Performance Evaluation of New Advanced Zr Alloys for PWRs/VVERs

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Ecolabelled printed matter, 441 799

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Contents

1 Introduction

To meet the current situation with more aggressive reactor environments (higher burnups, changing water chemistries and loading patterns), a large number of zirconium alloys have been and are being developed. The main driver for the material development in Pressurized Water Reactors (PWRs) has been to reduce corrosion rates and Hydrogen Pick-Up Fractions (HPUFs), which have occasionally limited the maximum discharged burnup.

However, to ensure that the new Zirconium Alloys performs satisfactory during normal operation, Anticipated Operational Occurrences (AOOs), postulated accidents and intermediate dry storage, it is crucial to assess the projected performance of components of the new zirconium alloy materials and relate the performance to the material characterises. This assessment is the objective of this Special Topic Report (STR).

The structure of the report is as follows:

- Section 2 describes the background history of Zr alloy material development for PWRs and VVERs[1].
- Section 3 describes the relevant fuel design criteria of the Fuel Assembly (FA) and its components relevant to the Zr alloy during normal operation, AOO, accident conditions and intermediate dry storage. A more detailed description of the fuel design criteria is provided in the Appendix.
- Section 4, 5 and, 6 covers the performance of current and new improved Zr alloys for PWR and VVER FA applications during normal operation, AOO, Design Basis Accidents (DBAs) and intermediate dry storage, respectively.
- Section 7 provides recommendations for Zr alloy development to achieve optimum performance in-PWR/VVER during normal operation and AOO as well as during DBAs and intermediate dry storage.

An Appendix discussing the Mechanical Design Bases during normal operation, AOO and DBAs is also provided.

[1] Voda Voda Energo Reactor (Russian type PWR)

2 Background history

For the first PWRs in the US thin wall stainless steel tubes were used as material for Fuel Rod (FR) cladding as well as Guide Tubes (GTs). Since the early 60s stainless steel was replaced by Zry-4 since Zr based alloys absorbs less thermal neutrons compared to stainless steel.

The development of Zr alloys for nuclear reactors was first revealed at the 1st U.N. Conf. on the Peaceful Uses of Atomic Energy in Genera in 1955 [Lustmann & Kerze, 1955a] by the U.S. They reported on alloys with additions of tin, iron, chromium and nickel. At the USAEC symposium on zirconium alloy development in 1962 [Kass, 1962] gave an overview on the further development of these alloys. These alloys were called 'Zircaloys' (Zry). Zry-2[2] became the preferred cladding for Western Boiling Water Reactors (BWRs), while Zry-4[3], with its lower hydrogen uptake rate in hydrogenated water, was the preferred cladding in Western PWRs.

At the 2nd U.N. Conf. on the Peaceful Uses of Atomic Energy in Geneva in 1958 the Union of Soviet Socialist Republics (USSR) revealed extensive corrosion data from their development of Zr-Nb alloys for fuel cladding and structural alloys in their water-cooled reactors [Ambartsumyan et al, 1958] and [Ivanov & Grigorovich, 1958]. From these studies the Zr-1% Nb alloy (E110) and Zr-1%Nb1.1%Sn0.4%Fe alloy (E635) were selected later for their VVER reactors. The development of VVERs started in 1964 (210 MW DEMNO plant), 1971 (VVER-440 MW) and 1997 (VVER 1000 MW). The FA structural components for these reactors were originally made by stainless steels but since 1994 by E110 and E635. The oxide layer build up and Hydrogen Pick-Up (HPU) in the E110 cladding tubes was usually extremely low except a few cases of abnormally enhanced corrosion caused by several coolant impurities [Adamson et al, 2006/2007a]. However, a poor behaviour of E110 in High Temperature (HT) steam during anticipated Loss of Coolant Accidents (LOCAs) and the rather high corrosion attack of E635 made an improvement of the alloys necessary.

The Zrys have provided good service in Western reactors for many years. The dimensional and corrosion behaviour under normal operation, AOOs and hypothetical DBAs were evaluated over many years by several organizations.

Corrosion and dimensional changes under normal corrosion as well as under anticipated accident conditions were found to be satisfactory up to late 70s.

