ZIRAT-9 SPECIAL TOPICS REPORT

Loss of Coolant Accidents, LOCA, and Reactivity Initiated Accidents, RIA, in BWRs and PWRs

Prepared by

Peter Rudling
Advanced Nuclear Technology International Europe AB,
Surahammar, Sweden

Ron Adamson
Zircology Plus, Fremont, California, USA

Friedrich Garzarolli
Erlangen, Germany

Reviewed by

Alfred Strasser,
Aquarius Services Corp., Sleepy Hollow, NY, USA

September, 2004

Advanced Nuclear Technology International
Ekbacken 33
SE-735 35 SURAHAMMAR
Sweden

info@antinternational.com

DISCLAIMER

The information presented in this report has been compiled and analysed by Advanced Nuclear Technology International Europe AB (ANT International) and its subcontractors. ANT International has exercised due diligence in this work, but does not warrant the accuracy or completeness of the information. ANT International does not assume any responsibility for any consequences as a result of the use of the information for any party, except a warranty for reasonable technical skill, which is limited to the amount paid for this assignment by each ZIRAT program member.

FOREWORD

To increase the readability of the report some of the figures have been duplicated, i.e., the same figure may appear in two different sections.

Also, the units presented in this report (US or SI units) are those provided in the orginal reference. However, in many cases the numeric values with both units are provided. At the end of this report a conversion table appear providing conversion factors between SI and US units.

Peter Rudling

CONTENTS

1 INTRODUCTION (PETER RUDLING)

Nuclear authorities in different countries are concerned about the applicability of the existing *LOCA* and *RIA* criteria to high burnup *LWR* fuel. This concern is related to the problem that at high burnups, some new phenomena occur that was not envisioned at the time the current *LOCA* and *RIA* criteria were developed. These criteria were based on results from fuel rods with Zry-2 and –4 claddings that either were non-irradiated or were irradiated to a low burnup level. The high burnup phenomena that may impact *LOCA*/*RIA* fuel performance are the development of:

- A porous rim, where the local burnup exceeds the pellet average by a factor of two or more and in which the pellet microstructure is markedly altered, appears at the peripheral region of the pellet. During *LOCA*/*RIA* the fuel pellet temperature in the pellet periphery may become significantly higher than that during in-reactor operation. This increase in temperature may result in significant transient fission gas release during the *LOCA*/*RIA* event
- A chemical bond between the pellet periphery and the cladding inner surface.
- A high rod internal gas pressure.
- Fuel relocation in ballooned area of the fuel rod.
- The development of an oxide at the fuel clad outer surface during in-reactor irradiation. The in-reactor fuel clad oxidation will also increase the hydrogen content of the cladding.
- If the in-reactor oxide becomes too thick, oxide spallation may occur which may redistribute the clad hydrogen concentration, forming hydride blisters that significantly may decrease the clad ductility during *RIA*.
- The hydrogen in the cladding generated during in-reactor operation will redistribute during the *LOCA* event, significantly impacting the ductility of the part of the cladding experiencing β-phase transformation during *LOCA*. The impact of hydrogen on fuel cladding ductility is very much dependant on the temperature. The hydrogen embrittlemement effect is much larger during post-*LOCA* events compared to that during the quenching phase from the *LOCA* oxidation temperature, the former occurring at much lower temperatures.

The *LOCA*/*RIA* fuel performance may also be impacted by the situation today of replacing the Zircaloy (Zry-2 and –4) materials with Zr-Nb materials such as ZIRLO, M5 and *MDA*. It is believed that the the new materials will have a better *LOCA* and *RIA* performance than that of the Zry-2 and –4 materials since the new Zr-Nb materials picks up less hydrogen (due to less corrosion) during the base irradiation prior to the hypothetical accident event. However, the *LOCA* and *RIA* performance of the new materials must be assessed.

This special topic report will give insight and understanding of the current *LOCA* and *RIA* issues and reviews the applicability of the data to high burnup fuel cladding.

2 REACTOR SAFETY (PETER RUDLING)

2.1 GENERAL PRINCIPLES OF REACTOR SAFETY

2.1.1 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*[1] components and systems.

  – The construction of barriers in the nuclear reactor systems to limit radioactivity escape into the environment. For *BWR*s and *PWR*s,

    ▪ 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, e.g. Sweden) outside the containment

---

[1] 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 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*, 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 borated water (boric add 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.

- 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.

2.1.2 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.

Events with a probability varying from ~1 to 10^{-2}/yr were characterised as anticipated operational occurrences, class II event. For these more probable transients, safety criteria do not allow fuel failures and do only accept that a very small number of fuel rods in the core may experience boiling crisis. More specifically, the Departure from Nucleate Boiling Ratio (*DNBR*) for *PWR*s and the Critical Power Ratio (*CPR*) for *BWR*s shall be determined so that with 95% probability at the 95% confidence level the critical heat flux is not exceeded. Thermal hydraulics is covered in more details in APPENDIX C. 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 were characterised as (postulated) accidents. For the less probable accidents fuel failures are allowed but the criteria are usually established to ensure core coolability. Postulated accidents may be divided into two parts, namely

- Events of low probability, class III events, with the potential for small radioactive releases outside plant site. In these postulated event the core would be kept covered with water. Examples of these events are:
  - Small pipe break.
  - Loss-of-flow accident

- Potentially more severe accidents of very low probability, class IV events. These are called Design Basis Accidents, *DBA*. The most severe design basis accident 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 a similar break in a recirculation pump intake line of a *BWR*, Large Break Loss-of-Coolant Accident, *LBLOCA*. Other design basis accidents are:
  - Earthquakes, Tornadoes, and Flooding
  - Control Element Ejection, (Reactivity Initiated Accident)
  - Spent-Fuel Handling Accident

In a severe accident the coolability criteria cannot be maintained resulting in a 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, refueling
II	Anticipated transients		$1\text{-}10^{-2}$	Loss of feed-water, pump trip, turbine trip
III	Anticipated transients with additional failures	Fuel failures OK, but fuel should retain coolable geometry,	$<10^{-2}$	Break outside containment, small primary break, turbine trip without bypass with scram on second signal
IV	Design basis accidents		$<10^{-4}$	Large break *LOCA*, Falling control rod
	Severe accidents		$<10^{-6}$	Station blackout, *LOCA* with large leak from drywell to wetwell

The work done during the 60s and 70s mentioned above were formalised in the legislating documents produced by the Nuclear Regulatory Commission in USA, *USNRC*. There are two parts of the legislating documents that have relevance to *LOCA* and *RIA*, namely, Title 10 Code of Federal Regulations, Part 50, 10CFR50, and 10CFR100, NRC, 1995. The Standard Review Plan, SRP, NRC, 1981, are used by NRC regulators to interpret 10CFR50, and 10CFR100. The main objective of 10CFR50 and 10CFR100 is to limit radioactive impact on the environment, as follows:

- In 10CFR50, the General Design Criteria, *GDC*s are specified and interpreted in SRP section 4.2-4.4, which imposes mechanical, nuclear and thermal hydraulic fuel design criteria that the fuel vendor and the utility must meet.

- In 10CFR100, it is specified that conservative dose calculations must be done to assess the potential impact on the environment during a *DBA*.

  This is handled differently in various countries.

  In USA it must be assumed that 100 % of the core is failed (even if that is not the case) during a *LBLOCA* and the calculated dose to the environment must be below the 10CFR100 dose limit, Shoop, 2004. The dose calculations for all the other *DBA*s are calculated based upon the results of the *DBA* analysis. For *RIA*, a conservative assumption is used that all rods which experienced a surface heat flux in excess of the *DNBR* (in *PWR*s) and *CPR* (in *BWR*s) have failed. For the dose calculations, only the source term generated by these assumed failed rods need to be taken into account. Historically, NRC has defined that during a *RIA*, the dose must be within 25 % of the 10CFR100 dose limits. For other *DBA*s (other than *LBLOCA* and *RIA*), NRC has licensed plants to specific requirements, normally 10 % of the 10CFR100 limits. The reasons for the difference in maximum allowable dose for different *DBA*s is that in the case of e.g a *LOCA*, there is some delay in the radioactivity leaking out to the environment while for other accidents (with lower maximum allowable doses) such as steam generator tube rupture, there is a direct path of the radioactivity to the environment.

  In Sweden dose calculations are not done for *LBLOCA* since all Swedish nuclear power plants are equipped with a filter that essentially eliminate any spread of activity to the environment during these types of accidents. Also in France these type of dose calculations are not performed. In Germany it must shown that less than 10 % of the fuel has failed during a *LOCA* and the number 10 % is then used in the dose calculations. In Germany it must shown, by code calculations, that less than 10 % of the fuel has failed during a *LOCA*, see Figure 2-2. However, even though the calculated number of failed rods is less than 10 %, the 10 % number must be used in the dose calculations to be conservative.

Figure 2-1 provides more details of these documents. See also APPENDIX A for more details.

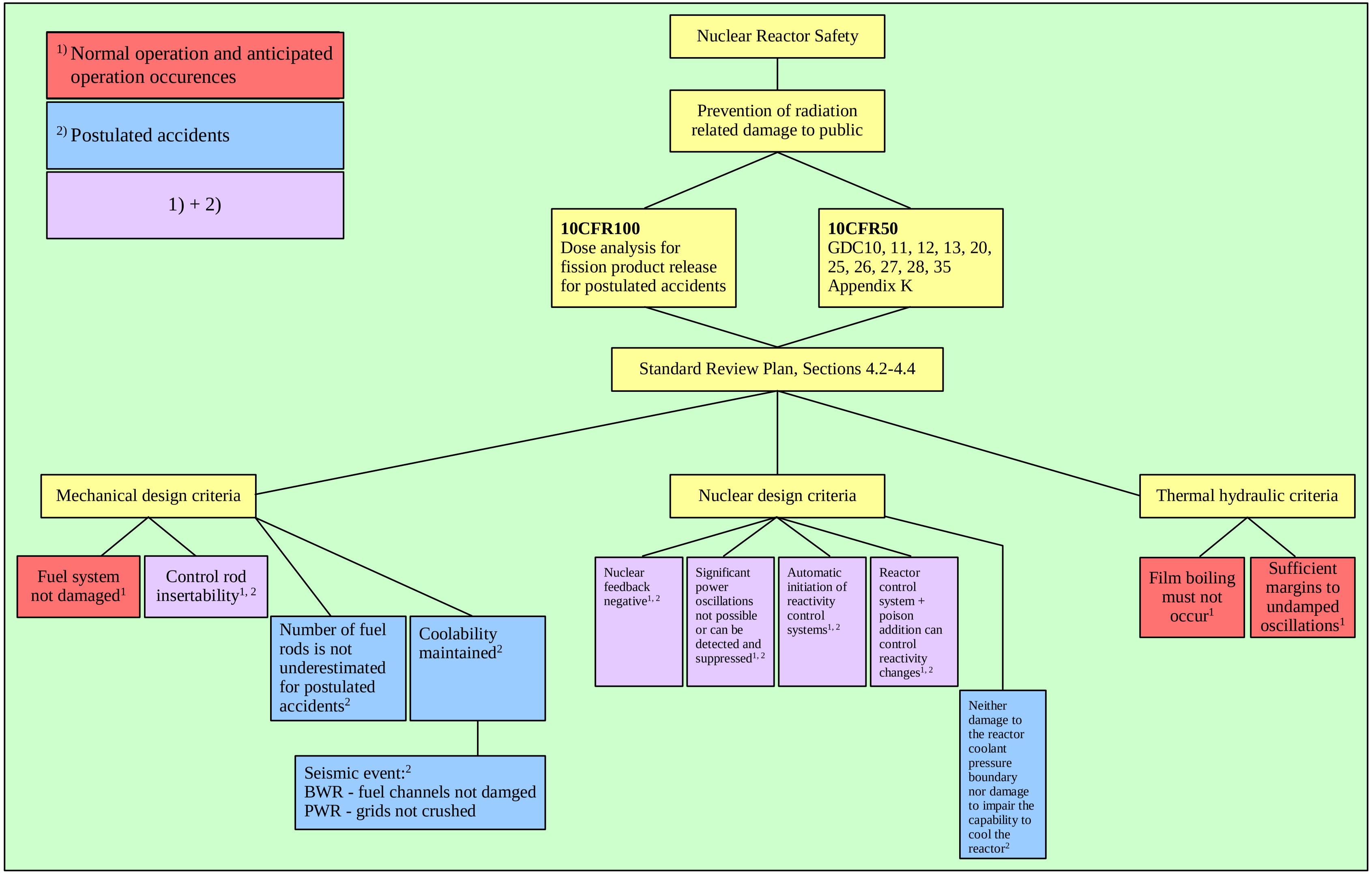


Figure 2-1: Outline on relation between 10CFR50, 10 CFR100 and SRP section 4.2-4.4.

As mentioned earlier the basis for the *DBA* criteria as well as the related analytical methods were experiments were tests performed on essentially non-irradiated fuel with Zry-2 and-4 claddings. However, by the mid 1980s, changes in pellet microstructure and increased fuel clad corrosion had been observed for higher burnups implying that the margins towards the safety critera may be perhaps lower at high burnups. Thus, to better evaluate the effects of the higher burnup on fuel behaviour, specifically under *RIA* and *LOCA* conditions, a number of programmes were initiated.

Specifically two tests to study *RIA* performance on highly irradiated fuel, performed by the French in the CABRI facility (REP Na-1) and by the Japanese in the NSRR Facility (HBO-1), triggered an enhanced effort of the nuclear industry to assess the effect of high burnup under *DBA* conditions. During these two tests, rod failure and fuel dispersal was observed at much lower enthalpy values than the fuel enthalpy limits for fuel rod failures and core coolability that had been established earlier by the various regulatory authorities.

2.1.3 Fuel operating margins

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

The fuel vendors are using fuel performance codes to determine the thermal limits on their fuel design that will ensure that the fuel design criteria are met, Figure 2-2. For each fuel design criteria there will be a thermal limit varying with burnup and the most limiting thermal limit will establish the operating regime of the 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 thermal limits provided by the fuel vendors for their fuel is not exceeded.

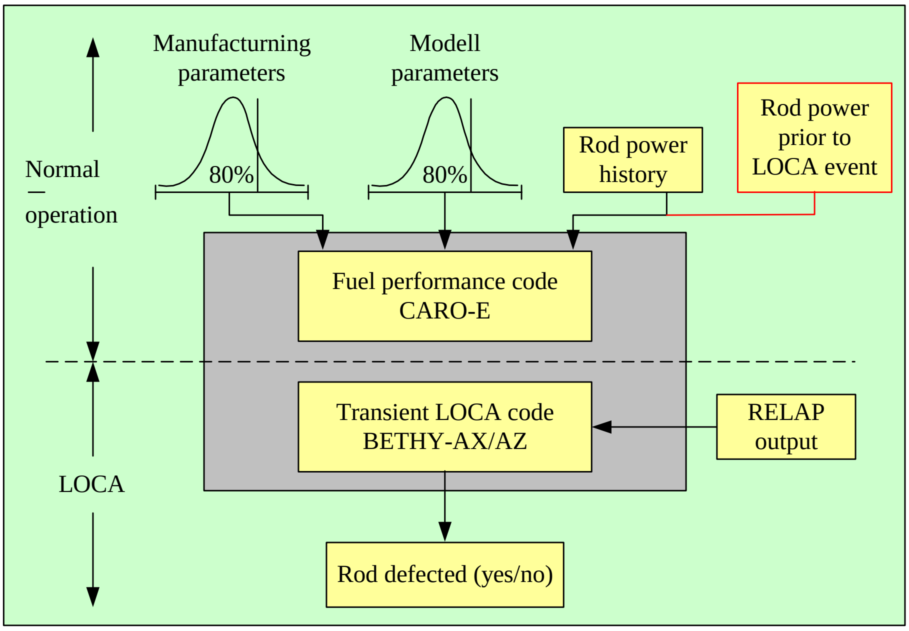


Figure 2-2: Schematics showing the process of establishement of thermal limits, modified figure according to Heins, 2004.

