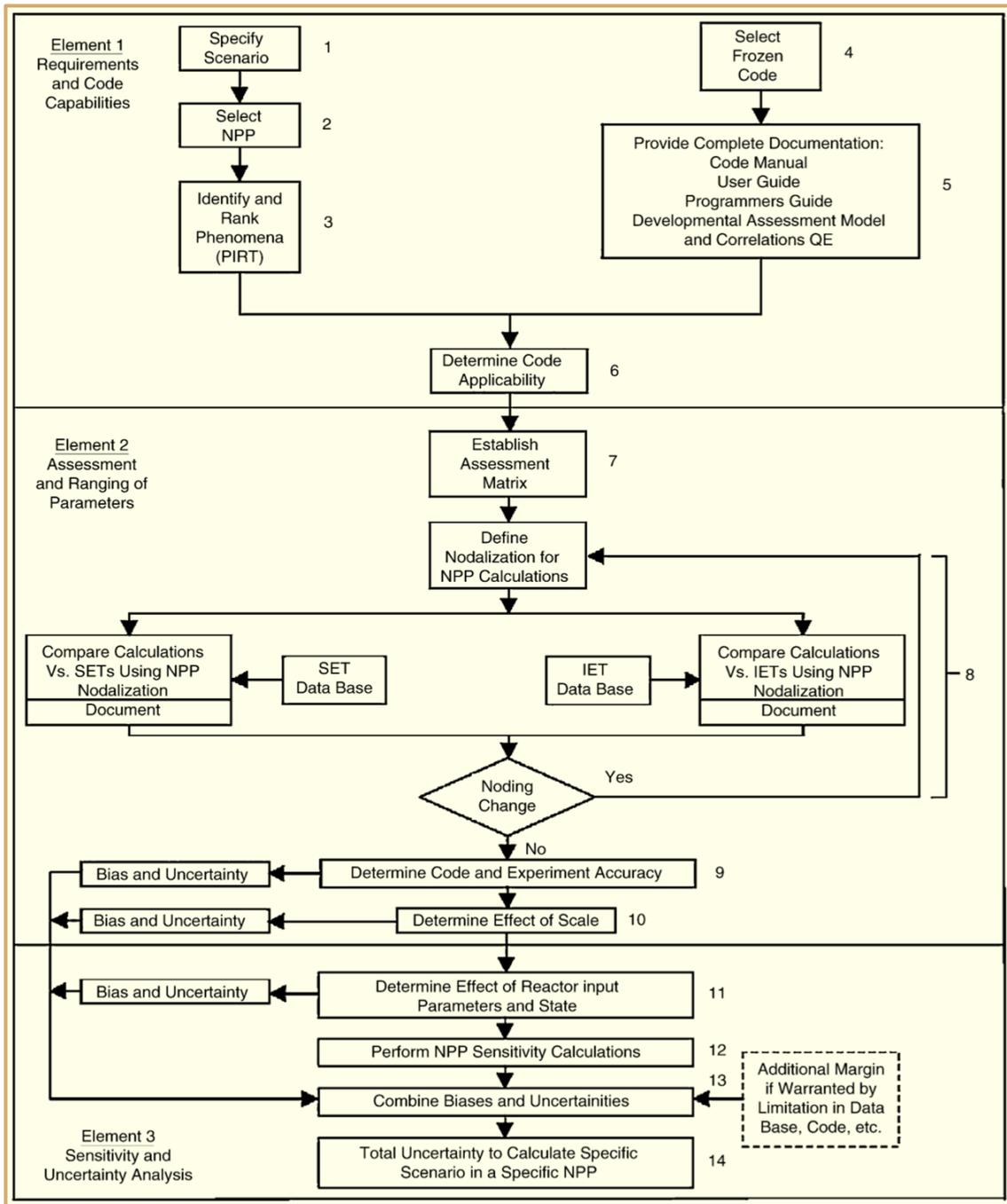


Figure G-1: The flow chart of the CSAU methodology (Figure is copied from reference [Martin, O'Dell, 2005]).

CSAU has been extensively used in the safety evaluations of nuclear power plants in the United States and internationally. It provides a robust framework for the regulatory review of safety analyses submitted by nuclear plant operators.

The methodology has also been applied in the development and validation of new and existing thermal-hydraulic safety analysis codes. It ensures that these codes are capable of providing reliable predictions for regulatory and design purposes.