In the 1980s the nuclear industry needed to improve its competiveness compared to other energy sources. To accomplish this, the nuclear industry strived to lower the fuel cycle cost, by increasing the allowable burnup and local power density. However, the development of PWR corrosion models - including the effect of irradiation which becomes important at an oxide layer thickness of >5 µm e.g. [Garzarolli et al, 1985], clearly showed that the desired burnups could not be reached with the old standard Zry-4, at least not in PWRs with modern thermal hydraulic conditions. Therefore, broad development programs to improve the corrosion resistance of Zry-4 cladding were started in the early 80s while development projects to develop new better Zr-alloy cladding were initiated in the mid-80s.

In the first phase, tests were performed on Zry-4 cladding with varying alloying contents, impurity contents, and material conditions. The goal was to optimize the corrosion behaviour of Zry-4. These tests, e.g. [Fuchs et al, 1991], showed that Sn, C and Si influence corrosion of Zry-4 in PWRs. As result, a Zry-4 cladding with a restricted chemistry, the "low-Sn-Zry-4"-cladding, was specified and applied for reloads since 1988. The "low-Sn-Zry-4" allowed an increase in fuel burnup. However, under full-core loading with low leakage and high heat flux for most (>70%) of the operating time, some of the "low-Sn-Zry-4" claddings exhibited an increased corrosion at least in the German PWRs constructed by Siemens. The increased corrosion was caused by dense hydride rims at the fuel clad outer surface. The mechanism of this type of increased corrosion is described in the literature; e.g. [Garzarolli et al, 2001a].

[2] Zry-2: 1.2-1.7%Sn, 0.07-0.20%Fe, 0.05-0.15%Cr, 0.03-0.08%Ni
[3] Zry-4: 1.2-1.7%Sn, 0.18-0.24%Fe, 0.07-0.13%Cr, <0.007%Ni

Further tests by Siemens (now AREVA[4]) revealed that the Transition Metal (TM) alloying content (e.g., Fe, Cr, Ni, V, etc) and the microstructure also have a pronounced effect on corrosion behaviour. [Broy et al, 2000] and [Seibold & Garzarolli, 2002] summarized the results of that time together with later data. By using these results it was possible to develop the "optimized Zry-4", with Fe in the upper range of the American Society for Testing and Materials (ASTM) specification and an optimized microstructure. This type of cladding was used for Siemens reloads after 1989. Figure 2-1 shows the improvement of corrosion behaviour of the different development steps.

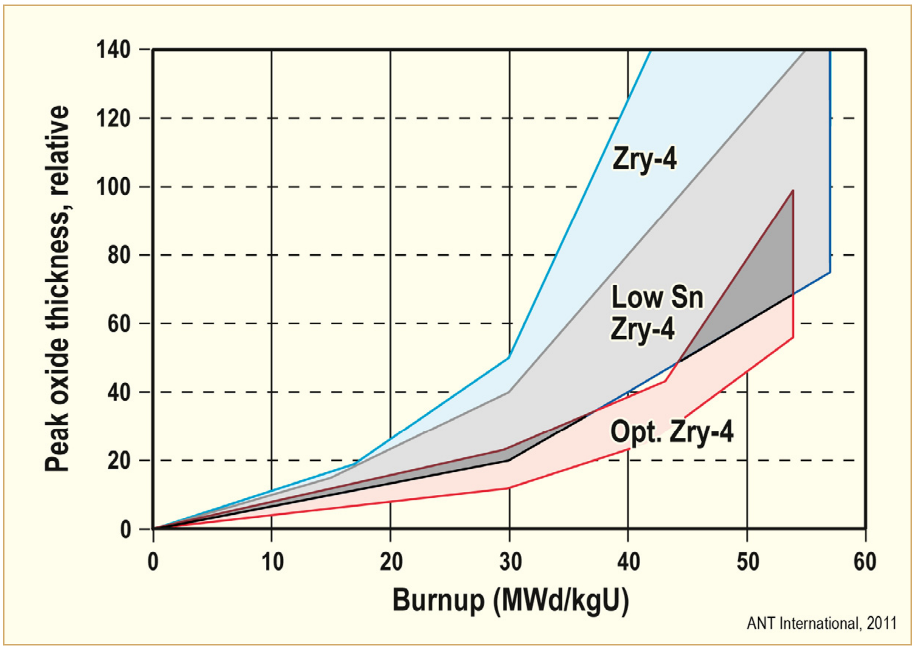

Figure 2-1: Corrosion behaviour of different Zry-4 cladding materials, after [Garzarolli et al, 1996b].