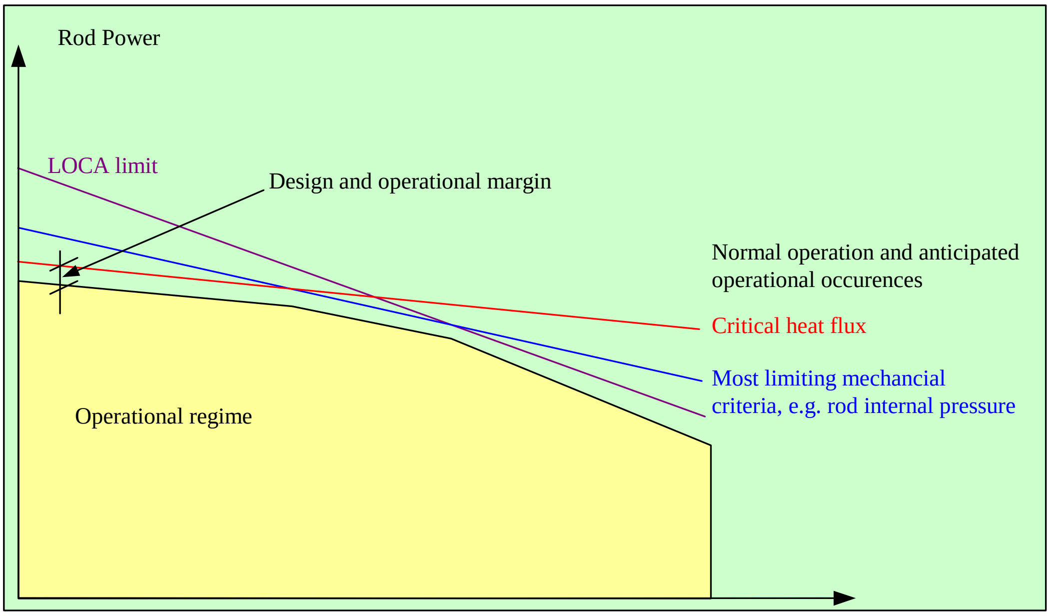


Figure 2-3: Schematic showing the operational regime. The red line represent the maximum rod power allowable to ensure that boiling crises will not occur during stationary condition or during a class II transient. The blue line represent the maximum rod power allowable to ensure that fuel failure will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

3 LOCA (PETER RUDLING)

3.1 REACTOR KINETICS

Loss-of-coolant accidents can be differentiated between several categories depending on the size of the postulated break in the primary coolant system.

For the design basis accident 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 3-2. A small break *LOCA* covers 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.

During a *LBLOCA*, the primary system pressure drops and all the reactor water is expelled into the containment. 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.

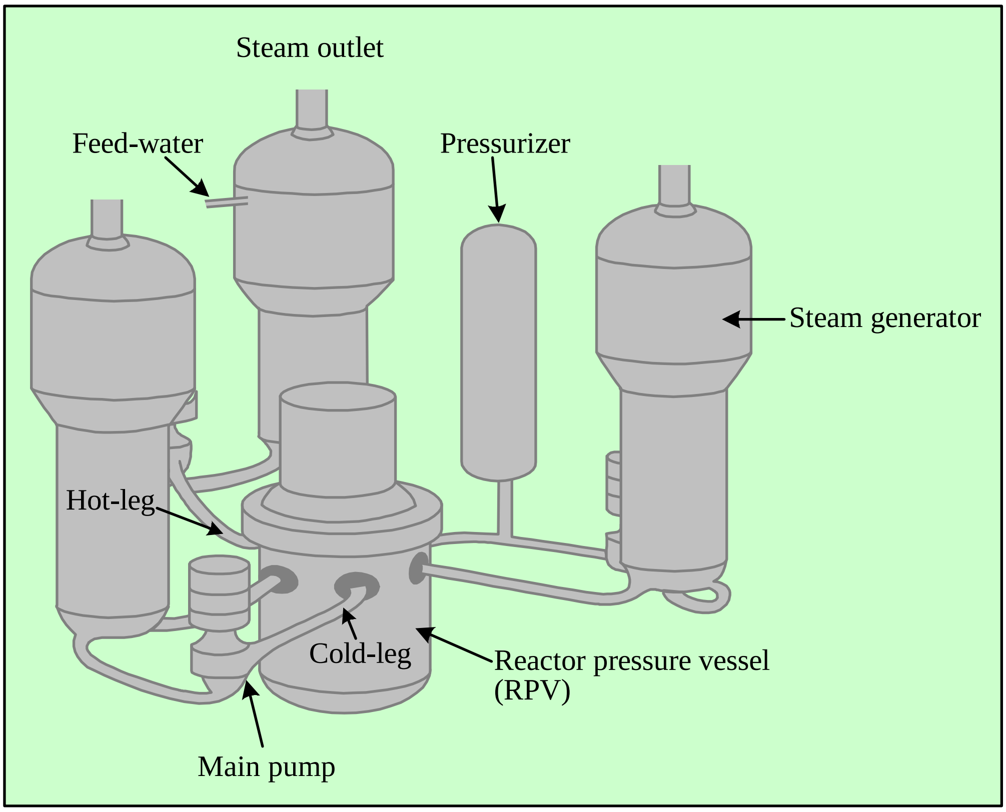
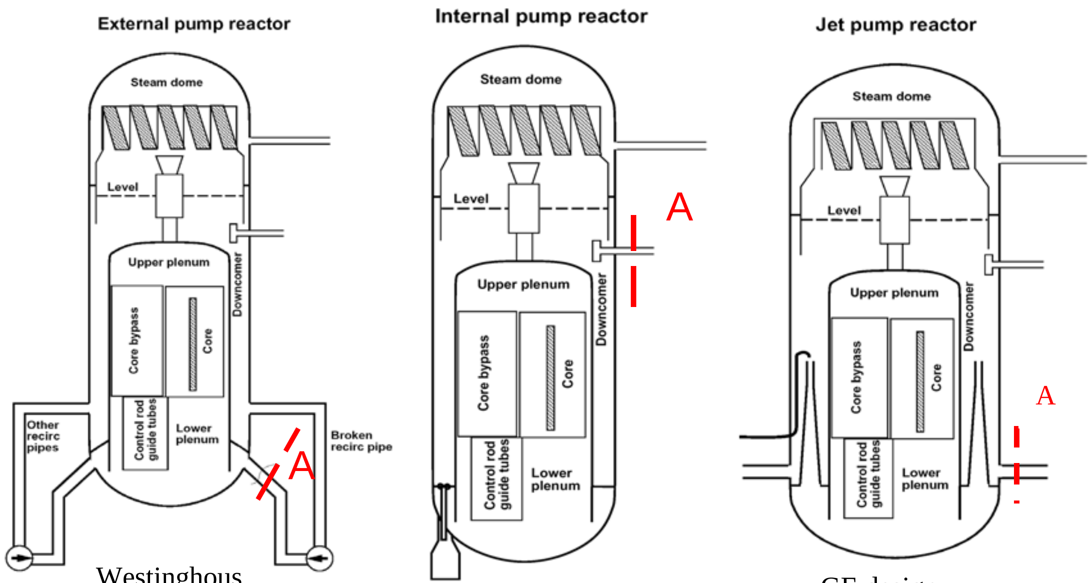


Figure 3-1: *PWR* geometry, information provided by ALARA Engineering, 2002.



Westinghous
e Sweden
design, e.g.
Ringhals 1

Framatome ANP
GmbH design
Westinghouse
Sweden design, e.g.
Oskarshamn 3

GE design

Figure 3-2: Basic *BWR* Geometries. *A* in the figure designates the location of the postulated pipe break, information provided by ALARA Engineering, 2002.

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

- *blowdown*, in which coolant would be expelled from the reactor vessel, Figure 3-3.
  - During the initial blowdown stage, the primary system pressure drops rapidly, Figure 3-5. Propagation of pressure waves may occur during this phase imposing mechanical loads that could damage the reactor system.

  - 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 fuel rods would be cooled to some extent by the steam-water flow, and the cladding temperature would drop for a short time.

    Subsequently, as the enthalpy of the fluid increases, the critical heat flux would fall 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 increase and the rod internal overpressure may result in burst failure of some of the hotter rods.

- *refill*, when the *ECCS* would begin to fill the vessel with water up to the bottom of the core,

  - The decrease in the *PWR* primary system pressure during blowdown would activate the *ECCS*. Also borated water would be 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 would rise sufficiently to cool the core, Figure 3-4.

  - Reflooding of the core would commence 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.

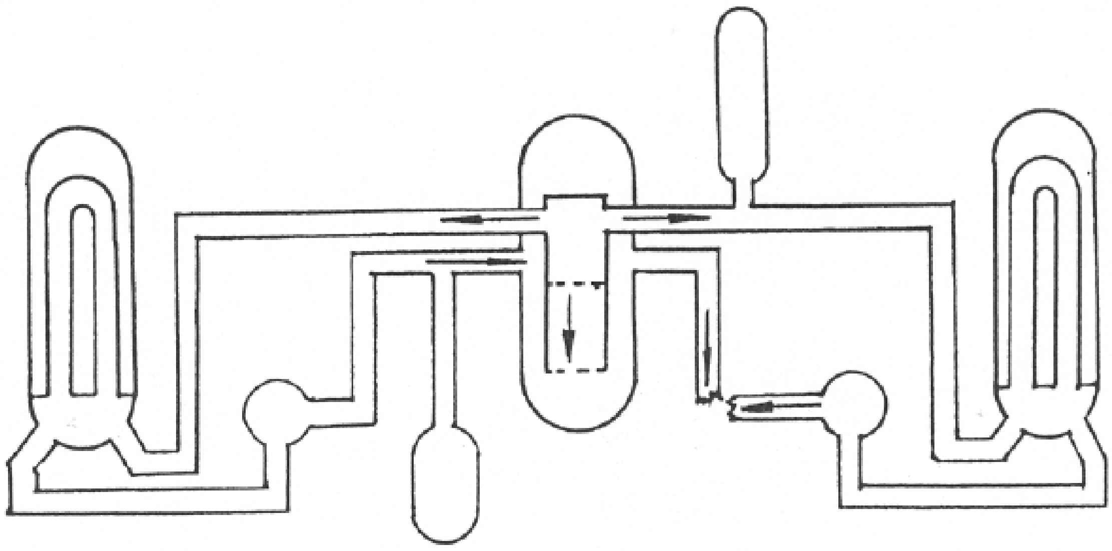


Figure 3-3: Cold-leg break in *PWR* – blowdown phase, information provided by ALARA Engineering, 2002.

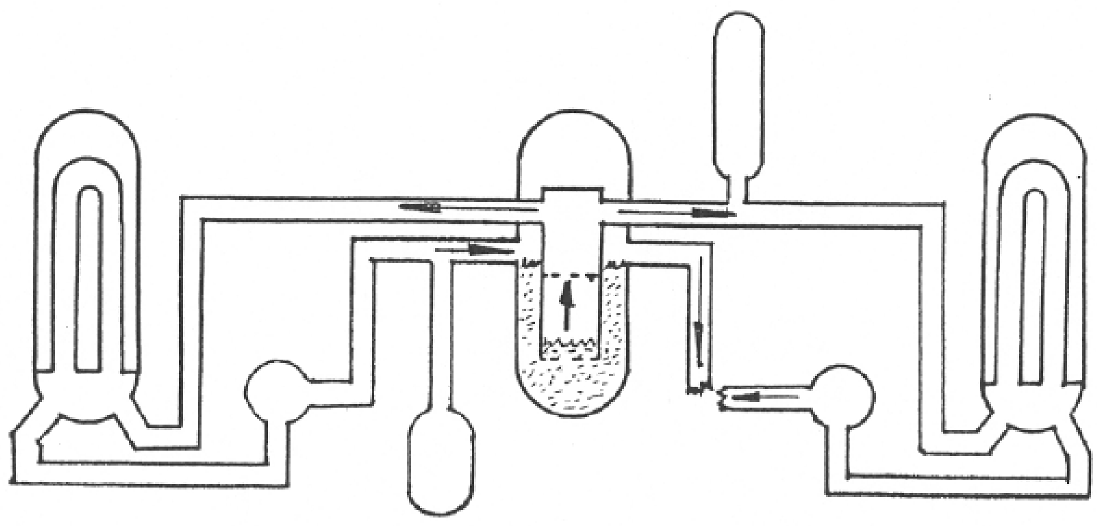


Figure 3-4: Cold-leg break in *PWR*, reflood phase, information provided by ALARA Engineering, 2002.

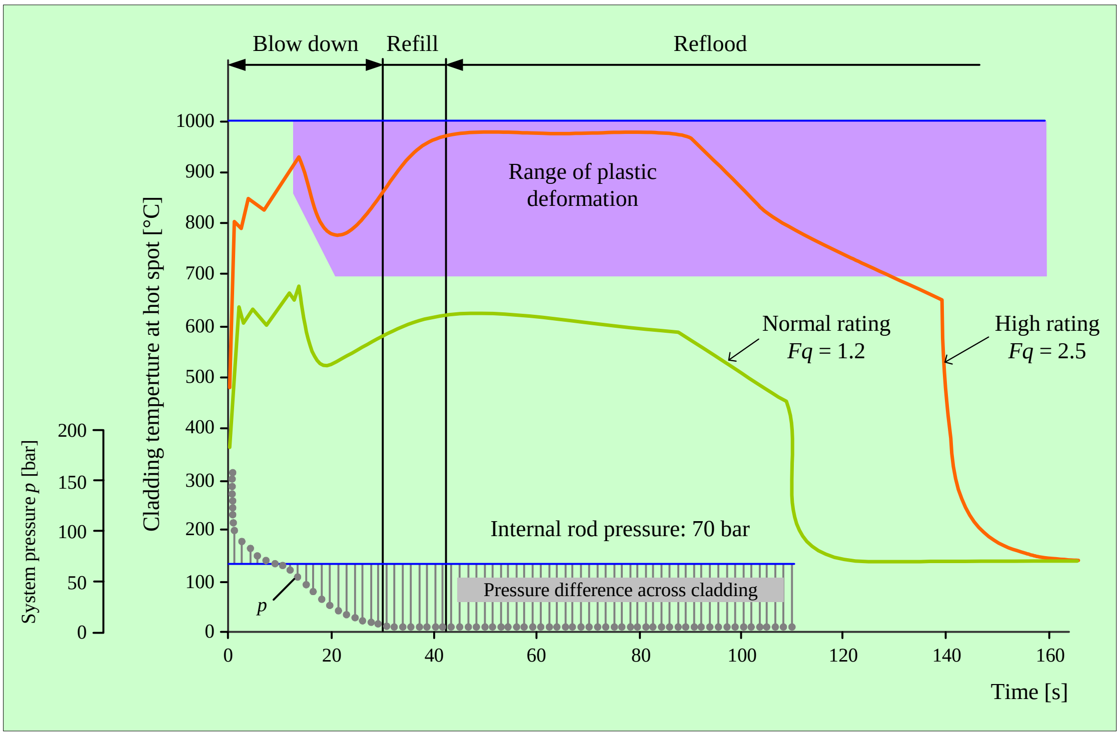


Figure 3-5: *PWR LBLOCA*. A pressure of 70 bar corresponds to 995 psi, a temperature of 1000ºC corresponds to 1832ºC. Modified figure according to Erbacher & Leistikow, 1987.

For a design basis *LOCA* in a jet pump *BWR*, Figure 3-2, the following phases can be identified, Figure 3-6:

- *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.

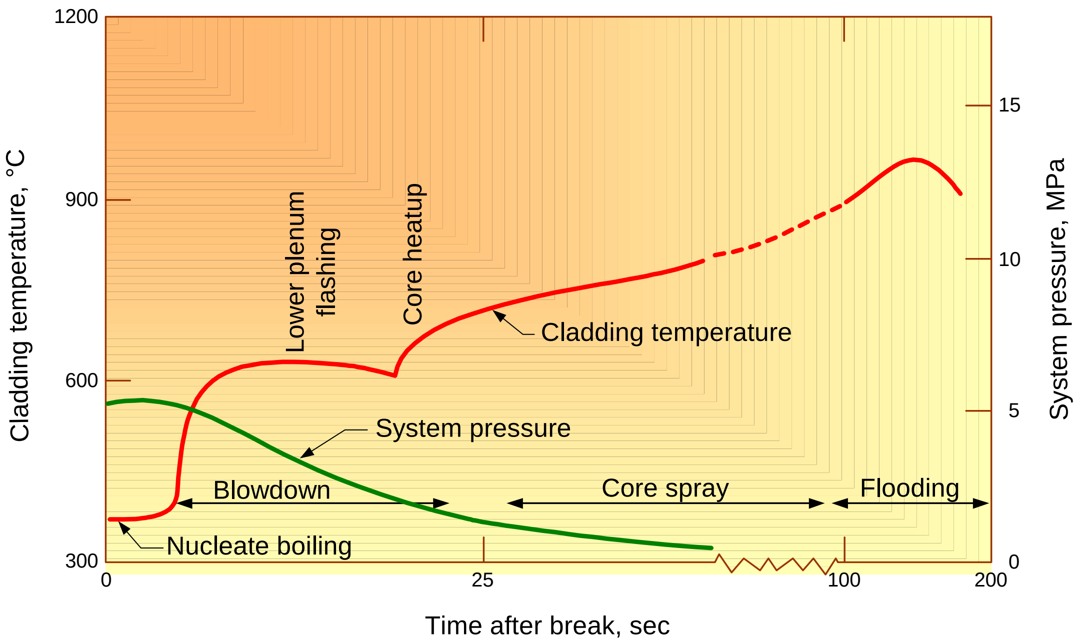


Figure 3-6: *BWR LBLOCA* in a jet pump reactor. A pressure of 10 MPa corresponds to 1421 psi, a temperature of 900ºC corresponds to 1652ºC.