The CSAU methodology represents a significant advancement in LOCA safety analysis. The methodology enhances the credibility of safety assessments and supports the regulatory process by providing clear and quantifiable margins of safety based on sound uncertainty treatment. CSAU methodology has had a significant influence on modern LOCA safety analysis methodologies, such as ASTRUM [Nissley et al,

2003] and RLBLOCA [EMF-2103 2001] in the USA as well as to some extent on other LOCA methodologies in other countries moving towards BEPU-type methodologies. These advanced methodologies have integrated key principles from CSAU, particularly its structured approach to dealing with uncertainties and its focus on scaling and applicability of models.

## Appendix H - References

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## Nomenclature

ABB	ASEA Brown Boveri
ABSS	AB Sandvik Steel
AEC	Atomic Energy Commission
AECL	Atomic Energy of Canada, Ltd.
ANL	Argonne National Laboratory
ANT	Advanced Nuclear Technology
AOA	Axial Offset Anomaly
AOO	Anticipated Operational Occurrences
AR	Annual Report
AREVA	French nuclear reactor and fuel vendor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
ATUCHA	Nuclear reactor in Argentina
AXIOM	Westinghouse alloy
BCC	Body Centered Cubic
Bct	body centered tetragonal
B-J	Baker-Just
BNFL	British Nuclear Fuel Limited
BOR	Research Fast Reactor
BQ	Beta-Quenched
BW	Basket-Weave
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
CABRI REP	RIA test performed in CABRI reactor
CANDU	Canadian Deuterium Uranium
CE	Combustion Engineering
CEZUS	Compagnie Européenne du Zirconium Ugine Sandvik
CILC	CRUD Induced Localized Corrosion
CQD	Code Qualification Document (CQD methodologies)
CR	Control Rod
CRUD	Chalk River Unidentified Deposit
CW	Cold Worked
CWSR	Cold-Worked Stress-Relieved
CWSRA	Cold Work and Stress Relieved Annealed
DAE	Department of Atomic Energy (India)
DBA	Design Basis Accidents
DBT	Ductile-Brittle Transition

DFM	Diffusion First Model
DHC	Delayed Hydride Cracking
DNB	Departure from Nucleate Boiling
DUPLEX	A fuel cladding with an outer thin corrosion resistant layer and a core with another creep resistant alloy
DX	Short for DUPLEX
EB	Electron Beam
ECBE	Effective Control Blade Exposure parameter
ECCS	Emergency Core Cooling Systems
ECR	Equivalent Clad Reacted
EFPD	Effective Full Power Days
ELS	Extra-Low Sn
ESSC	Enhanced Spacer Shadow Corrosion
FA	Fuel Assembly
FBR	Fast Breeder Reactor
FCC	Face Centered Cubic
FDBQ	Final Dimension $\beta$ -Quenched
FDRH	Fuel Design Review Handbook
FFPH	Fuel Fabrication Process Handbook
FGR	Fission Gas Release
FMTR	Fuel Material Technology Report
GC	Guide Channels
GCR	Gas Cooled Reactor
GE	General Electric
GNF	Global Nuclear Fuel
GT	Guide Thimbles or Tubes
HCP	Hexagonal Close-Packed
HiFi	Zry-2 type of alloy with high Fe content
HPA	High Performance Alloy
HPU	Hydrogen Pickup
HPUF	Hydrogen Pick-Up Fraction
HT	High Temperature
HTGR	High-Temperature Gas-Cooled Reactor
HTP	High Temperature
HWC	Hydrogen Water Chemistry
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
ID	Inner Diameter
IGSCC	Intergranular Stress Corrosion Cracking
IPHT	In Process Heat Treated

IRI	Incomplete Rod Insertion
IZNA	Information on Zirconium Alloys
JAERI	Japan Atomic Energy Agency
KAERI	Korean Atomic Energy Research Institute
K <sub>I</sub>	Stress intensity
KCl	Kalium Chloride
KKG	KernKraftwerke Gösgen
KKL	Kern Kraftwerk Leibstadt
KLD	Longitudinal Direction
KWU	Kraftwerk union
LBLOCA	Large Break LOCA
LHGR	Linear Heat Generation Rate
LK	Låg Korrosion (Low Corrosion in Swedish)
LME	Liquid Metal Embrittlement
LOCA	Loss-of-Coolant-Accident
LTP	Low Temperature
LWGR	Light Water Graphite Moderated Reactor
LWR	Light Water Reactor
MDA	Mitsubishi Developed Alloy
MHI	Mitsubishi Heavy Industries
MIR	Russian test reactor
MOX	Mixed Oxide
ND	Normal Direction
NDA	New Developed Alloy
NEI	Nuclear Energy Institute
NG	Nuclear Grade
NFI	Nuclear Fuel Industries
NMC	Nobel Metal Chemistry
NMCA	Noble Metal Chemical Addition
NSF	NbSnFe Alloy
NSRR	Nuclear Safety Research Reactor
NWC	Normal Water Chemistry
OBE	Operating Basis Earthquake
OD	Outer Diameter
O-L	O'donnell Langer
OLNC	Online Noble Chemistry
PCI	Pellet Cladding Interaction
PCMI	Pellet Cladding Mechanical Interaction
PCT	Peak Cladding Temperature