However, it was soon found out that even the best Zry-4 does not permit achievement of the final target burnup in modern PWRs. Therefore, in a second phase several promising alternate Zr alloys were examined with test FRs and corrosion coupons in water rods in PWRs. The following alloying systems were tested in different fuel vendor programs, e.g. [Sabol et al, 1989, 1997], [Fuchs et al, 1991], [Broy et al, 2000], [Seibold & Garzarolli, 2002], [Mardon et al, 2000], [Tsukuda et al, 2003], [Wikmark et al, 2008]:

- ZrSnFe(CrV)
- ZrNbFe(CrV)
- ZrSnNbFe(CrV)
- ZrNb
- ZrFe(CrV)

[4] French Equipment Manufacturer

Most of these tests were performed in the BR3 test-PWR, the Gösgen Nuclear Power Plant (NPP) in Switzerland (which has quite demanding thermal hydraulic conditions) the North Anna PWR in the US and Vandellos-II PWR in Spain. Test FRs with alternate Zr alloys were irradiated for up to 9 years and almost 100 MWd/kgU in these programs. Examinations were performed on the test FRs after subsequent cycles in the reactor pool and in hot cells. These examinations showed that corrosion behaviour depends on the contents of Sn, a TM, and Nb. Sn-free FRs exhibited the lowest corrosion rate. From these programs new Zr alloys were selected for reload applications such as:

- Siemens used DX-ELS cladding (a Zry-4 cladding with an thin outer layer with significantly reduced Sn and increased TM content) since 1988/1989.
- Westinghouse introduced Zirconium Low Oxidation (ZIRLO) (a Zr alloy with 1%Sn, 1% Nb, and 0.1% Fe, based on the Russian alloy E635) for reloads since 1991.
- AREVA applied M5 (a Zr1% Nb alloy with some S, based on the Russian alloy E110) for reloads since1998.
- The Japanese suppliers selected Mitsubishi Developed Alloy (MDA) (0.8% Sn 0.5% Nb 0.2% Fe 0.1% Cr) and New Developed Alloy (NDA) (1% Sn 0.1% Nb 0.3% Fe 0.2% Cr) for reloads since 2004.
- Westinghouse concluded that ZIRLO should further be improved by lowering the Sn content and started to test Low-Tin & Optimized ZIRLO in 1999/2002. Optimized ZIRLO is used it for reloads since 2008.
- Tests on HT corrosion behaviour under anticipated LOCA condition showed a much poorer behaviour for the Russian E110 material than for the M5 of AREVA and other Zr-alloys. It was found out that this poor behaviour was connected with the feed material from which the alloy was fabricated. As a consequence, the Zr source material was changed from electrolytic + crystal bar, which had been used historically, zirconium to sponge zirconium [Nikulin et al, 2007].

Several utilities and organizations concluded that a further cladding material development is necessary for the future demands. These conclusions are as follows:

- The Japanese suppliers reduced the Sn content of their MDA and NDA to Modified-Mitsubishi Developed Alloy (M-MDA) and S2 alloys and started test rod irradiation in 2001. Furthermore, they developed their J-alloys (Zr1.6-2.5Nb) and started irradiation of test rods in 2006.
- Westinghouse developed their AXIOM alloys with 0-0.5 Sn (Zr0.7-1NbSnFeCrCu) and started irradiation of test rods in 2004-2006 (in 4 PWRs).
- AREVA developed their Quaternary alloys (Zr1Nb0-0.5Sn0.1-0.2Fe), and started irradiation 2003 (in 2 PWRs).
- In Korea, the High performance Alloy for Nuclear Application (HANA) alloys were developed, tested in HALDEN and irradiated in a PWR since 2008.
- In Russia, the alloys E110 and E635 are being continuously improved, e.g. [Shishov et al, 2007].