Internal pump reactors, Figure 3-2, have similar clad temperature evolution as that in the jet pump reactor. However, the *LOCA* oxidation event is much more severe in an external pump reactor, Figure 3-2, compared to both a jet and internal pump reactor, as shown in Figure 3-7. Due to this rector 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 Liedenfrost temperature has been reached resulting in a quench of the clad temperature. Comparing Figure 3-7 with Figure 3-6, 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.

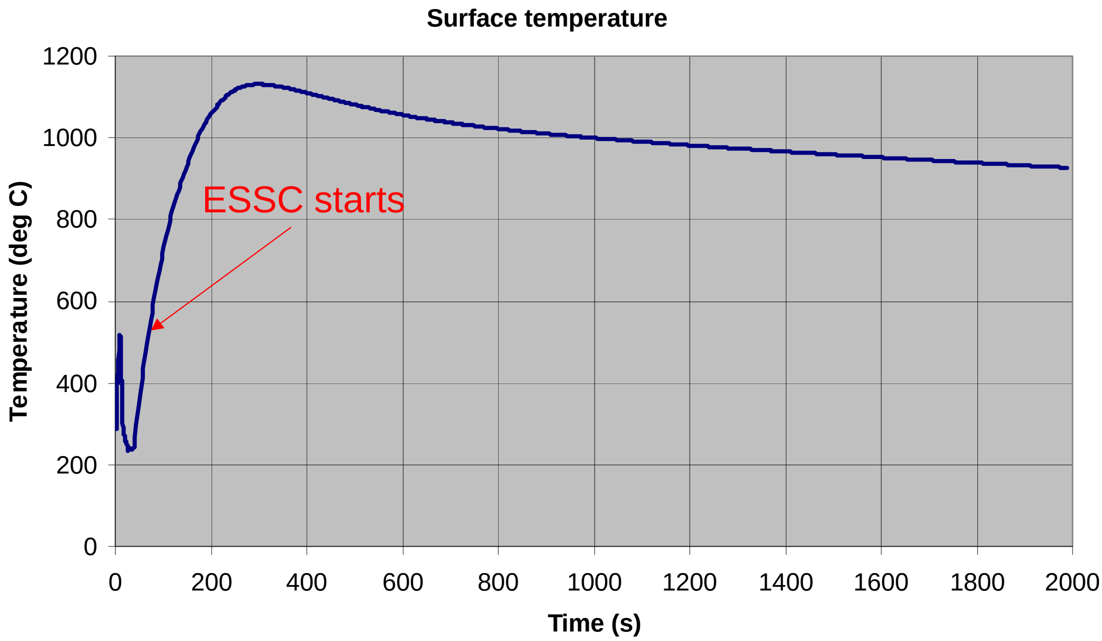


Figure 3-7: Fuel clad evolution during a *LOCA* in an external pump *BWR*. A temperature of 1000ºC corresponds to 1832ºC. Information provided by ALARA Engineering, 2002.

3.2 FUEL BEHAVIOUR DURING LOCA

3.2.1 Introduction

The fuel rod behaviour during the *LOCA* event is schematically shown in Figure 3-8 and can be separated into three different phases:

- The loss of coolant results in a dramatic increase in fuel clad temperature. Ballooning of the cladding occurs due to the high clad temperature and large clad stresses[2]. At a certain clad strain clad burst will occur. Ballooning may also result in fuel relocation, see section 3.2.2.8.4.

  - Higher fuel rod internal pressure, e.g., due to higher fission gas release, and higher fuel stored energy (due to higher Linear Heat Generation Rate prior to the *LOCA* event, see Figure 3-5) leading to higher fuel clad temperature will result in earlier burst during the *LOCA* event.

  - The fuel rod clad temperature is a function of the fuel assembly component design such as e.g. grids and fuel outer channel (*BWR*s).

    During the *LOCA* heat transfer mostly occurs by radiation and consequently cool surfaces such as grids/spacers and fuel channels (in *BWR*s) improves fuel rod cooling.

    - In *PWR*s, the grids will also impact the reflood phase. Mixing vanes results in better cooling since the vanes will break up the water droplets leading to an increase in the wetted fuel rod surface area, see e.g., Hochreiter & Cheung, 2002.

    - In *BWR*s, the spacers will not impact the fuel rod cooling away from the spacer but the spacer will improve cooling of the fuel rod surfaces close to the spacer due to its cool surface. Also, the existence of the fuel outer channel surfaces will help to cool the fuel rods. A fuel outer channel, such as the SVEA design with more cool surfaces (due to the internal water cross) will tend to reduce cladding temperature further (due to radiation heat transfer form the rods to the cool surfaces).

  - Ballooning may also restrict coolant flow

---

[2] The large clad stresses results from the large fuel rod internal overpressure due to the loss of the system pressure

- The cladding is oxidized by the steam. 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.

- The embrittled cladding may rupture by thermal shock caused by quenching or due to post-*LOCA* events. To maintain "coolable geometry", fuel rod rupture is not acceptable. "Coolable geometry" implies that the fuel must be contained in the fuel rods during quenching and post-*LOCA* event (such as e.g. a seismic event). The key parameters that gouverns the clad embrittlement effect is:

  – the degree of clad oxidation that is a function of oxidation temperature and time.

  – the fuel clad hydrogen content

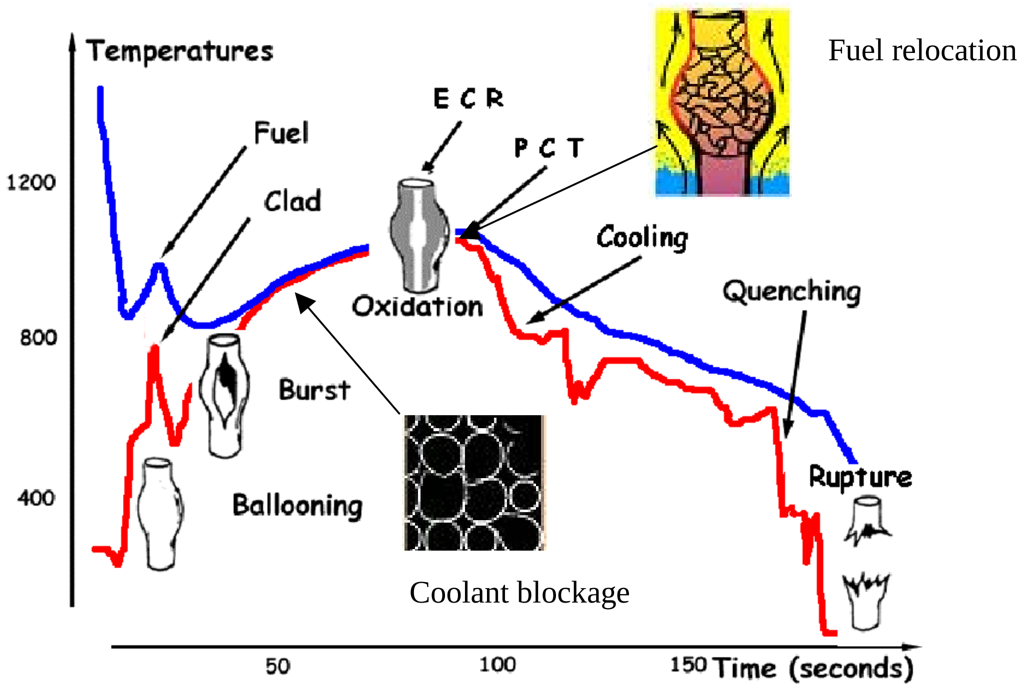


Figure 3-8: Typical *LBLOCA* in a *PWR*.

The different *LOCA* phases are described more in the following subsections.

3.2.2 Fuel clad ballooning, burst and blockage

When the system pressure drops below the rod internal pressure, tensile stresses in the fuel cladding will tend to plastically deform the fuel cladding. The creep strength of Zircaloy decreases rapidly with temperature and at 700ºC strain rates can become very high. Most of the circumferential expansion at burst rupture occurs during localized ballooning deformation after the onset of plastic instability[3]. During this period, the cladding temperature is nearly constant and the strain rate is quite high (1 to 10 s^{-1}). The ductility of Zircaloy at these high temperature is high and strains of 50 % or more are possible to obtain. If large clad strains are obtained blockage of the coolant sub-channel may result impacting the fuel coolability.

The basic parameters controlling fuel clad deformation are:

- stress,
- temperature and
- creep strength, being affected by oxidation, grain size and anisotropy.

When the temperature of the stressed tube is uniform, deformation is unstable, i.e. if the tube increases in diameter at any axial position, the increased stress resulting from the larger diameter and reduced wall thickness will lead, by positive feedback, to runaway deformation and rupture.

However during a *LOCA*, convective cooling by uprising steam, will tend to produce axial temperatures gradients in the fuel rods, so that fuel clad swelling, if it occurs, will vary along the rods.

Fuel clad ballooning and effects on coolant blockage have been studied in single- and multi-rod out-of-pile and in-pile tests of fuel rods by different types of heating. The interested reader is referred to Erbacher & Leistikow, 1987 and Grandjean & Hache, 2004, that nicely reviews this topic. Single-rod tests performed out-of-reactor form the bulk of published results. *However, one has to be cautious when interpreting the applicability of the single rod results for a "real LOCA" with high burnup fuel in lattice configuration. Several multi-rod programs have also been performed to better simulate the "real situation"*. More details are given in sections 3.2.2.9 and, 3.2.2.10. The most important tests performed are provided in Table 3-1.

[3] Plastic instability is defined as the transition from uniform tube expansion to localized ballooning.

Some of the main lessons drawn from the results of the ballooning and burst tests, which were performed under various experimental conditions are as follows:

- In the range of stresses and temperatures which may be produced during a *LOCA*, clad strains in the range 30%-90% can be produced

- The behaviour of the fuel cladding in a temperature transient is strongly influenced by the temperature distribution axially and azimuthally. This distribution is in turn dependent on heat transfer mechanisms at the surfaces of the cladding. Meaningful experimental simulations must therefore accurately reproduce these mechanisms. *This implies the use of realistically heated fuel rod simulators, realistic conditions of surface heat transfer, and the use of multi-rod assemblies to reproduce the heat-transfer conditions in the sub-channels between rods.*

- *Co-planar deformation*[4] with strains up to and including those leading to mechanical interaction between fuel rods have been demonstrated experimentally. No *LOCA* experiment with a multi-rod array and simulated reflood cooling has produced deformations which would inhibit fuel rod cooling. It has not yet been assessed experimentally what degree of *blockage*[5] would be needed to result in loss of coolability.

- The significant influence of hydrogen on fuel clad ballooning behaviour.

- The lack of bundle behaviour results, e.g. fuel relocation, of high burnup fuel rods.

---

[4] Fuel rod deformation (ballooning) occuring at the same elevation of different rods in the fuel assembly lattice.
[5] Blockage (%) = 100X ((Difference in cross-section between ballooned and non-ballooned rod)/(original coolant channel cross-sectional area)).

Table 3-1: Important clad ballooning and rupture / flow blockage tests performed.

Out-of-Pile Tests		**In-Pile Tests**	
Single rod test	**Multi rod test**	**Single rod test**	**Multi rod test**
EDGAR (CEA, France) • Direct heating		PBF-LOC (INEL, USA)[1]	PHEBUS (IRSN, France)[2], [3], [4], [5]
REBEKA (KfK, Germany) • Internal heating + heated shroud	REBEKA (KfK, Germany) • REBEKA-4 test (5x5 bundle with unpressurized rods in outer ring) FEBA (KfK, Germany) • 5x5 rod bundle; • "conventional" simulators; forced reflood • Blockage over 3x3 or 5x5 rods; τ=62% or 90% ; thick sleeves; SEFLEX (KfK, Germany) • 5x5 rod bundle; REBEKA simulators ; forced reflood • Blockage over 3x3 rods; • τ= 90% ; thinned cladding;	FR2 (KfK, Germany)[6]	NRU-MT (AECL, Canada)[7], [8] • NRU MT-4 vs. MT3 tests (32 full length rods, 12 inner rods pressurized)
ORNL • Internal heating + heated shroud	ORNL (MRBT)[9] • (4x4, 8x8) lattice	EOLO-JR (Ispra)[10]	
Creep rupture test (Erlangen, Germany), see Figure 3-12	JAERI (Japan)[11] • (7x7) lattice		
	THETIS (AEA Winfrith, UK) • 7x7 rod bundle; "conventional" simulators; forced or gravity reflood • Blockage over 4x4 rods; τ=80% or 90%; thin sleeves; CEGB (Berkeley, UK) • 44 rod bundle ; • Blockage over 4x4 rods; τ= 90% ; forced reflood FLECHT–SEASET (W, USA) • 21 and 163 rod bundles; forced or gravity reflood • Short concentric sleeves, coplanar or not; long non-concentric sleeves, non-coplanar		

1) MacDonald, et al., 1981, 2) Adroguer, et al., 1983, 3) Del Negro, et al., 1982, 4) Hueber, et al., 1979,
5) Manin, et al., 1980, 6) Erbacher, et al., 1978, 7) Mohr, et al., 1983, 8) Russcher, et al., 1981,
9) Chapman, 1978, 10) Friz, et al., 1981, 11) Kawasaki, 1978

In the following subsections, the influence of various parameters on clad ballooning, burst and coolant blockage are discussed.

3.2.2.1 Impact of temperature

Even though the inherent ductility of Zircaloy is large, rupture at low overall strains due to non-uniform straining may occur. A prime cause for such non-uniform straining is circumferential variation in temperature around the cladding. The cladding will strain preferentially in the hottest region around the circumference since the creep-strength of Zircaloy is highly temperature-sensitive, Figure 3-9. Such non-uniformity in the circumferential temperature may be caused by:

- The fuel pellet stack not remaining co-axial with the cladding as the latter balloons.
- Non-uniformity in heat transfer,
- The anisotropic properties of the cladding, see section 3.2.2.3.

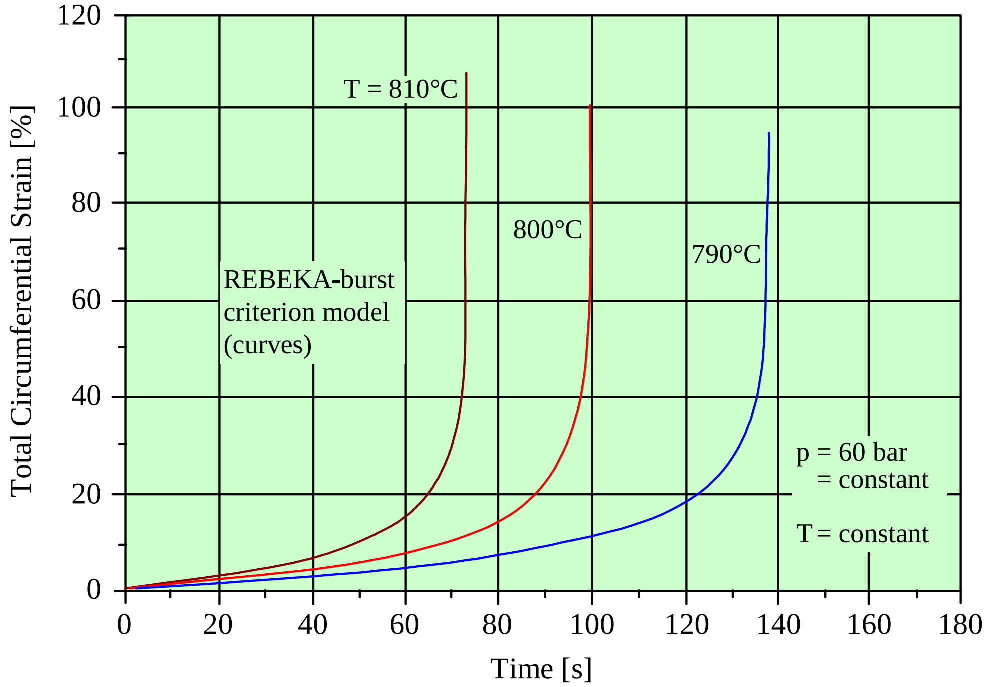


Figure 3-9: Sensitivity of Zircaloy-4 deformation to temperature, unirradiated single-rod test, modified figure according to Erbacher & Leistikow, 1987.