PFM	Precipitation First Model
PGP	Particle Growth Parameter
PHWR	Pressurised Heavy Water Reactor
PIE	Post Irradiation Examination
PP	Parallel Plate
PQD	Post Quench Ductility
PR	Same as pRXA
pRXA	Partially Recrystallised Annealed Condition
PT	Pressure Tubes
PWR	Pressurised Water Reactor
QC	Quality Control
RA	Reduction in Area
RBMK	Reaktor Bolshoi Mozhnosti Kanalov
RCCA	Rod Cluster Control Assemblies
RCT	Ring Compression Tests
REA	Rod Ejection Accident
RFA	Robust Fuel Assembly
RIA	Reactivity Initiated Accident
RIC	Radiation Induced Conductivity
RIED	Radiation Induced Electrical Degradation
RX	Recrystallized
RXA	Recrystallised Annealed
SBLOCA	Small Break LOCA
SCC	Stress Corrosion Cracking
SEM	Scanning Electron Microscopy
SGHWR	Steam Generating Heavy Water Reactor
SIMFEX	SIMulated Fuel EXpansion
SNF	Spent Nuclear Fuel
SNO	SnNbOxygen parameter impacting creep
SOCAP	Second Order Cumulative Annealing Parameter
SPP	Second Phase Particles
SR	Same as SRA
SRA	Stress Relieved Annealed
SRP	Standard Review Plan
SSM	Sandvik Special Metal
SS	Stainless Steel
SSE	Safe Shutdown Earthquake
STEM	Scanning Transmission Electron Microscopy
STR	Special Topic Report
S/V	Surface-to-Volume

TD	Theoretical Density
TEM	Transmission Electron Microscopy
TIG	Tungsten Inert Gas
TrM	Transition Metal
TREX	Tube Reduced EXtraction
TSSD	Terminal Solid Solubility upon Dissolution
TTT	Temperature-Time-Transformation
TVEL	Russian Fuel Vendor
USNRC	United States Nuclear Regulatory Commission
VVER	Voda Voda Energo Reactor (Russian type PWR)
W	Westinghouse
WCh	Wah Chang
WZ	Western Zirconium
ZIRAT	ZIRconium Alloy Technology
ZIRLO	Zirconium Low Oxidation

## Unit conversion

TEMPERATURE		
$^{\circ}\text{C} + 273.15 = \text{K}$	$^{\circ}\text{C} \times 1.8 + 32 = ^{\circ}\text{F}$	
T(K)	T( $^{\circ}\text{C}$ )	T( $^{\circ}\text{F}$ )
273	0	32
289	16	61
298	25	77
373	100	212
473	200	392
573	300	572
633	360	680
673	400	752
773	500	932
783	510	950
793	520	968
823	550	1022
833	560	1040
873	600	1112
878	605	1121
893	620	1148
923	650	1202
973	700	1292
1023	750	1382
1053	780	1436
1073	800	1472
1136	863	1585
1143	870	1598
1173	900	1652
1273	1000	1832
1343	1070	1958
1478	1204	2200

Radioactivity	
1 Sv	= 100 Rem
1 Ci	= $3.7 \times 10^{10}$ Bq = 37 GBq
1 Bq	= $1 \text{ s}^{-1}$

MASS	
kg	lbs
0.454	1
1	2.20

DISTANCE	
x ( $\mu\text{m}$ )	x (mils)
0.6	0.02
1	0.04
5	0.20
10	0.39
20	0.79
25	0.98
25.4	1.00
100	3.94

PRESSURE		
bar	MPa	psi
1	0.1	14
10	1	142
70	7	995
70.4	7.04	1000
100	10	1421
130	13	1847
155	15.5	2203
704	70.4	10000
1000	100	14211

STRESS INTENSITY FACTOR	
MPa $\sqrt{\text{m}}$	ksi $\sqrt{\text{inch}}$
0.91	1
1	1.10