In the following, the corrosion and dimensional behaviour and design criteria will be reviewed for normal and anticipated accident conditions.

3 Fuel design criteria related to performance of Zr alloys

A comprehensive treatment of fuel design criteria on fuel performance during normal operation, AOOs, DBAs and during dry storage is provided in the Appendix.

The thermo-mechanical properties of the Light Water Reactor (LWR) FA are crucial for its satisfactory performance in-reactor. The Standard Review Plan (SRP) discussed in Section 4.2, lists different thermo-mechanical failure modes of the FA and its components that either had occurred up to its last revision or were thought to be potential failure modes. The SRP, Section 4.2, also lists the design criterion to each failure mode to ensure that the FA behaviour is satisfactory. These design criteria are set to ensure that:

- The FA will not *fail* during normal operation (Class I conditions) and AOOs (Class II events). *Failing* in this sense has a broader meaning, namely that the FR may not be breached and that the dimensional changes of the assembly during irradiation must be limited. The latter requirement is to ensure that control rods can be inserted and that the fuel can be handled during shutdown. Also the grid or outer channel cross section must not have increased to such an extent that it is impossible to pass it through the upper core grid during reloading.

- The fuel remains coolable during a DBA accident (Class III and IV events). The Class IV DBAs are LOCA, Reactivity Initiated Accident (RIA) and earthquakes. During LOCA and RIA, fuel must not be dispersed outside the FR cladding which means that the fuel cladding may not fail in a brittle fashion during the *reflooding*[5] phase during LOCA and due to Pellet Cladding Mechanical Interaction (PCMI)[6] during the RIA transient. During Class III and IV situations, limited fuel failures are however accepted. Another criterion that must be fulfilled in these situations is that it should be possible to insert the control rods to be able to shut down the reactor.

Additional regulations are related to interim dry storage to ensure that, fuel is not dispersed outside the intact FR cladding during normal handling, transport and dry storage and during a DBA.

- During normal dry storage, the regulating criterion is to limit the amount of clad creep strain (either directly or indirectly by limiting the clad temperature and stress) to eliminate fuel clad creep rupture, which is the most likely fuel cladding failure mechanism.

- The DBA to be considered during handling and transport varies considerably from a 9 meter drop of a loaded cask onto an “unyielding” concrete surface to airplane crashes and earthquakes during normal storage. In this case, the regulating criteria are set to minimize the amount of fuel radial hydrides formation and hydride reorientation (by limiting the clad temperature, stress and the number of drying cycles). Radial hydrides are the major actor to reduce the fuel clad ductility resulting in fuel clad failures during a DBA.

Table 3-1 lists the primary failure causes related to the Zr alloy properties that have occurred in commercial reactor during Class I and II operation. Table 3-2 provides corresponding information for hypothetical failures.

In this PWR/VVER Zr alloy performance evaluation report, the focus has been to assess the Zr alloy performance relative to the failure mechanisms listed in Table 3-1 and Table 3-2.

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[5] This is the last phase during a LOCA situation when the core is reflooded with water that cools the fuel cladding surface imposing very large thermal stresses that may fracture the fuel cladding.
[6] PCMI, i.e., interaction without the influence of fission products such as iodine (that would instead result in PCI.

Table 3-1: Primary failure causes for PWR/VVER fuel that have occurred during normal operation and AOOs.