The impact of azimuthal temperature clad variation on burst strain is shown in Figure 3-10. Small azimuthal temperature variation in the cladding tube results in a rather homogeneous decrease of the cladding tube wall thickness along the clad circumference resulting in large burst strains. However, a large azimuthal temperature difference on the other hand results in preferential wall thickness reduction at the hottest part of the cladding and therefore a low burst strain. *It appears that the magnitude of this azimuthal temperature variation in the cladding is one of the strongest parameter that impacts the cladding tube burst strain, flow blockage and coolability during a LOCA.*

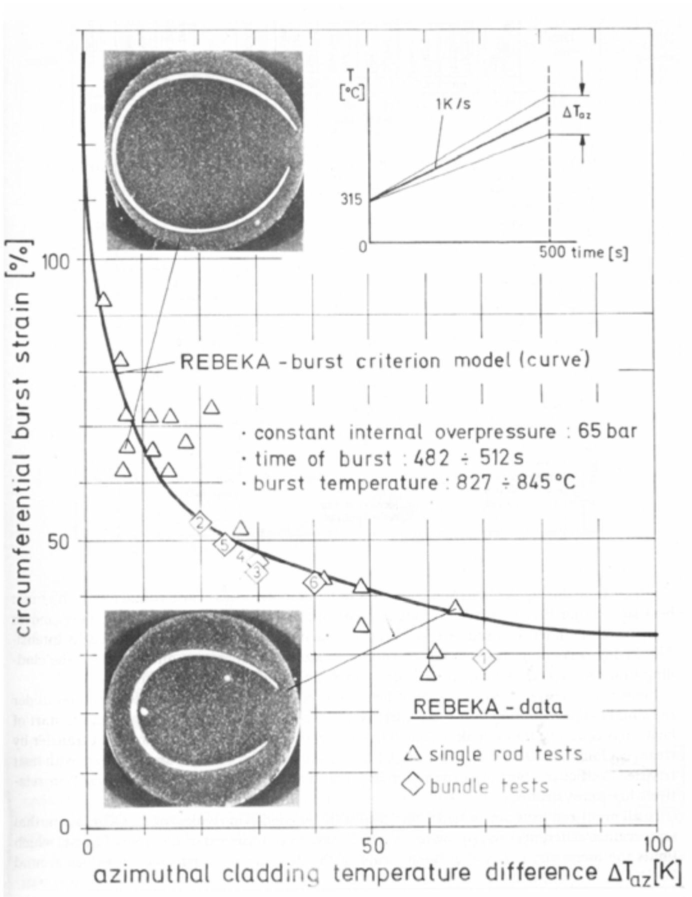


Figure 3-10: Burst strain ofZircaloy-4 cladding tubes versus azimuthal temperature difference (REBEKA l tests) Erbacher & Leistikow, 1987.

4 RIA (PETER RUDLING)

4.1 REACTOR KINETICS

The interested reader is referred to APPENDIX B for a more detailed discussion of reactor kinetics.

In a pressurized water reactor (*PWR*), the most severe *RIA* scenario is the control rod ejection accident (*CREA*). The *CREA* is caused by mechanical failure of a control rod mechanism housing, such that the coolant pressure ejects a control rod assembly completely out of the core, Glasstone & Sesonske, 1991. The ejection and corresponding addition of reactivity to the core occurs within about 0.1 s in the worst possible scenario. The actual time depends on reactor coolant pressure and the severity of the mechanical failure. With respect to reactivity addition, the most severe *CREA* would occur at hot zero power (*HZP*) conditions, i.e. at normal coolant temperature and pressure, but with nearly zero reactor power, Agee, et al., 1995 and Nakajima, et al., 2002. Figure 4-1 shows that the *rod worth*[24] decreases with increased power level *and* with a decrease in control rod insertion within the core, and therefore it is concluded that HZP should be the most limiting initial condition for the REA.

In a boiling water reactor (*BWR*), the most severe *RIA* scenario is the control rod drop accident (CRDA). The initiating event for the CRDA is the separation of a control rod blade from its drive mechanism, Glasstone & Sesonske, 1991. The separation takes place when the blade is fully inserted in the core, and the detached blade remains stuck in this position until it suddenly becomes loose and drops out of the core in a free fall.

In a boiling water reactor (*BWR*), the most severe *RIA* scenario is the control rod drop accident (*CRDA*). The initiating event for the *CRDA* is the separation of a control rod blade from its drive mechanism, Glasstone & Sesonske, 1991. The separation takes place when the blade is fully inserted in the core, and the detached blade remains stuck in this position until it suddenly becomes loose and drops out of the core in a free fall. Hence, the control rod drops out of the core due to gravity, and in contrast to the *CREA* in *PWR*s, coolant pressure does not influence the rod ejection rate. With respect to reactivity addition, the most severe *CRDA* would occur at cold zero power (*CZP*) conditions, i.e. at a state with the coolant close to room temperature and atmospheric pressure, and the reactor at nearly zero power, Agee, et al., 1995 and Nakajima, et al., 2002. The degree of reactivity addition during *CRDA* is strongly affected by the coolant subcooling, since vapour generation effectively limits the power transient.

[24] See Appendix B.4.7

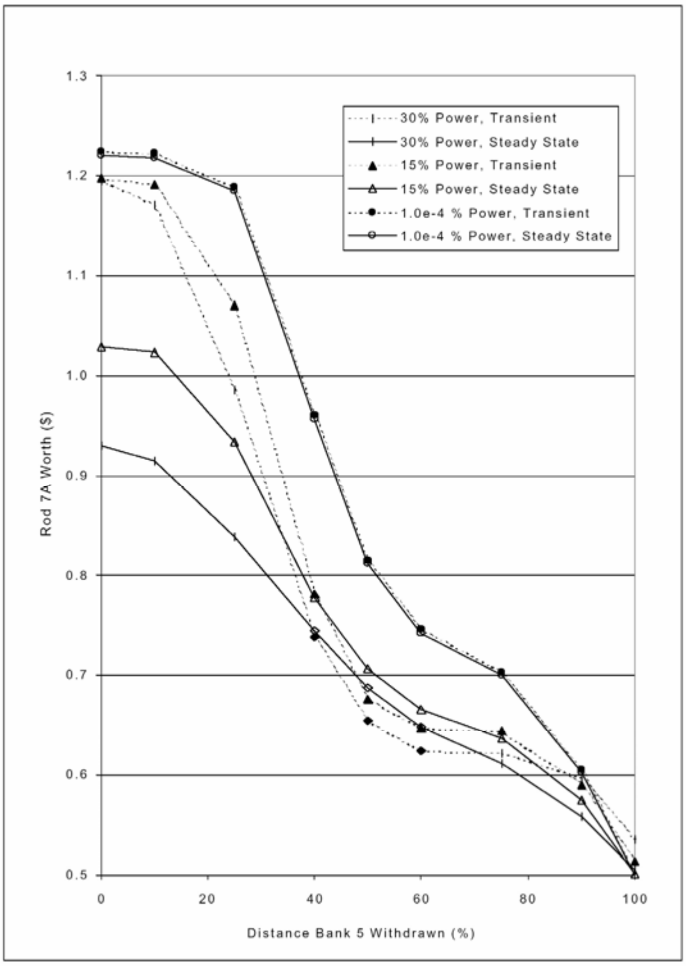


Figure 4-1: TMI-1 *PWR* EOC control rod 7a worth variation with power level, bank 5 position, and calculation procedure, Diamond, et al., 2001.

If the reactivity addition during a *CREA* or *CRDA* is sufficient, the reactor becomes prompt critical, i.e. $ ≥1, and power will rise rapidly until the negative fuel temperature feedback (Doppler effect) terminates the power rise within a few hundredths of a second, see section B.4.2 for more details. Under a *CRDA*, additional negative feedback is obtained from vapour generation in the coolant. After the power surge is terminated, the power is finally reduced to zero by insertion of fault-free control elements due to reactor trip.

In the considered *RIA* scenarios, the fuel assemblies near to the ejected control element are thus subjected to a fast and short power pulse. The shape and duration of the power pulse depend on the assumed scenario, core and fuel design, and the burnup dependent state of the fuel. Figure 4-2 shows e.g. the power trace from a *CREA* at both BOC and EOC beginning from HZP at approximately the same rod worth in a *PWR*. The differences in the results between the BOC and EOC cases is related to that there is more fissile material in the core at BOC than at EOC; hence, the fission reaction rate and power induced by a given rod ejection of equal worth will be higher at BOC than at EOC.

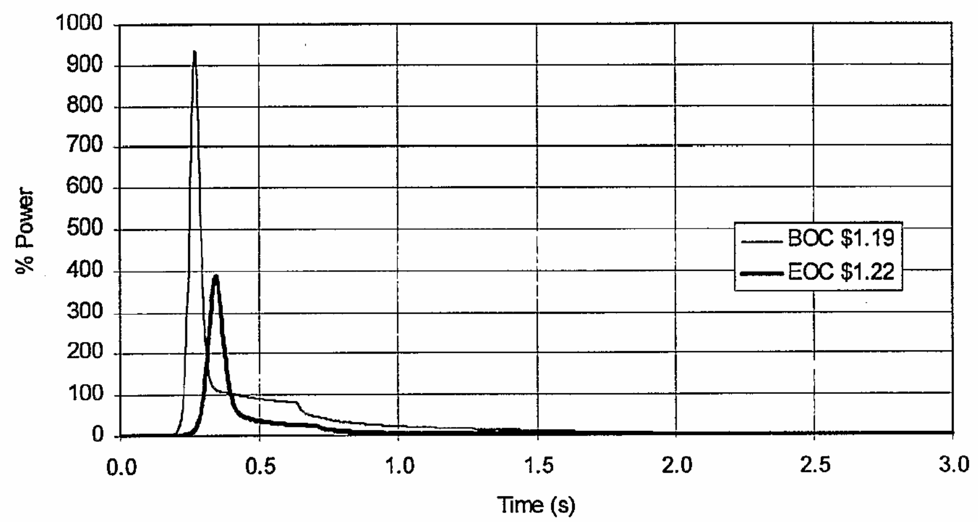


Figure 4-2: Power Transients for CREA from HZP, Diamond, et al., 2001.

Analyses of postulated *RIA* scenarios with state of the art three-dimensional neutron kinetics codes indicate that the width of the power pulse is in the range from 30 to 75 ms in fuel with burnup exceeding 40 $MWdkg^{-1}U^{-1}$, Meyer, et al., 1997. Results also indicate that the power pulse in *BWR*s is longer than that in *PWR*s, partly due to that the mass of the *BWR* control rods is so much larger than that of the *PWR* control rod cluster, Figure 4-3. Calculations by Jernkvist et al, 2004, indicate a typical *PWR* pulse width of 25 ms while the corresponding pulse width in a *BWR* ranges from 36-70 ms. Figure 4-3 also shows that the pulse width has an inverse relationship with the fuel enthalpy rise which means that high ΔH are tighed together with short pulse widths.

5 COMPUTER CODES AND LICENSING (PETER RUDLING)

Fuel vendors are required by the regulatory body to perform safety analysis to show that the fuel is designed to satisfy the fuel design criteria both during transients and accidents such as *LOCA* and *RIA*. These analyses are carried out with computer codes, either as standalone codes or in a coupled manner. In these safety analysis also steady-state calculations is also needed to establish the initial conditions prior to the transient or accident. For example, for *LOCA* analysis it is very important to have an accurate value for the stored energy of the fuel pellets at the time the accident is initiated which can be released during the accident: this stored energy comes from a steady-state calculation for normal reactor operation.

It is essential that the performance of the fuel rods during a *LOCA* and *RIA* is well-known so its behaviour can be correctly modelled by which it can be shown that the design criteria of the fuel can be met. For example, a deeper understanding of *RIA* failure mechanisms it is very important to have accurate values for the fission gas content in grain boundaries and porosities at the time of accident initiation, especially for UO_2-RIM and MOX clusters which may promote fuel swelling and grain separation during the accident; this fission gas content is obtained from a steady-state calculation at normal operating conditions, see also sections 3.2.2.8.1 and 3.2.2.8.2.

The prediction of the behaviour of fuel rods during a *LOCA* or *RIA* is a difficult task given the complexity of the system being modelled. In the case of e.g. a *LOCA* the code generally attempts to predict the deformation of a fuel rod, the termination of deformation by rupture, the temperature reached by the cladding, and in some codes the interaction between neighbouring rods. A code typically draws input information on coolant condition from a thermal- hydraulic code, and data on fuel from a steady-state code, while a range of sub-codes calculate fission gas release, deformation etc.

The thermal-hydraulic computer codes for calculating fuel and cladding temperatures (and other core properties, e.g., zirconium-water reaction) during the blowdown, refill, and reflood stages of an *LOCA* accompanied by operation of the *ECCS* are highly complex. The computer programs must e.g. in the case of a *PWR* calculate:

- The energy sources,
  - the energy sources must include the stored energy in the fuel prior to reactor shutdown as a result of the loss of coolant,
  - the fission heat generated subsequently,
  - the radioactive decay of the fission (and neutron capture) products in the fuel rods, and
  - the heat generated by the zirconium-water (steam) reaction.

- Hydraulic parameters such as,
  - the rate of flow of water through a postulated pipe break in the primary coolant circuit,
  - rate of flow through the core and other system components and,
  - the pressure and water level in the reactor vessel during the blowdown, refill, and reflood stages.
- The heat transfer mechanisms, at various stages of the hypothetical accident, i.e.,
  - during and after blowdown,
  - during the blowdown phase.
- When the critical heat flux (CHF) is attained,
  - different heat-transfer correlations must be used in the pre-CHF and post-CHF stages,
  - following blowdown, the heat-transfer coefficients are based upon experimental data modified to assure that they are conservative.

From the above quantities, the other important properties of the core and fuel are evaluated as functions of time.

These code calculations are either performed by the fuel vendor or the utility. The fuel vendor may in some cases carry out a generic analysis for a specific fuel design which means that this fuel design will meet the *LOCA* and *RIA* fuel design criteria for the reactor types and other conditions that were used for the generic analysis. In other cases either the fuel vendor or the utility must perform reactor specific analysis to show that the *LOCA* and *RIA* criteria are met for the core loading for a specific reactor cycle.

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

In recent work in the U.S., the FRAPCON steady-state fuel rod code was modified for burnup up to 65 MWd/kg. Previously the code had been validated up to about 40 MWd/kg. Nine different models[29] within the code were found to need modification because burnup effects on the phenomena being modelled were large enough to warrant a change. NRC uses FRAPCON-3, Strosnider, 2003, to audit similar vendor codes that calculate *LOCA* stored energy, end-of-life rod pressure, gap activity, and perform other licensing analyses.

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

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

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

However, the transient codes need to incorporate models, correlations, and properties for cladding plastic stress-strain behaviour at elevated temperatures, effects of annealing, behaviour of oxides and hydrides during temperature ramps, phase changes, and large cladding deformations such as ballooning. The mechanical description of the cladding is two-dimensional ideally, but models of lower dimension are used as well. Other differences also come into models like fission gas release, which can have long-term and short-term components. The transient codes are used for analysing fuel rod response to transients and accidents like *RIA* and *LOCA* and may include failure models. In principle, the nine models[38] affected by high burnup for the steady-state fuel codes also identify the important phenomena that would need to be modified in the transient codes. In some cases, like fission gas release, the transient models would be quite different from the steady-state models. For burnup higher than 40-50 MWd/kg, special attention should be devoted to the proper modelling of thermal characteristics of the RIM-zone with its structural change and the consequent degradation of the fuel thermal conductivity. FRAPTRAN is used by NRC for special calculations and to interpret test results. For reactor power calculations, neither the industry nor the NRC are, as a rule, using 3-D neutronics codes. Postulated accidents like the rod ejection in a *PWR*, the rod drop in a *BWR*, and the *BWR* ATWS power oscillations are very localized in nature and cannot be analyzed well without 3-D kinetics codes. Recently, FRAPTRAN was updated to install the high-burnup thermal models that had been developed for the FRAPCON-3 code, which had been updated earlier.

---

[29] These models were: fission gas release; fuel thermal conductivity (including effects of burnable absorber additions); fuel swelling; fuel pellet cracking and relocation; radial power distribution; solid-solid contact gap conductance; cladding corrosion and hydriding; cladding mechanical properties and ductility; cladding axial growth.

6 CURRENT ISSUES (PETER RUDLING)

Nuclear bodies in different countries, have expressed concern about the applicability of the existing *LOCA* and *RIA* criteria to high burnup *LWR* fuel. *This concern is related to that at high burnups, some new phenomena occur that was not envisioned at the time the current LOCA and RIA criteria were developed. These criteria were based upon results from fuel rods with Zry-2 and –4 claddings that either were non-irradiated or were irradiated to a low burnup level.* The high burnup issues today, i.e., areas where there may be some lack of knowledge to correctly model *LOCA* and *RIA* behaviour, are:

- Fuel pellet rim structure development occurring at burnups exceeding about 50 MWd/kgU. The local burnup in the pellet rim exceeds the pellet average by a factor of two or more and the pellet microstructure is markedly different compared to that in the remainder of the fuel pellet. During a *RIA* event the fuel pellet temperature in the pellet periphery may become significantly higher than that during in-reactor operation. This increase in temperature may result in violent transient fission gas release causing fuel failure and fuel dispersal.