Primary failure cause	Short Description
Excessive corrosion and hydriding	An accelerated corrosion process results in cladding perforation. This corrosion acceleration can occur in a PWR/VVER due to: • FR Chalk River Unidentified Deposits (CRUD) deposition on the FRs, where the CRUD layer can – Form a thermal barrier, overheating the fuel cladding or – Together with subcooled boiling concentrate water soluble species in the CRUD/oxide layer to such an extent that the protectiveness of the zirconia barrier layer at the Zr alloy metal/zirconium oxide interface is deteriorated • Overheating of the FR cladding due to Departure from Nucleate Boiling (DNB) as a result of – Excessive FR/FA bowing. – Excessive FR reactivity due to a post Axial Offset Anomaly, AOA event (e.g. due to power change during AOA which may result in that the concentrated boron in CRUD/oxide will go in solution due to the power change resulting in a overheating of the fuel cladding) • PWRs only – Dissolution of Second Phase Particles (SPPs) in Zrys (Zr-Sn-Fe-Cr alloys) at high burnups – The late in –life corrosion acceleration occurs earlier with increasing Sn content in Zrys • PWRs only - Formation of a zirconium hydride layer at the Zr alloy metal/zirconium oxide interface which requires a fuel clad minimum hydrogen content of 300 wtppm and a heat flux larger than 50 W/cm^2
PCI/PCMI	Pellet Cladding Interaction (PCI) failures may occur in fuel claddings during a power ramp provided that the following three conditions are met: • Critical clad stresses - The initial pellet-cladding gap is small enough before the ramp (depends on burnup and power history) and the ramp is large enough to generate sufficient cladding stresses to initiate a Stress Corrosion Crack (SCC) at the fuel clad inner surface that may propagate through the whole cladding thickness. – It appears that a Zr or Zr-Fe liner/barrier at the cladding inner surface may reduce the stresses thus making it harder for SCC crack initiation – Chromate additions to the fuel pellets may make the fuel pellet softer, thus reducing the clad stresses – Pellet geometry also impacts the clad stresses during a power ramp • Critical concentrations of fission products (Cs/Cd and/or fresh Iodine) must exist to enable SCC crack initiation which means that a critical burnup must be reached. – Alumina additions to the fuel pellets are concentrated in the fuel pellet grain boundaries and acts as Cs/Cd and/or fresh Iodine traps thus lowering the concentrations or chemical form of these fission products at the clad inner surface, thus reducing the PCI failure tendency • Sensitized material – A condition which is met for all types of Zr alloys as well as Zr or Zr-Fe liner/barriers
Grid to rod fretting	Grid-rod fretting - Excessive vibrations (due to grid inadequate support of the FRs) may result in FR fuel failures. The grid design (including the spring design/material is crucial) is crucial for adequate FR support. To get adequate rod support the required spring properties are: • Good spring properties, i.e., high creep strength to ensure elasticity of the grid spring throughout the life of the FA. Since Zr alloys have poor elasticity properties, nickel-base alloys (characterised by excellent spring properties) are mostly used as the grid spring material. • In the case that the grid spring material is made of a Zr alloy, it is important that excessive hydriding of the Zr alloy does not occur, since mechanical failure of the hydrided grid spring material could lead to grid-rod fretting failures in PWRs.
Yield tensile strength, ductility and fracture toughness	Yield tensile strength –The Zr alloy yield strength must be large enough to ensure that yielding does not occur. The most limiting condition is unirradiated Zr alloys (i.e. fresh fuel) since irradiation damage increases the yield strength with a factor of 2-3 during the first couple of months irradiation. The yield and Ultimate Tensile Strength (UTS) are not impacted significantly by Zirconium hydrides. Ductility and fracture toughness must be adequate for the Zr alloy components to ensure that failure does not occur either during reactor operation or during outage handling. • The irradiation damage that increases the yield strength simultaneously decreases the ductility and fracture toughness. • Hydrides can reduce ductility and fracture toughness significantly. This effect is very much temperature dependant such that, for a given concentration, the embrittlement effect is small at operating temperature and is much larger at lower temperature; e.g., during an outage or post-irradiation storage.
Excessive dimensional changes	Three processes are involved in dimensional changes of reactor components: 1) hydriding, 2) irradiation growth and, 3) irradiation creep Excessive FA bowing may result in: • lower thermal margins (DNB and LOCA) due larger than anticipated water gaps between neighbouring FAs • Insertion Rod cluster Incident (IRI) – difficulties to insert the control rod clusters • Handling difficulties during outage which may: – Extend the time to loading/unloading of the core – Cause grid damage
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Table 3-2: Hypothetical PWR/VVER failure causes in the SRP Section 4.2 (that have not occurred in LWRs).

Primary failure cause	Short description
Fuel clad creep burst due to rod overpressure	If the rod internal pressure becomes larger than the reactor system pressure, the fuel cladding may start to creep outwards. If the fuel cladding outward creep rate becomes larger than the fuel swelling rate (due to fission product production during irradiation), the pellet-cladding gap may increase. This phenomenon is named *liftoff*. Since this gap constitute a significant barrier towards the heat flux, an increased gap may potentially result in an increase in fuel pellet temperature. This higher temperature will in turn increase the fission product release rate thus increasing the FR overpressure even more leading to an even higher outward cladding creep rate. Thus, a thermal feedback effect could quickly lead to fuel failure. The margins to *liftoff* depend to a large extent of the fuel clad creep properties and these properties are actually limiting in many cases the fuel burnup. A larger fuel clad creep strength will increase the *liftoff* margins. Tests conducted in Studsvik and the Halden research reactor have, however, shown that recorded temperature increases in the fuel are not due to “classical” *liftoff* (reopening of the pellet-cladding gap) but due to decreased thermal conductivity as a result of pellet crack formation within the fuel pellet. Thus, the hypothetical thermal feedback effect that was thought off when this criterion was established will most likely not occur. However for other reasons the rod internal overpressure must be limited. If this overpressure becomes large enough (clad tensile stress > 80 MPa), reorientation of hydrides in the fuel cladding may occur that, in turn, may deteriorate the PCMI performance during a RIA. Also, the rod overpressure must be limited for LOCA and intermediate dry storage concerns.
Fatigue	Fatigue stresses may be induced in the FA components due to, e.g., the turbulent coolant flow. According to the SRP, section 4.2, the cumulative number of strain fatigue cycles on the structural components should be significantly less than the design fatigue lifetime, which is based upon the data by [O’Donnell & Langer, 1964], and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. In design calculations the fuel vendor must show that alternating bending stresses due to dynamic loads must be below 50 MPa. This limit is based upon [O’Donnell & Langer, 1964]. However in fuel design calculations it is always shown that the maximum fatigue stresses are very low compared to the design limits and therefore making this failure mechanism very unlikely to occur.
Outside-in-cracking	Outside-in-cracking (PCMI) failures may occur during a power ramp in fuel claddings with a thick enough outer hydride rim provided that the clad tensile stresses (due to the ramp) are large enough to drive a crack from the outer clad surface inwards through the whole clad thickness.
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4 Zr Alloy performance during normal operation and AOO

4.1 In-PWR/VVER Corrosion and HPU

4.1.1 Introduction

Corrosion of zirconium alloys is a thermodynamic and electrochemically based process affected by the following parameters, see Figure 4-1:

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

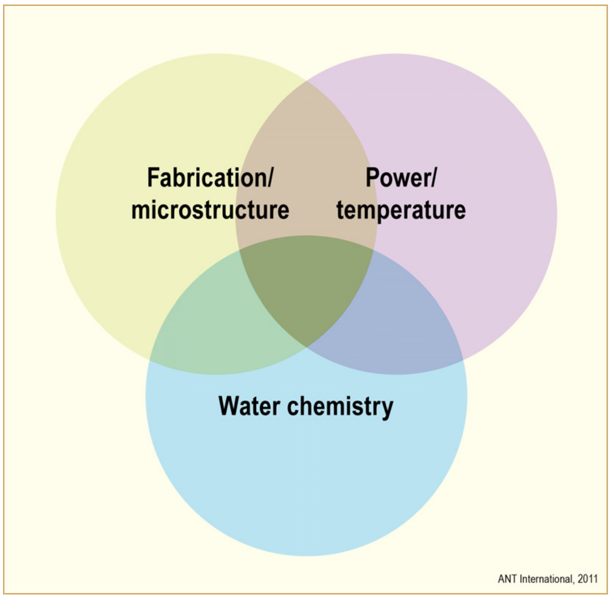

Figure 4-1: Parameters impacting corrosion performance of Zr Alloys.