- Fuel-clad bonding, that occurs at burnups larger than about 50 MWd/kgU. It is not clear how the fuel-clad bonding impacts the fuel rods *LOCA*/*RIA* performance.

- The influence of hydrogen on *LOCA* ballooning, burst behaviour and embrittlement during reflooding and post-*LOCA* for various alloys (Zircaloy, *ZIRLO*, *M5*, *MDA*) used today

- Fuel relocation mechanism during *LOCA* and consequences.

  – What are the effects of fuel accumulation in the ballooned and burst region on the local increase in linear heat rating, surface heat flux and the local decrease in fuel-clad gap and what is the resulting impact on *PCT* and *ECR* developed during the *LOCA* event

- The effect of fuel clad oxidation prior to the *LOCA*/*RIA* event on fuel rod *LOCA*/*RIA* performance. The in-reactor fuel clad oxidation will also increase the hydrogen content of the cladding.

  In the case of a *RIA* event, increased hydrogen content with higher burnups may result in a larger degree of cladding embrittlement due to formation of:

  – More radial hydrides (in RXA material only)

  – Increased tendency to form hydride rims at the clad outer surface. Rim thickness may also increase with increased hydrogen content

  – If the in-reactor oxide becomes too thick, oxide spallation may occur that may redistribute the clad hydrogen concentration, forming hydride blisters that significantly may decrease the clad ductility during *RIA*

- The fuel coolability issue related to coolant flow blockage of highly deformed cladding with possibly relocated fuel. Results from earlier flooding experiments of unirradiated *PWR* rod lattice suggest that a blockage of 90% (the blockage was done my artificially changing the fuel rod diameter, thus the fuel rods were not ballooned) still results in a coolable geometry of the fuel assembly[31]. These results may have to be verified for high burnup ballooned rods with significant fuel relocation and nuclear heating[32], Figure 6-1, to assess:

  – What is the maximum flow blockage ratio that is acceptable to ensure coolability of an irradiated rod bundle and,

  – what is the maximum flow blockage ratio that would occur for an irradiated fuel rod bundle.

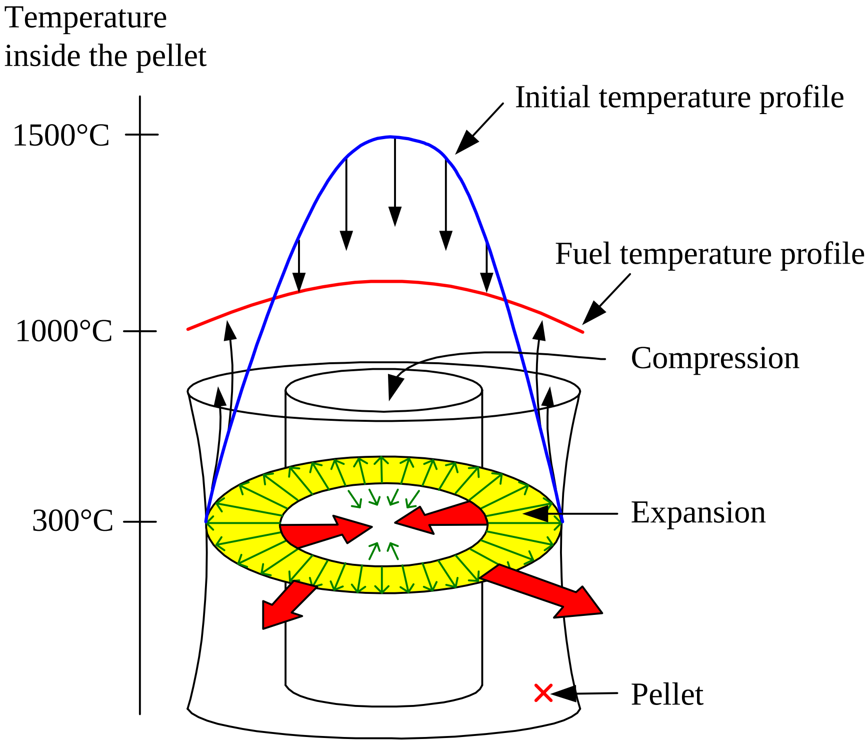


Figure 6-1: Temperature distribution in pellet before and during a *LOCA* event during nuclear heating.

[31] It is interesting to note that no fuel coolability study with a *BWR* rod lattice have been performed.
[32] During nuclear heating, the heat source is the fuel pellet as would be the case for a real *LOCA* event. In many tests, heating is carried out from the outside of the fuel rod (such as in the current experiments carried out at ANL to study *LOCA* behaviour of irradiated fuel rods). In tests where heating of the rod is done from the outside, the largest fuel pellet radial temperature will occur at the pellet periphery while during nuclear heating the corresponding maximum temperature will occur in the centre of the fuel pellet, see Figure 6-1. It may be that the difference in the fuel pellet radial temperature gradient during nuclear heating and external heating will have an impact on e.g. the fuel pellet stress state that could impact fuel relocation behaviour.

The entire database on fuel rod LOCA and RIA performance is essentially data on Zry-2 and Zry-4. Even though the more corrosion resistant alloys during base irradiation (with a resulting lower hydrogen pickup fraction before the LOCA/RIA event would occur) such as M5, ZIRLO and MDA perform at least as good as the Zry-2 and Zry-4 materials the LOCA/RIA performance of these alloys still need to be verified.

7 CURRENT PROGRAMS TO RESOLVE ISSUES (PETER RUDLING)

The major programs focusing on resolving the *LOCA* and *RIA* issues described in section 6 are summarized below, see NEA/CSNI/R(2003)9 for more details. Two types of tests are used to investigate the *LOCA*/*RIA* performance, namely separate effects and integral tests, see also section 8.

The programs related to *LOCA* are:

- In USA:
  - Argonne National Laboratory, ANL: hot cell *LOCA* tests of fuel rods and mechanical properties of cladding. The program involves integral fuel tests of high burnup fuel under simulated *LOCA* conditions (heating the rod from the outside) and high temperature oxidation studies. It also includes mechanical property tests relevant to both *LOCA* and *RIA* modeling. The entire effort is supported by analytical studies. Also the *LOCA* and *RIA* performance of new high burnup Zr-alloys will be assessed in this program.
  - Pacific Northwest National Laboratory (NRC program): steady-state and transient fuel rod codes and analysis.
  - Brookhaven National Laboratory (NRC program): neutron kinetic codes and analysis of plant transients.
- In Norway, *the Halden Reactor Project* will study the *LOCA* performance of internally heated fuel rods, nuclear heating.
- In France, the following programs have been and are being performed:
  - The French *LOCA* test separate effects program was carried out from 1991 to 2001. Four series of tests were conducted:
    - TAGCIS (1991-1993) consisted of >110 *LOCA* type thermal shock tests on as-fabricated and pre-oxidized cladding,
    - TAGCIR (1993-1996) ran 25 oxidation and thermal shock tests on cladding irradiated 5 annual cycles to 60-63 GWD/MT with 60-120μm oxide films,
    - HYDRAZIR (1996-1999) made oxidation and thermal shock tests on unirradiated pre-hydrided cladding with 0-5000 ppm H,
    - CINOG (1997-2001) made oxidation and thermal shock tests on unirradiated M5 with Zircaloy-4 as reference material.
  - Grenoble Research Center (France): high temperature fission product release tests.

- In Japan, work is carried out in the Fuel Safety Research Laboratory of JAERI, (Japan Atomic Energy Research Institute) and is sponsored by MEXT, (Ministry of Education, Culture, Sports, Science and Technology).

  The objective is to evaluate the influence of burn-up extension to a range exceeding the current limit of 55GWd/t for *BWR*s and 48GWd/t for *PWR*s on fuel behaviour under *LOCA* condition and to obtain a wide-range data base available for regulatory judgment. The *integral thermal shock test* aims to investigate the cladding resistance to failure during a *LOCA* in relation to the current *ECCS* criteria, 15% *ECR*. The program consists of oxidation rate measurement tests, tube burst tests, cladding mechanical properties tests and integral thermal shock tests, Figure 7-1 and, Figure 7-2. The comprehensive series of experiments use, pre-oxidized, pre-hydrided and/or irradiated claddings samples. The integral thermal shock tests simulate the whole *LOCA* sequence including cladding burst, double-sided cladding oxidation and thermal shock during axial constraint by reflooding, see Figure 8-16. The oxidation tests, the mechanical properties tests, and the integral thermal shock tests with the pre-hydrided cladding (non-irradiated) have been performed to investigate separate effects of hydrogen absorption during normal operation. Also some tests of irradiated cladding has just started, see Figure 3-86.

  Results obtained so far are as follows:

  – The influence of pre-hydriding on the oxidation rate varies depending on the oxidation temperature and the hydrogen concentration. The largest influence is observed at 1173 and 1223 K and the enhancement by pre-hydriding is estimated to be less than 5% for the hydrogen concentration lower than 800 wtppm in several minutes.

  – Hydride redistribution and morphology change take place by temperature changes expected in a *LOCA*. This can greatly affect the mechanical property of cladding.

  – ß-quenched cladding with high hydrogen concentrations exhibits very low ductility.

  – The integral thermal shock tests varying hydrogen content and axial load during quenching, it was shown that;

    ▪ The failure threshold generally decreased with the increase in axial tensile load.

    ▪ The influence of pre-hydriding was obviously seen on the failure threshold value under restraint conditions.

– A computer code will be developed in a couple of years to analyze the failure behaviour of a fuel rod during a *LOCA* based on the results from the current research program and international cooperation. The experimental and analytical investigation will be summarized by 2004, though further work is envisaged to support higher burn-up, in the range up to 75 GWd/t and advanced designs of fuel. The sponsor of this work is METI (Ministry of Economy, Trade and Industry).

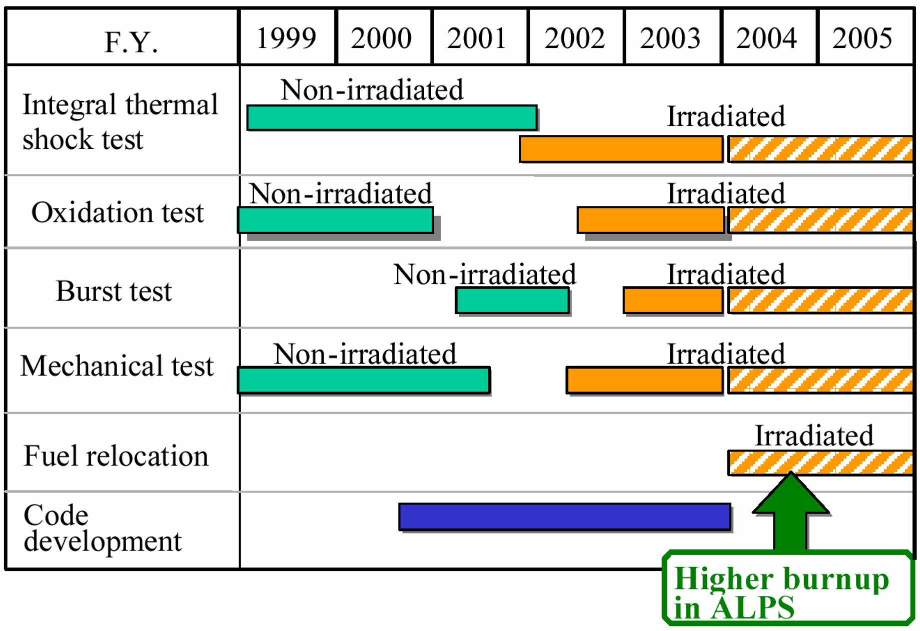


Figure 7-1: Schedule of Research Program on High Burnup Fuel Behaviour Under *LOCA* Conditions, Nagase, et al., 2002.

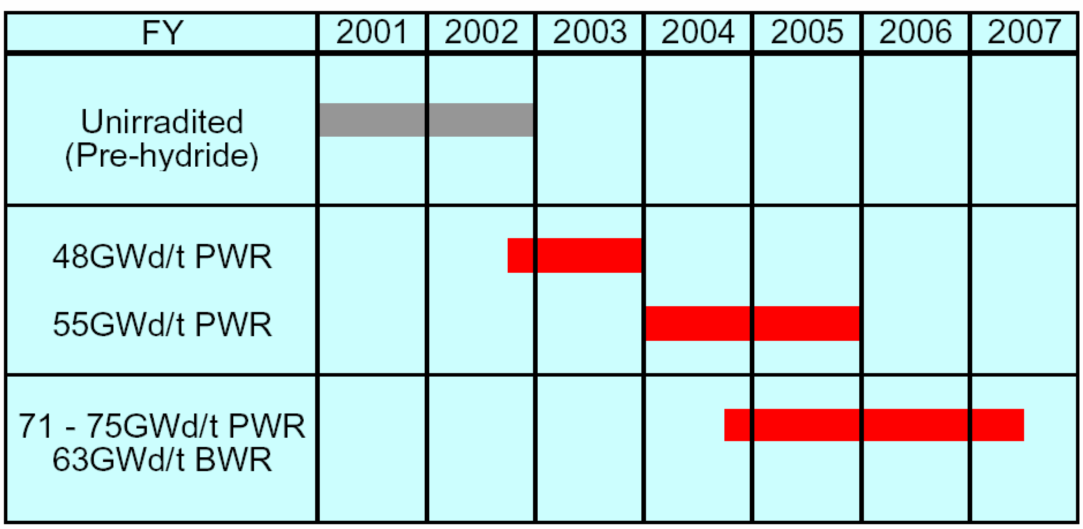


Figure 7-2: Planned thermal shock tests, Nagase & Fuketa, 2003.

8 MECHANICAL PROPERTY TESTING TECHNIQUES AND ANALYSES (RON ADAMSON)

In the R&D effort to get a better understanding of the fuel rod performance during A *LOCA* and *RIA* event as well as to establish more relevant fuel design criteria than we have today, a number of different mechanical tests are used and described in the following.

8.1 LOSS OF COOLANT ACCIDENT (LOCA)

Laboratory testing is used to determine cladding behavior during the *LOCA*. In most cases unirradiated material can be used, as the temperatures reached during even the first transient history stage are high enough to anneal the irradiation damage, Brachet, et al., 2002(a), Portier, et al., 2004. For time-temperature histories of typical *LOCA* conditions, see Figure 3-5 to Figure 3-8

8.1.1 LOCA Ballooning/Burst

Integral testing in the present sense means testing of entire fuel rod segments, either singly or in bundle arrays. The most simple case of course is a single rod heated at some controlled rate in 1 atmosphere steam with a tubing internal pressure and perhaps some axial restraint or applied stress. An example test rig from the EDGAR program is shown in Figure 8-1. Reviews of such

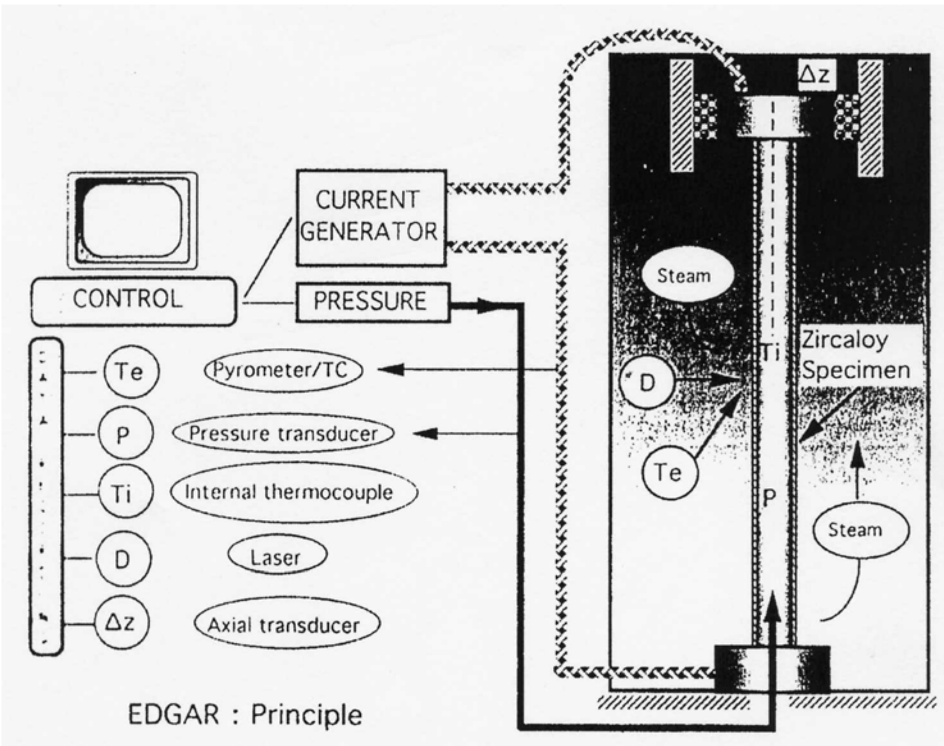


Figure 8-1: Schematic view of the EDGAR *LOCA* test facility, Forgeron, et al., 2000.

testing are given, for example, Erbacher & Leistikow, 1987, Forgeron, et al., 2000, Brachet, et al., 2002(a), and in a more simple application, Rosenbaum, et al., 1987. Results of such tests are dependent on heating rate, internal pressure, hydrogen and oxygen content, temperature non-uniformities, temperature of burst, and alloying content. As such, it is important to understand test variables when applying the duty to a specific case. Examples of the effect of temperature on Zircaloy bursting are given in Figure 8-2 and Figure 8-3. A strong temperature dependency is noted. Another important variable is temperature distribution around the cladding circumference (asimuthal ΔT), as illustrated schematically in Figure 8-4 and with data in Figure 8-5. Uniform temperature distribution as promoted in a well-controlled single rod test, produces high burst strain while asimuthal ΔT can significantly lower the strain. In practice, aximuthal ΔT tends to exist, although the effects appear to lessen with full bundles or in-reactor testing, Grandjean & Hache, 2004.

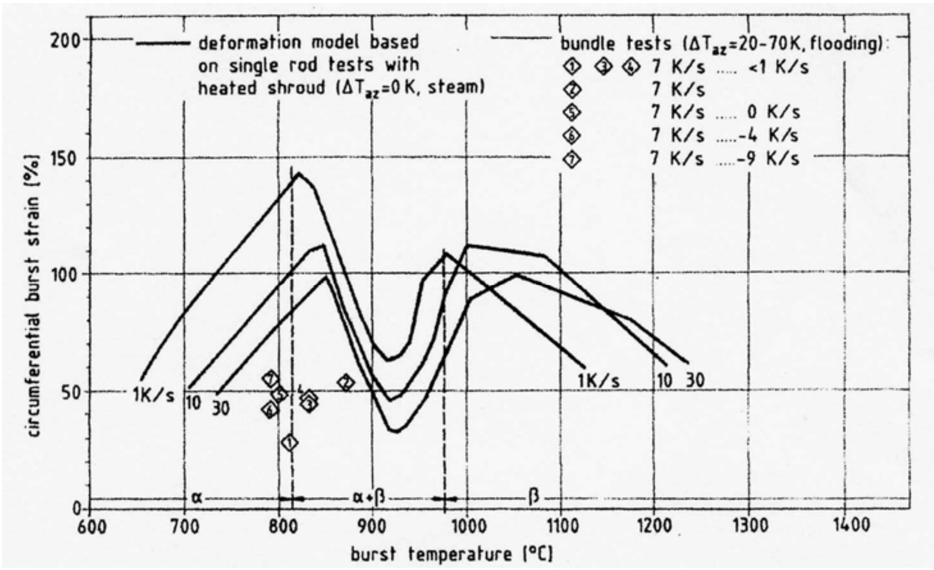


Figure 8-2: Burst strain versus burst temperature of Zircaloy-4 cladding tubes (REBEKA tests), Erbacher & Leistikow, 1987.

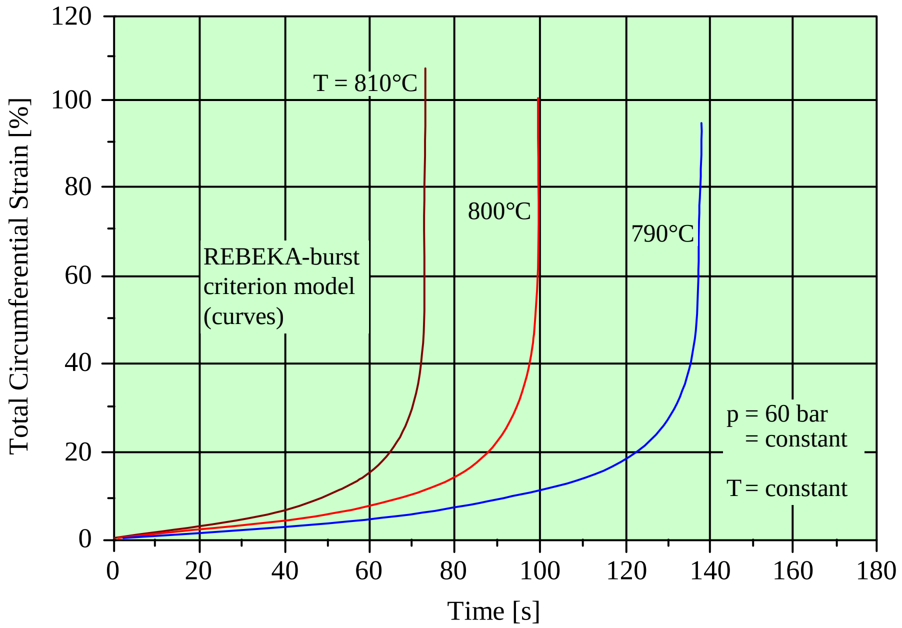


Figure 8-3: Sensitivity of Zircaloy-4 deformation to temperature, modified figure according to Erbacher & Leistikow, 1987.

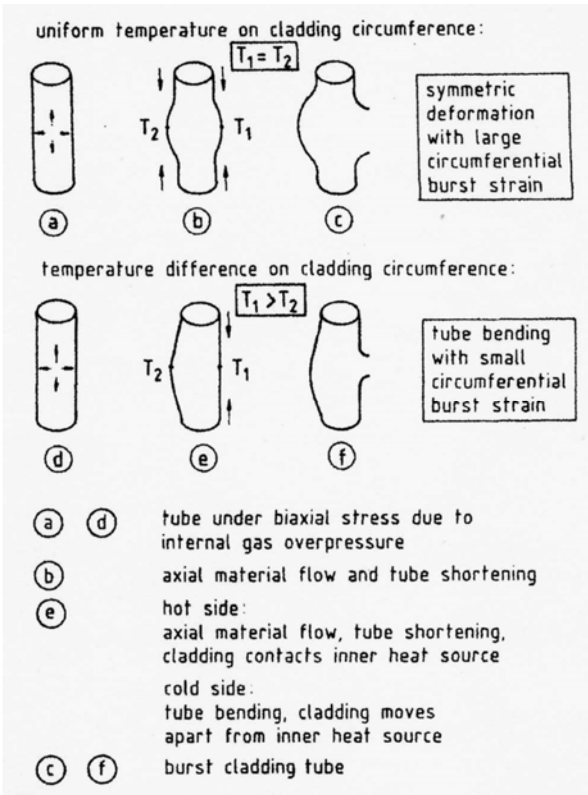


Figure 8-4: Strain anisotropy and bending of Zircaloy-4 cladding tubes in the alpha phase, Erbacher & Leistikow, 1987.

9 SUMMARY

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 non-irradiated 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 work done during the 60s and 70s were formalised in the legislating documents produced by the Nuclear Regulatory Commission in USA, *USNRC*. Most countries have criteria that are similar or identical with the safety criteria developed by *USNRC*.

There are two parts of the *USNRC* legislating documents that have relevance to *LOCA* and *RIA*, namely, Title 10 Code of Federal Regulations, Part 50, 10CFR50, and 10CFR100, NRC, 1995. The main objective of 10CFR50 and 10CFR100 is to limit radioactive impact on the environment, as follows:

- In 10CFR50, the General Design Criteria, *GDC*s are specified and interpreted in SRP, which imposes mechanical, nuclear and thermal hydraulic fuel design criteria that the fuel vendor and the utility must meet.

- In 10CFR100, it is specified that conservative dose calculations must be done to assess the potential impact on the environment during a *DBA*.

It is within the 10CFR50 and 10CFR100 documents the fuel clad criteria that:

A) Limits *PCT* and *ECR* during a *LOCA* event and the maximum ΔH during a *RIA* event to maintain coolable geometry of the fuel-these limits must never be exceeded for a class IV event (if the limits are exceeded, it is considered as a sever accident).

B) Specifies the maximum ΔH limit during cold start-up of a *BWR* above which it must be assumed that the fuel rods have failed.

C) Specifies that during full power operation in a *BWR* and *PWR* that if dry-out and dnb, respectively, occur it must be assumed that the fuel rods have failed

   The number of failed rods in B. and C. above may then be used in calculations to ensure that the dose environmental impact is below a certain value.

Fuel vendors are required by the regulatory body to perform safety analysis to show that the fuel is designed to satisfy the fuel design criteria both during transients and accidents such as *LOCA* and *RIA*. These analyses are carried out with computer codes, either as standalone codes or in a coupled manner. In these safety analysis also steady-state calculations is also needed to establish the initial conditions prior to the transient or accident.

It is essential that the performance of the fuel rods during a *LOCA* and *RIA* is well-known so its behaviour can be correctly modelled by which it can be shown that the design criteria of the fuel can be met.

The prediction of the behaviour of fuel rods during a *LOCA* or *RIA* is a difficult task given the complexity of the system being modelled, see Figure 9-1 and Figure 9-2.

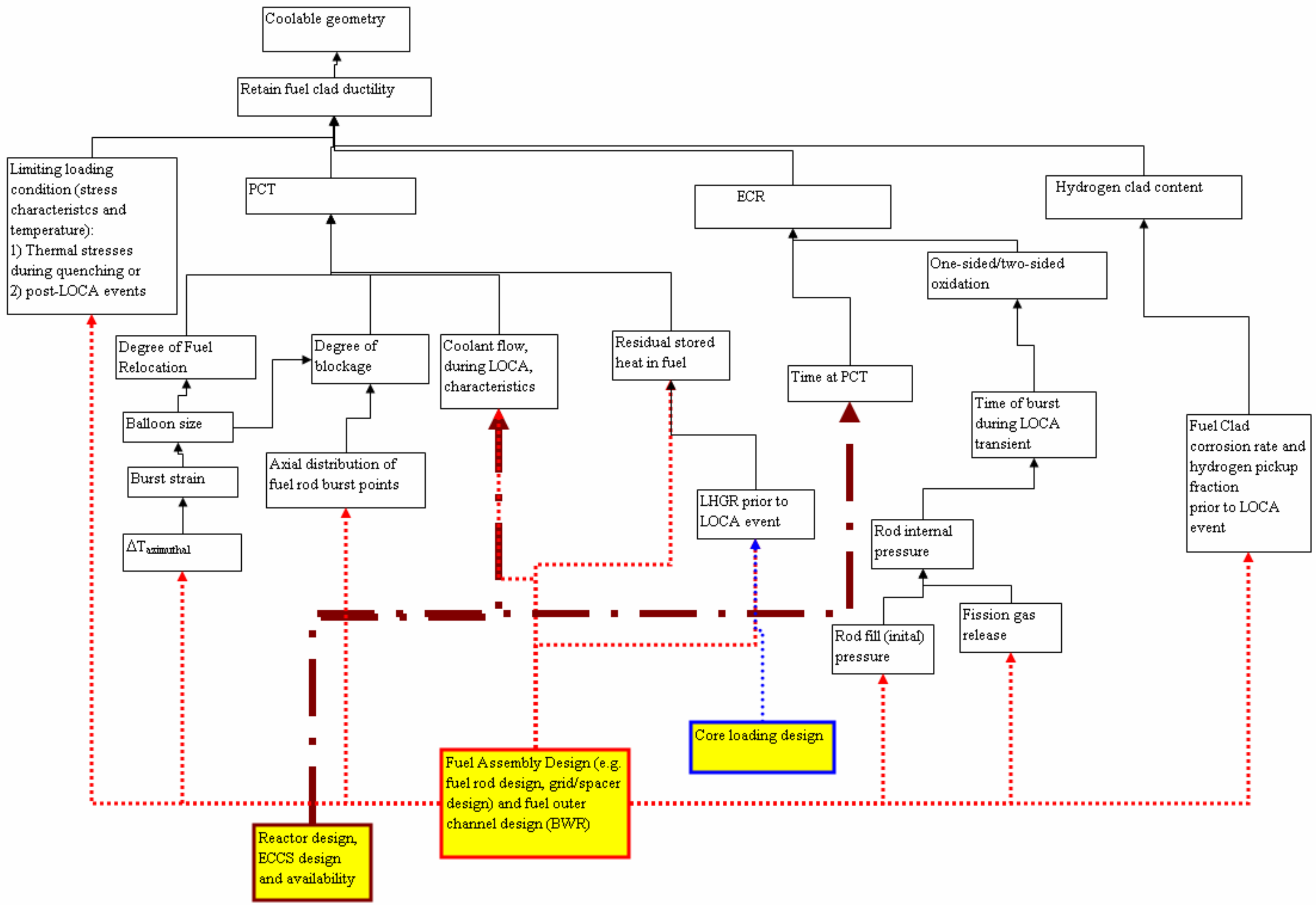


Figure 9-1: Parameters impacting *LOCA* fuel performance.

10 REFERENCES

Abriata J. P. et al., Bull. Alloy Phase Diagrams 7(2), pp. 116-124, 1986.

Adamson R. and Rudling P., "Mechanical Properties of Zirconium Alloys", ZIRAT 6 Special Topics Report, pp. 67-88, 2001.

Adamson R., et al., "*High Burnup Fuel Issues*", ZIRAT-8 Special Topics Report, 2003.

Adroguer B., Hueber C., and Trotabas M., "*Behavior of PWR Fuel in LOCA Conditions, PHEBUS Test 215 P*," presented at OECD-NEA-CSNI/IAEA Specialists' Meeting on Water Reactor Fuel Safety and Fission Product Release in Off-Normal and Accident Conditions, Ris0 National Laboratory, Denmark, May 1983.

Agee L. J., Dias A. F., Eisenhart L. D. and Engel R. E., "*Realistic scoping study of reactivity insertion accidents for a typical PWR and BWR core*", Proc. CSNI specialist meeting on transient behaviour of high burnup fuel, pp 291-304, Cadarache, France, September 12-14, 1995.

Andersson T. and Wilson A., "*Ductility of Zircaloy Canning Tubes in Relation to Stess Ratio in Biaxial Testing*", IN: Zirconium in the nuclear industry: 4th int. symp. ASTM STP-681, J. H. Schemel and T. P. Papazoglou (eds), American Sociaty for Testing Materials, pp. 60-71, 1979.

Aomi et al., Proceedings of the ANS Conference, p. 231, Park City, USA, 2000.

Arsene S. and Bai J. B., "*A new approach to measuring transverse properties of structural tubing by a ring tes*t", Laboratory MSS/MAT, Ecole Centrale Paris, CNRS URA 850, 92295 Chatenay Malabry Cedex, France 1996.

Arsene S., Bai J. B. and Bompard P., "*Hydride Embrittlement and Irradiation Effects on the Hoop Mechanical Poperties of PWR and BWR Cladding Tubes*", Metallurgical and Material Transactions A, Volume 34A, March, 2003.

Asmolov V., Yegorova L., Lioutov K., Smirnov V., Goryachev A., Chesanov V. and Prokhorov V., "*Understanding LOCA-Related Ductility In E110 Cladding*", Proc: 30th Nuclear Safety Conference, Washington, DC, USA, October, 2002.

Bai J. B., "*Influence of an oxide layer on the hydride embrittlement in Zircaloy-4*", Scripta Metallurgica at Material, 29, pp 617-622, 1993.

Baker L. and Just L. C. "*Studies of metal-water reactions at high temperatures. III. Experimental and theoretical studies of the zirconium-water reaction*", ANL 6548, May 1962.

Balourdet M., Bernaudat C., Basini V. and Hourdequin N., "*The PROMETRA Programme: Assessment of Mechanical Properties of Zircaloy 4 Fuel Properties During an RIA*", Transactions of the 15th International Conference on Structural Mechanics in Reactor Technology (SMIRT-15), pp. II-485-492, Seoul, Korea, August 15-20, 1999.

Barrio F. J. and Herranz L. E., "*Predictive methodology of a cladding failure criterion based on PROMETRA database: Application to CABRI-CIP01 experiment*", Proc. EHPG Meeting, Gol, September 8-13, 2002. OECD Halden Reactor Project Report HPR-359, 'Volume 2, Halden, Norway.

Barrio F. J., Garcia J. C., Vallejo I., Barrio M. T. and Herranz L. E., "*Influence of Power Pulse Width on High Burnup Fuel Rod Response to Reactivity Insertion Accidents: A Study Baed on the CABRI-CIP01 Experiment*", Enlarged Halden Program Group Meeting, Storefjell, Norway September, 2002.