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

During the initial oxidation/corrosion of zirconium alloys, a thin protective black oxide is formed. As the zirconium oxide grows in thickness the outer part of the oxide (phasing the water/steam phase) is transformed into a greyish porous oxide. The oxide grows inwards into the zirconium alloy material.

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

The corrosion and hydriding process of Zirconium alloys is schematically shown in Figure 4-2.

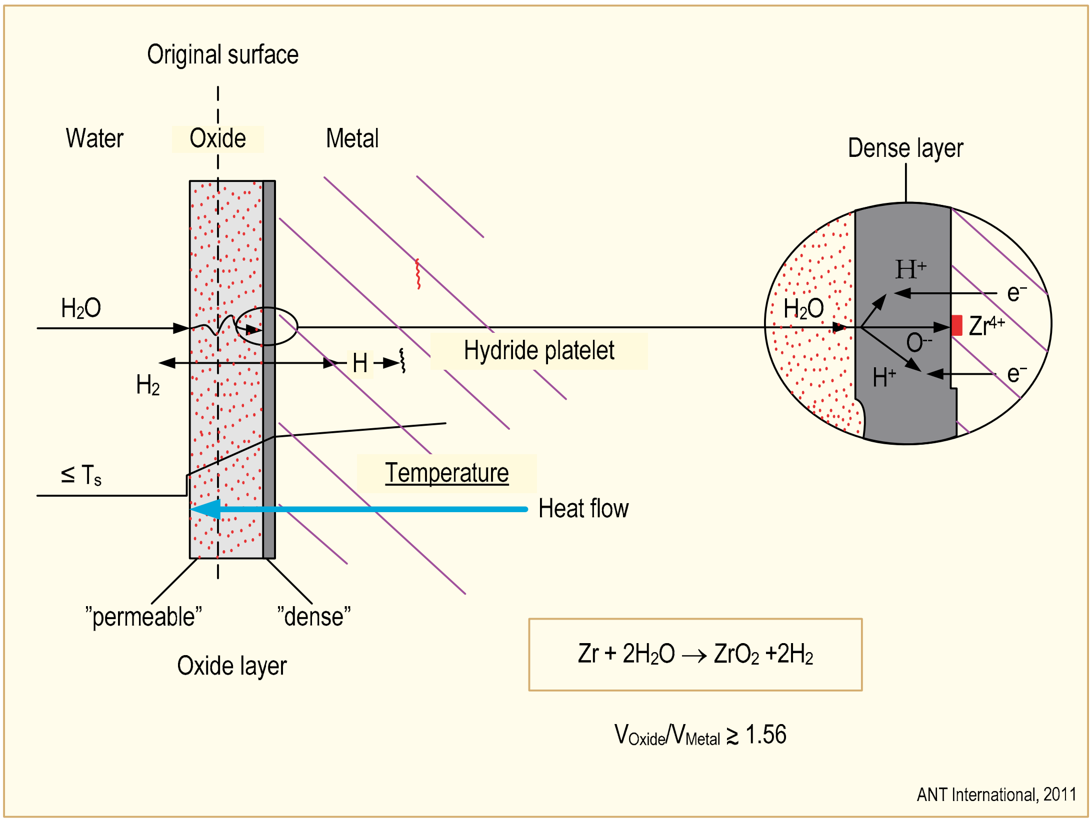

Figure 4-2: Schematic showing the corrosion and HPU process in zirconium alloys.

The corrosion rate increases with increasing temperature at the zirconium metal/zirconium oxide interface according to an Arhenius dependency with an activation temperature of 16000 to 25000 K, i.e.,

$$\frac{ds}{dt} = C \exp\left(-\frac{Q}{RT}\right)$$

The factors affecting the temperature are (Figure 4-3):

- Rod power.
- Oxide thickness and its thermal conductivity.
- CRUD thickness and its thermal conductivity.
- Heat transfer coefficients.
- Bulk coolant temperature.