Bates D. W., Koss D. A., Motta A. T. and Majumdar S., "*Influence of Specimen Design on the Deformation and Fracture of Zircaloy Cladding*", Proceedings of the ANS Int. Topical Mtg. on LWR Fuel Performance, American Nuclear Society, pp. 1201-1210, Park City, UT, 2000.

Bibilashvili Yu. K., Sokolov N. B., Salatov A. V., Tonkov V. Yu., Fedotov P. V., Andreeva-Andrievskaya L. N., Deniskin V. P., Nalivaev V. I., Parshin N. Ya., Afanasyev P. G., Konstantinov V. S., Semishkin V. P. and Shumski A. M., "*WWER-1000 Type Fuel Assembly Tests on Electroheated Facilities In LOCA Simulating Conditions*", IAEA-TECDOC-1320, 2002.

Bouffioux P. and Rupa N., "*Impact of Hydrogen on Plasticity and Creep of Unirradiated Zry-4 Cladding Tubes*", Zirconium in the Nuclear Industry: Twelfth International Symposium, ASTM STP 1354, Sabol G. and Moan G., Eds. American Society for Testing and Materials, pp. 399-422, West Conshohocken, PA, 2000.

Bouffioux P. and Legras L., "*Effect of hydriding on the residual cold work recovery and creep of zircaloy 4 cladding tubes*", Proc: ANS conference on fuel performance, Park City, USA, 2000.

Brachet J. C., Portier L., Forgeron T., Hivroz J., Hamon D., Guilbert T., Bredel T., Yvon P., Mardon J–P. and, Jaques P., "*Influence of hydrogen content on the α/β phase tranformation temperatures and on the thermal-mechanical behaviour of Zry-4, M4 and M5 (ZrNbO) alloys during the first phase of LOCA transient*", Zirconium in the Nuclear Industry: Thirteenth International Symposium, ASTM STP 1423, G. D. Moan and P. Rudling, Eds., ASTM International, West Conshohocken, PA, 2002(a).

Brachet J. C., Pelchat J., Hamon D., Maury R., Jacques P. and, Mardon J-P., "*Mechanical Behaviour At Room Temperature And Metallurgical Study Of Low-Tin Zy-4 And M5tm (Zr-Nbo) Alloys After Oxidation At 1100°C and Quenching*", IAEA-TECDOC-1320, 2002(b).

Brachet J. C., Portier L., Hamon D., Trouslard Ph., Urvoy S., Rabeau V., Lesbros A. and Mardon J. P., "*Quantification of the α(O) and Prior-β phase fractions and their oxygen contents in high temperature (HT) oxidised Zr alloys (Zy-4, M5™)*", SEGFSM Topical Meeting on LOCA Issues, Argonne National Laboratory, May 25-26, 2004.

Böhmert, Dietrich M., Linek J., "*Comparative analysis of high temperature corrosion of ZrNbl and Zircaloy-4*", Nucl. Eng. Design, 147, pp. 53-62, 1993.

Chapman R H., "*Multi-rod burst test program: progress report, July-December 1977*". ORNL/NUREG/TM-200, June 1978.

Chung H. M., Garde A. M. and Kassner T. F., "*Deformation and Rupture Behaviour of Ziradol Cladding under Simulated Loss-of-Coolant Accident Conditions*", Zirconium in the Nuclear Industry, ASTM STP 633, A. L. Lowe, Jr. and G. W. Party, Eds., American Society for Testing and Materials, pp. 82-97, 1977.

Chung H. M. and Kassner T. F., "*Embrittlement Criteria for Zircaloy Fuel Cladding Applicable to Accident Situations in Light-Water Reactors*", NUREG/CR-1344, January 1980.

Chung H. M. and Kassner T. F., "*Cladding metallurgy and fracture behavior during reactivity-initiated accidents at high burnup*", Nucl. Engng. Design, 186, pp 411-427, 1998.

Chung H. M., "*Fundamental metallurgical aspects of axial splitting in zircaloy cladding*", Proc. ANS Topical Meeting on Light Water Reactor Fuel Performance, Park City, Utah, April 10-13, 2000.

Chung H., Strain R. V, Bray T. and Billone M. C., Argonne Nat. Lab., USA, "*Progress in ANL/USNRC/EPRI Program on LOCA*", Ref: Proceeding of the Topical Meeting on LOCA Fuel Safety Critera, NEA/CSNI/R(2001)18, Aix-en-Provence, 2001.

Chung H. M., "*The Effects of Aliovalent Elements on Nodular Oxidation of Zr-Base Alloys*", Nuclear Safety Research Conference (NSRC-2003), Washington, DC, 2003.

Chung, H. M., "*Differences in Behaviour of Sn and Nb in Zirconium Metal and Zirconium Dioxide*", SEGFSM Topical Meeting on LOCA Issues Argonne National Laboratory, May 25-26, 2004.

Cox B., "*Oxidation of Zirconium and its Alloys*", Advances in Corrosion Sclence and Technology. Plenum Press 1976.

Cox, B., "*Pellet-clad interaction (PCI) failures of zirconium alloy fuel cladding – a review*", J. Nucl. Mat. 172, pp 249-292, 1990.

APPENDIX A - FUEL SAFETY CRITERIA

The fuel safety criteria legislation are provided in the documents 10CFR50 and 10CFR100, both of which have the objective to minimise the potential impact of radiation related damage to the public, Figure A-1.

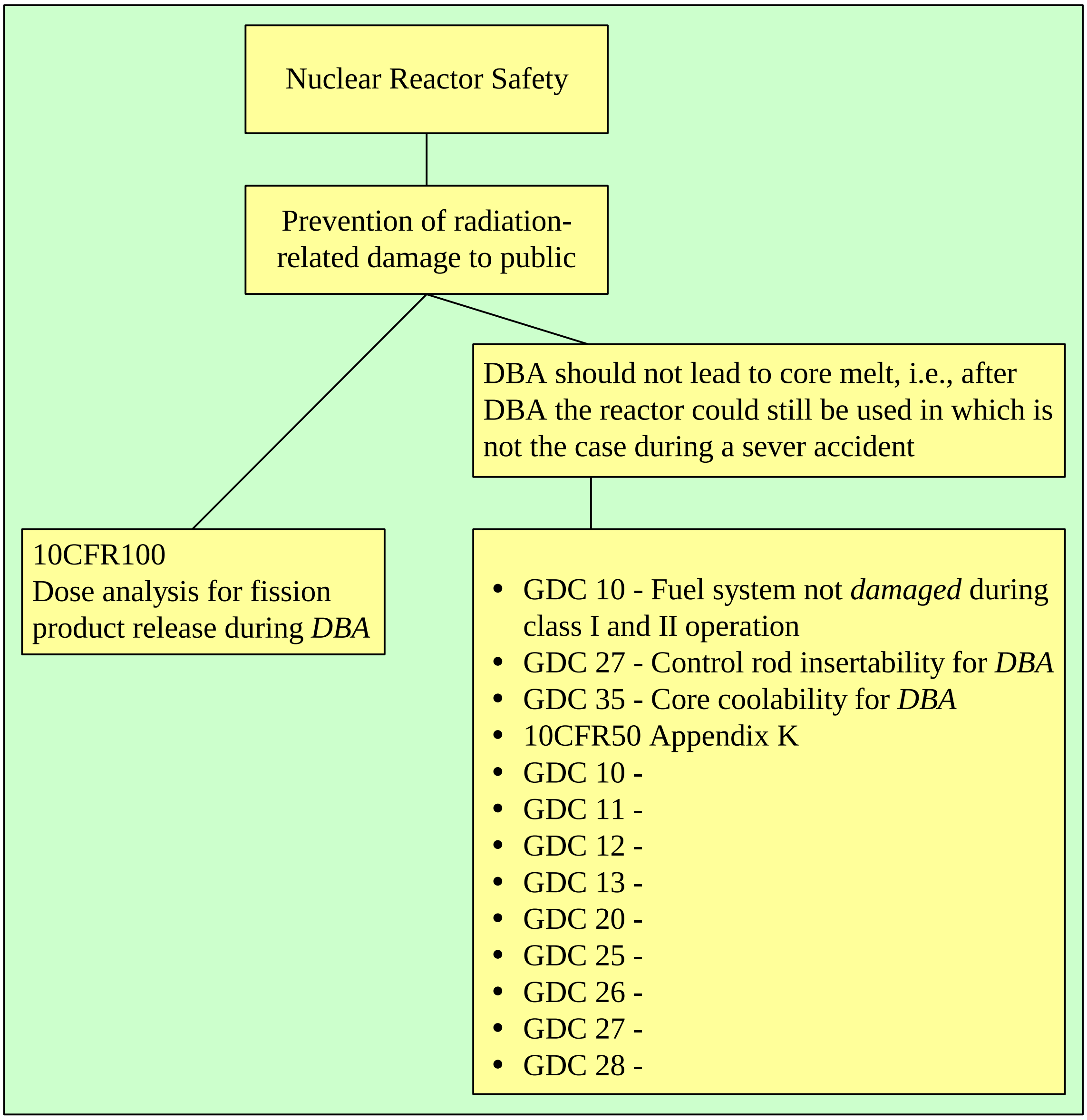


Figure A-1: Relation between 10CFR50 and 10CFR100. *DBA* stands for Design Basis Accidents.

The 10CFR50 and 10CFR100 are interpreted in the Standard Review Plan, SRP[33]. The contents of sections 4.2-4.4 in SRP are summarised in the following, see also Fig. 2-1.

A.1 MECHANICAL DESIGN REVIEW

The objective of the mechanical design review is to ensure that: 1) the *fuel system*[34] is *not damaged*[35] during class I and II operation, 2) *fuel system damage* is never so severe as to prevent control rod insertion, 3) number of *fuel rod failures*[36] is not underestimated for *DBA* and, 4) *coolability*[37] is always maintained.

- *Fuel system damage* - to meet the requirements of *GDC*10 as it relates to SAFDLs for normal operation, including Anticipated Operational Occurrences, AOOs, fuel system damage criteria should be given for all known damage mechanisms related to:
  - Stress, strain or loads
  - Fatigue
  - Fretting
  - Oxidation, hydriding, CRUD deposition
  - Dimensional changes
  - Rod internal pressure
  - Hydraulic loads
  - Loss of control rod reactivity, i.e., control rod reactivity should be maintained
- *Fuel rod failure* to meet the requirements of *GDC*10 as it relates to SAFDLs for normal operation, including AOOs and 10CFR100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be given for all known fuel rod failure mechanisms related to:
  - Internal hydriding
  - Cladding collapse
  - Overheating of cladding. Thermal margin criteria DNBR and CPR must not be violated
  - Overheating of pellets. Centreline melting not allowed

---

33 SRP (NUREG-0800), 1981.
34 *Fuel system* – Fuel assembly at its components (including fuel rods) and control rods
35 ”*Not damaged*” means that the fuel rods should not fail and 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 *GDC*10 and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits, SAFDLs.
36 *Fuel rod failures* means that the fuel rod leaks and the first fission products barrier is breached
37 *Coolability* – The fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after the accident.

- Excessive Fuel Enthalpy for *RIA*:
  - *BWR* at zero or low power – fuel rod failures are assumed to occur if ΔH > 170 cal/g,
  - *BWR* and *PWR* at full power – the thermal margins (*CPR* and *DNBR*) are used as fuel failure criteria
- PCI
- Bursting related to *LOCA*
- Mechanical fracturing of fuel rod during *LOCA* or seismic event. Cladding integrity may be assumed if the applied stress is less than 90 % of the irradiated yield strength at the appropriate temperature.

• *Fuel coolability* – Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalised cladding melting, gross structural deformation and extreme coplanar fuel rod ballooning more described in the following:

- Cladding embrittlement – To ensure that unacceptable clad embrittlment will not occur the Peak Cladding Temperature, *PCT* must be below 2200F and maximum cladding oxidation must be below 17 %
- Violent expulsion of fuel- To prevent widespread fragmentation and dispersal of fuel and avoiding the generation of pressure pulses in the primary system during a *RIA*, ΔH < 280 cal/g,
- Fuel rod ballooning - To meet the requirements of Appendix K of 10CFR50 as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution. Burst strain and flow blockage models must be based on applicable data in such a way that: (1) the temperature and differential pressure at which the cladding will rupture are properly estimated, (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated.
- Structural deformation due to externally applied forces– This addresses the combined effects of *LOCA* and subsequently an independent seismic event (not as a result of the *LOCA*)

APPENDIX B - NUCLEAR REACTOR KINETICS

B.1 CONTROL ELEMENT EJECTION

The sudden ejection of a control element (or control-rod duster), as a result of a mechanical failure, is a conceivable but improbable accident. In most situations, the increase in reactivity (and thermal power) would be small and could be handled by the reactor protection system.

One design basis accident is the ejection of a control element from an operating reactor leading to a power excursion of sufficient magnitude to cause some damage to the fuel cladding. Observations made with test reactors have shown that the amount of damage resulting from the ejection of a control element would be governed mainly by the energy generated as a result of the excursion. This in turn depends on the reactivity worth of the ejected element and the power distribution attained by the remaining control-element pattern.

B.2 REACTIVITY COEFFICIENTS

The concept of reactivity coefficients has been introduced in order to simplify the analytical treatment, e.g. quantifying the feedback reactivities in the point kinetic equation and increase in the understanding of reactivity changes due to various physical parameters. Reactivity coefficients are thus an analytical matter; in terms of *LWR* safety criteria, there is a general requirement that the total of all reactivity coefficients be negative when the reactor is critical, for providing negative reactivity feedback.

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

- Fuel temperature
- Moderator (coolant) temperature
- Steam volume (void) fraction in the coolant
- System pressure

The fuel temperature or Doppler coefficient responds promptly to the enthalpy deposited in the fuel, whereas the other coefficients are delayed. The fuel time constant, which depends mainly on the fuel specific heat, conductivity and diameter, affects the time delay of changes in moderator temperature and void fraction.

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

In *PWR* under normal operation sequences there is no void in the core. However, in the case of abnormal events like loss of primary coolant or loss of pressure the coolant may start to boil and void appears and reduces the neutron absorption in boron which results in a positive contribution to the void coefficient. At operating temperature when the boron concentration is low, this effect will be small and the void coefficient remains negative. At low temperature when the boron concentration is high, the effect is large and the void coefficient may turn positive. This is the reason for the maximum Boron concentration of about 1700-1800 ppm in the coolant.

The system pressure in a *BWR* is related to the saturation temperature of the moderator. Depressurization of the system will cause production of steam bubbles in the water. Such an event introduces a negative reactivity change in a *BWR*.

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

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

B.3 THE FISSION PROCESS

The neutrons released in fission can be divided into two categories, namely, *prompt neutrons* and *delayed neutrons*. The former, which constitute more than 99 percent of the total fission neutrons, are released within 10^{-14} s (or less) of the instant of fission. The emission of the prompt neutrons ceases immediately after fission has occurred, but the delayed neutrons continue to be expelled from the fission fragments over a period of several minutes, their intensity falling off rapidly with time. The half-life of the delayed neutrons vary to some extent with the fissile nuclide and ranges approximately from 0.2 to 55 seconds. The average (total) numbers of neutrons ν liberated for each neutron absorbed in a fission reaction ranges from 2-3 for 0.0253-eV neutrons in the uranium-233, uranium-235 and plutonium-239.

B.4 NUCLEAR REACTOR KINETICS AND CONTROL

When a reactor is in a critical state, in which just as many neutrons are produced as are lost, no distinction need be made between the prompt and delayed fission neutrons. If the circumstances are such, however, that the neutron density varies with time, the delay in the production of some of the neutrons has important consequences. In spite of the small fraction of delayed neutrons, e.g., only about 0.65 percent in the fission of uranium-235 and less for the other fissile nuclides, these neutrons have a profound influence on the control of nuclear reactors.

β represents the fraction of the fission neutrons that are delayed, then 1- β represents the prompt-neutron fraction.

The fractional departure of a system from criticality is often expressed by the *reactivity*, ρ, and defined by:

$$\rho \equiv \frac{k_{eff} - 1}{k_{eff}}$$

When a reactor is critical on prompt neutrons alone, it is said to be *prompt critical*. I can be shown that a reactor becomes prompt critical when

$$\rho = \beta$$

i.e., when the reactivity is equal to the fraction of delayed neutrons. A thermal reactor in which the fissile material is uranium-235 becomes prompt critical when ρ = 0.0065, so that

$$k_{eff} = \frac{1}{1-\rho}$$

is then close to 1.0065.

The prompt critical condition is sometimes used as the basis for the reactivity unit *dollar* (*$*) which is defined as

$$Reactivity\ in\ dollars \equiv \frac{\rho}{\beta}$$

When a reactor is prompt critical, the reactor has a reactivity of exactly one dollar.

B.4.1 Neutron Diffusion in Multiplying Systems

In the critical (or steady state) condition, just as many neutrons are produced by fission as are lost by absorption and by leakage from the reactor in a given time. The *infinite multiplication factor*, represented by the symbol k_∞ is related to a hypothetical system of infinite size, where there is no loss of neutrons by leakage. Neutrons are produced by fission and are removed only by absorption in the various materials present in this system. The infinite multiplication factor is then defined by

$k_\infty =$ Rate of neutron production/Rate of neutron absorption

The condition for criticality, i.e., for a self-sustaining fission chain to be possible, in the infinite system is that the rate of neutron production should be equal to the rate of absorption in the absence of an extraneous source, i.e., k_∞ = l.

however, in a system of finite size, some neutrons are lost by leakage. The criticality condition is then conveniently defined in terms of the *effective multiplication factor*; k_{eff}. The effective multiplication factor is then given by

k_{eff}= Rate of neutron production/(Rate of neutron absorption + Rate of neutron leackage)

APPENDIX C - THERMAL HYDRAULICS

C.1 BOILING HEAT TRANSFER

C.1.1 Pool Boiling

Suppose the temperature difference between the heated surface and the liquid *saturation temperature*[42], i.e., $t_s - t_{sat}$, is steadily increased; the corresponding variation in the heat flux, i.e., q/A, across the surface is then as shown in Figure C-1. Although the data in this figure are representative of natural convection boiling from a heated surface in a pool of water at atmospheric pressure and a liquid temperature of 100°C, some of the same general characteristics apply to forced convection boiling and to other pressure and temperature conditions.

The curve for pool boiling can be divided in to a number of regions, in each of which the mechanism of heat transfer is somewhat different from that in the others.

Until the heated surface exceeds the saturation temperature by a small amount, heat is transferred by single-phase convection; this occurs in region I. The system is heated by slightly superheated liquid rising to the liquid pool surface where evaporation occurs.

In region II, vapour bubbles form at the heated surface; this is the *nucleate boiling* range in which formation of bubbles occurs upon nuclei, such as solid particles or gas adsorbed on the surface, or gas dissolved in the liquid. The steep slope of the heat flux curve in region II is a result of the mixing of the liquid caused by the motion of the vapour bubbles. A maximum flux is attained when the bubbles become so dense that they coalesce and form a vapour film over the heated surface manifesting the change from region II to III.

In region III, the heat must then pass through the vapour film by a combined mechanism of conduction and radiation, neither of which is particularly effective in this temperature range. Consequently, beyond the maximum, the heat flux decreases appreciably despite an increase in temperature. The maximum flux, which is a design limitation, is referred to as the DNB, *departure form nucleate boiling* value. In region III, the film is unstable; it spreads over a part of the heated surface and then breaks down. Under these conditions, some areas of the surface exhibit violent nucleate boiling, while *film boiling*, due to heat transfer, occurs in other areas.

For sufficiently high values of $t_s - t_{sat}$, region IV, the film becomes stable, and the entire heated surface is covered by a thin layer of vapour; boiling is then exclusively of the film type. If attempts are made to attain large heat fluxes with film boiling, as high as those possible with nucleate boiling, for example, the temperature of the heated surface may become so high as to result in damage to the material being heated. This is called a “burnout”.

---

[42] The saturation temperature is the temperature of the saturated vapour, i.e., saturated steam, at the existing pressure.

Subcooled boiling occurs when the bulk temperature of the liquid is below saturation but that adjacent to the heated surface is above saturation. Vapour bubbles form at this surface but collapse when leaving the surface in the colder bulk liquid, thus there is no net generation of vapour. Very high heat fluxes can be obtained under these conditions; When the bulk temperature of the liquid reaches the saturation point, the vapour bubbles no longer collapse and then bulk boiling occurs.

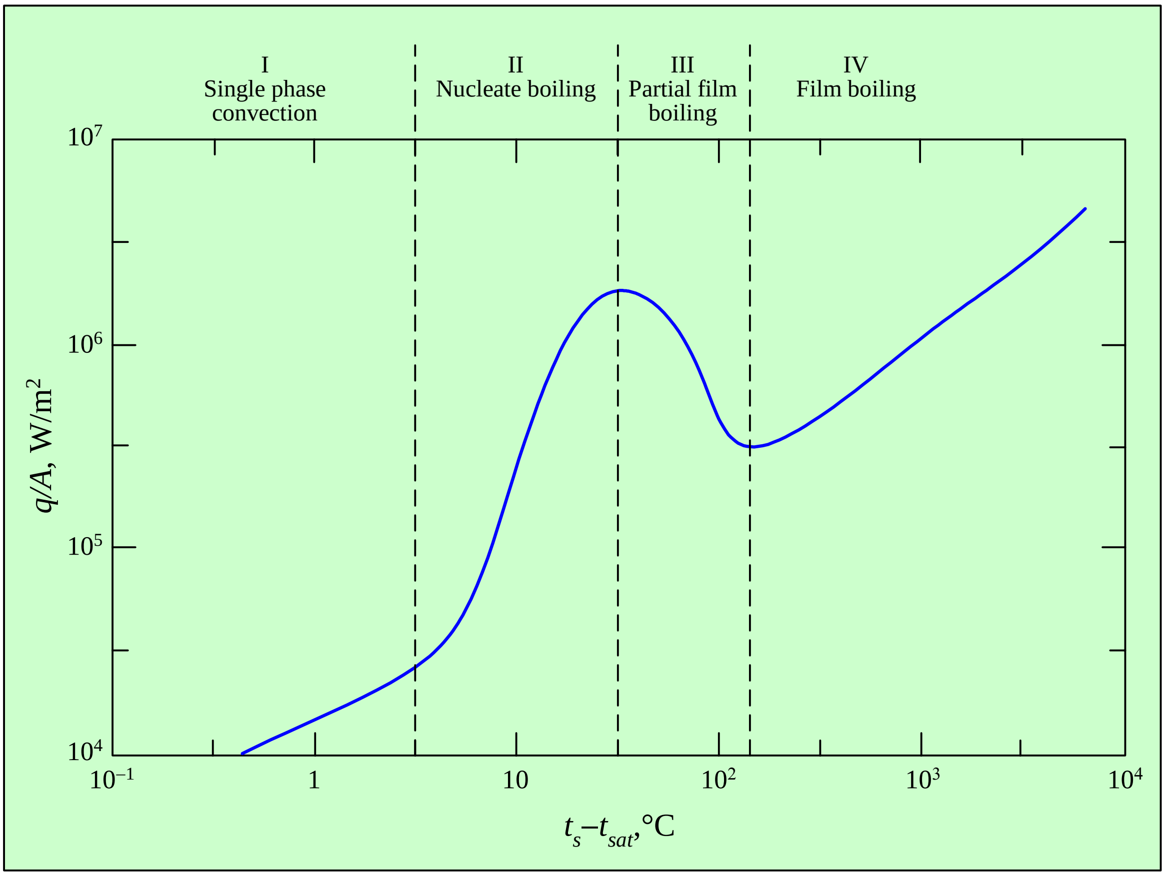


Figure C-1: Variation of heat flux with surface-liquid temperature difference in pool boiling.

C.1.2 Flow Boiling

In practical reactor systems the coolant is not stationary, and the boiling which takes place, called *flow boiling*, is hydrodynamically quite different from pool boiling. Flow boiling commonly occurs under forced-convection conditions, as in boiling-water reactors (*BWR*s) and to some extent in pressurized-water reactors (*PWR*s).

Suppose that water, below the saturation temperature, is forced through a channel between or around the solid fuel elements of a reactor; heat is then transferred from the solid surface (or wall) to the water. As long as the fuel-wall temperature, which increases along its length remains below the steam saturation temperature, single-phase heat transfer only will occur. In *PWR*s, the pressure on the cooling system is increased in order to raise the saturation temperature and thus prevent bulk boiling, but some local boiling is tolerated;

The various flow-boiling regimes of the coolant along a typical reactor fuel rod are shown in Figure C-2. Here, t_m is the bulk temperature of the liquid. At first, the only effect is an increase in the heat of the coolant, and this is followed by a region of subcooled boiling. After this there is the region in which ordinary nucleate boiling occurs accompanied by steam generation, i.e., bulk boiling. In this section of the fuel element the heat-transfer rate is high.

Consider next the flow situation along the channel as boiling progresses. The mechanism is quite complex and dependent upon how the vapour and liquid, flowing in the same direction, distribute themselves in the channel. In the local boiling region, bubbles grow and are carried along in the superheated region close to the wall, but condense on being mixed with the subcooled liquid in the flow channel. After enough heat is transferred to the flowing coolant, so that its temperature is above saturation, the vapour bubbles no longer collapse but are carried along in the main body of the stream somewhat uniformly distributed. At a sufficiently high vapour fraction, a change in the flow regime occurs wherein the vapour phase becomes continuous in the central core of the channel. Liquid drops are dispersed in this vapour core while vapour bubbles are dispersed in a continuous layer of liquid flowing along the heated wall.

APPENDIX D - PULSE REACTOR TEST DATA

D.1 SPERT-CDC TESTS

The total deposited energies for the pre-irradiated test rods in SPERT-CDC, along with their respective burnups, are presented in Table D-1. The energy deposition was determined by measuring the activity of a cobalt wire located in the vicinity of the test capsule. The technique provides the total energy deposition during the transient and has an accuracy of ±12 %. Around 10 to 20 % of the energy deposition occurs after the power pulse, i.e. during low powers prior to reactor scram. This delayed energy deposition does not contribute to the final fuel enthalpy, and the radially averaged peak fuel enthalpy, $\overline{H}_p$, was consequently estimated using a correction of 17% to account for the delayed energy deposition, MacDonald, et al., 1980.

Several of the low-burnup rods failed during or following the power pulse. Also, both of the failed high-burnup rods (rods 756 and 859) exhibited brittle-type clad fracture. Rod 756 failed at $\overline{H}_p$ = 599 Jg^{-1}, whereas rod 859 failed at $\overline{H}_p$ = 356 Jg^{-1}. A large hydride blister was found in rod 859, and the cause of failure for both rod 756 and 859 was attributed to heavy accumulations of zirconium hydride. Moreover, the non-prototypical test conditions could also have contributed to the cladding failure. The cold initial cladding temperatures in combination with the narrow power pulses, which were utilised in the SPERT-CDC tests, resulted in relatively low cladding temperatures at the time of maximum cladding stresses.

Table D-1: SPERT-CDC tests on pre-irradiated fuel rods. Data compiled from MacDonald, et al., 1980 and Meyer, et al., 1997.

Test *ID*	Fuel burnup [$MWdkg^{-1}U^{-1}$]	Pulse width [ms]	Energy deposition [Jg^{-1}]	Peak fuel enthalpy [Jg^{-1}]	Failure enthalpy [Jg^{-1}]	Fuel dispersal
571	4.55	31	674	574	Survived	-
568	3.84	24	833	674	615	Yes
567	3.10	18	1105	896	896	Yes
569	4.14	14	1457	1181	1181	Yes
703	1.14	15	804	682	Survived	-
709	0.99	13	996	846	846	Yes
685	13.1	27	779	662	Survived	-
684	12.9	20	837	712	Survived	-
756	32.7	17	737	599	599	No
859	31.8	16	795	645	356	No

The applicability of the SPERT-CDC test results to *LWR* conditions must be questioned, since the operating conditions for the test rods before transient testing were not representative of those for *LWR* fuel. The rods were pre-irradiated in the ETR facility at very high linear heat generation rate, 46-67 kWm^{-1}, resulting in fuel restructuring and central hole formation. Hence, these tests are not typical of fuel rods in commercial light water reactors.

D.2 PBF TESTS

The main objective of the *RIA* 1-1 tests was to provide a comparison of irradiated and un-irradiated fuel behaviour, by testing two pairs of Saxton (*PWR*) rods at the US NRC licensing fuel enthalpy limit of 280 $calg^{-1}$ (1172 Jg^{-1}) UO_2. Two of the rods were pre-irradiated in Saxton to a burnup of 4.6 $MWdkg^{-1}U^{-1}$, and these rods failed by rod fragmentation and fuel dispersal at $\overline{H}_p$ = 1193 Jg^{-1} (285 $calg^{-1}$). See Table D-2.

The *RIA* 1-2 tests were conducted using four individually shrouded fuel rods, which had been irradiated up to a burnup of 5.2 $MWdkg^{-1}U^{-1}$ in the Saxton reactor. Two of the rods were opened and back-filled to about 2.4 MPa to simulate end-of-life rod internal pressure. One rod was opened, instrumented and back-filled to a rod internal pressure of 0.105 MPa. The four fuel rods used in tests *RIA* 1-2 were subjected to a radially averaged peak fuel enthalpy of $\overline{H}_p$ = 775 Jg^{-1} (185 $calg^{-1}$).

Table D-2: Summary of PBF tests. Data compiled from MacDonald, et al., 1980 and Meyer, et al., 1997.

Test	Rod *ID*	Fuel burnup [$MWdkg^{-1}U^{-1}$]	Pulse width [ms]	Energy deposition [Jg^{-1}]	Peak fuel enthalpy [Jg^{-1}]	Failure enthalpy [Jg^{-1}]	Fuel dispersal
RIA 1-1	801-1	4.6	13	1528	1193	1193	Yes
	801-2	4.7	13	1528	1193	1193	Yes
	801-3	0.0	13	1528	1193	1193	No
	801-4	0.0	13	1528	1193	1193	No
RIA 1-2	802-1	5.2	16	1005	775	Survived	-
	802-2	5.1	16	1005	775	Survived	-
	802-3	4.4	16	1005	775	586	No
	802-4	4.5	16	1005	775	Survived	-
RIA 1-4	804-1	6.1	11	1235	1160	<<1160	No
	804-3	5.5	11	1235	1160	<<1160	No
	804-7	5.9	11	1235	1160	<<1160	No
	804-9	5.7	11	1235	1160	<<1160	No
	804-10	4.4	11	1130	1068	<<1068	No
	804-4	5.0	11	1130	1068	<<1068	No
	804-6	5.1	11	1130	1068	<<1068	No
	804-8	4.7	11	1130	1068	<<1068	No
	804-5	5.5	11	1026	980	980	No

Only one rod failed in the *RIA* 1-2 experiment. The failure was caused by 22 small (<1 cm long) longitudinal cracks, starting at about 14.5 cm and extending to about 68.1 cm from the bottom of the 91 cm long fuel stack. The radial average fuel enthalpy at the 14.5 and 68.1 cm locations was about 586 Jg^{-1}. This failed rod had a burnup of 4.4 $MWdkg^{-1}U^{-1}$, and was not opened for re-pressurisation and instrumentation. It should also be mentioned that this rod was not pre-pressurised with helium, but was air filled at atmospheric pressure.

Significant differences were observed between the two intact rods with high internal pressure and the intact low-pressure rod. The maximum hoop plastic strain for the high-pressure rods was twice that of low-pressure rods; the maximum cladding hoop plastic strain measured in the two high-pressure rods was about 6 % whereas for the low-pressure rod, it was around 3 %. Moreover, more pronounced clad ridging was observed for the high-pressure rods, at intervals equal to the length of fuel pellets.

The nine rods in test *RIA* 1-4 were ramped to $\overline{H}_p$ = 980 – 1160 Jg^{-1} and all of them failed by PCMI-induced cracks.

D.3 NSRR TESTS

D.3.1 Tests on PWR fuel rods

D.3.1.1 MH and GK rods

The MH and GK test series consisted of altogether five *PWR* 14×14-test rods. They were subjected to power pulses, yielding $\overline{H}_p$ from 196 to 389 Jg^{-1}, as shown in Table D-3. All the test rods survived the transients. During the power pulse, prompt cladding axial displacement was observed, indicating the occurrence of PCMI.

The maximum axial displacement was reached 5 to 10 ms after the peak power and then decreased after 5 s to its equilibrium position. Cladding surface temperature was increased to saturation temperature (100°C) and nucleate boiling condition did occur for a few seconds.

Cladding dimensional changes were measured after testing. Plastic hoop strains were found in rods where $\overline{H}_p \geq 230$ Jg^{-1} (55 $calg^{-1}$), with the maximum value of 2.3 % for rod GK-1, which had experienced $\overline{H}_p$ = 389 Jg^{-1}. In Table D-4, we list the recorded strains for these rods.

All the test rods experienced considerable fission product gas release, which contributed to a 4 to 8 MPa pressure increase in the rods. Fission product gas release fraction determined by rod puncturing ranged between 0.04 and 0.13 during the transient, as shown in Table D-4. Destructive examination of fuel pellet material revealed cracks extending through a 1 mm thick rim at the pellet surface.