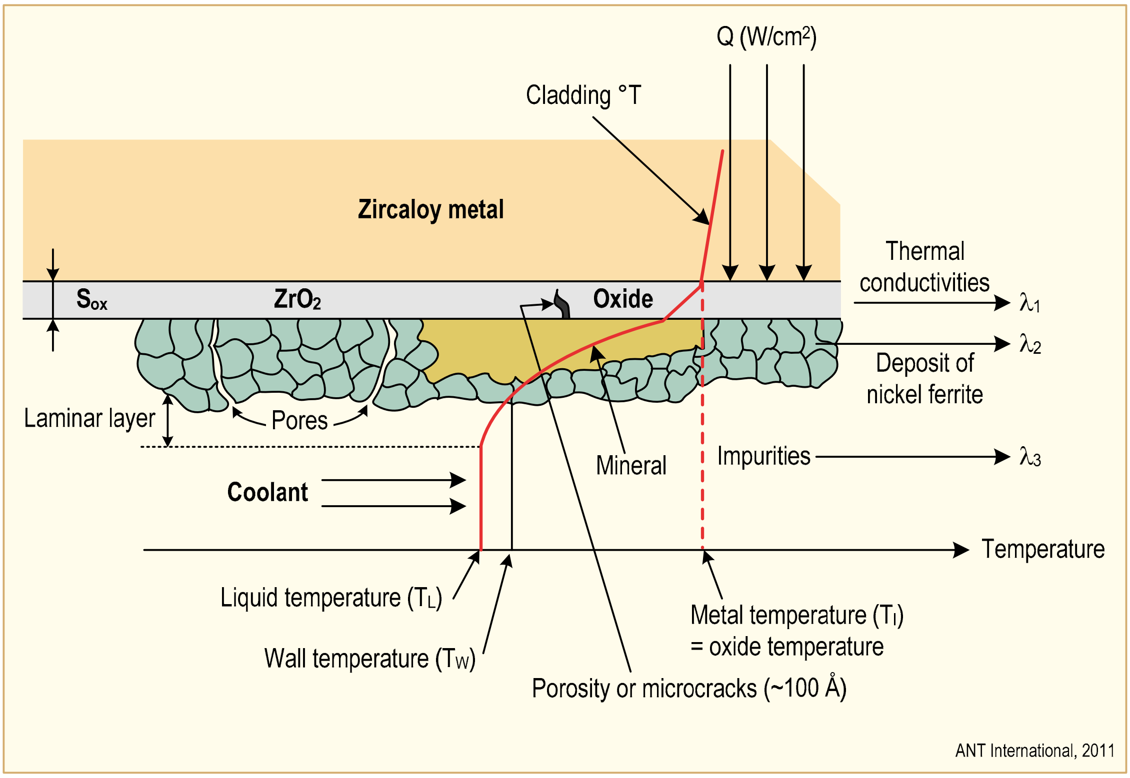

Figure 4-3: Factors that affect cladding temperature, after [IAEA[7], 1998a].

The corrosion rate is usually increased by neutron irradiation depending on oxygen/hydrogen of the coolant and particular material parameters. Different impurities or additives in the water can affect the corrosion attack, such as oxygen, fluorine and LiOH.

The interested reader is referred to: ZIRAT6[8]/IZNA1[9] STR Water Chemistry and CRUD Influence on Cladding Corrosion, ZIRAT7/IZNA2 STR Corrosion of Zirconium Alloys, ZIRAT8/IZNA3 STR The Effects of Zn Injection (PWRs and BWRs) and Noble Metal Chemistry (BWRs) on Fuel Performance, ZIRAT9/IZNA4 STR Corrosion of Zr-Nb Alloys in PWRs, ZIRAT12/IZNA7 STR Corrosion Mechanisms in Zirconium Alloys.

4.1.2 Standard alloys

4.1.2.1 Programs

After it became clear that the corrosion resistance of even the best Zry-4 does not allow the desired burnup demands, the fuel suppliers started to develop alternate cladding materials with significantly improved corrosion resistance. The different new Zr alloy materials for various FA components that have been applied for reloads are listed in Table 4-1. The table also provides the material chemical composition and final heat treatment condition as well as an estimated value of the relative in-PWR corrosion and creep behaviour rates.

[7] International Atomic Energy Agency
[8] Zirconium Alloy Technology
[9] Information on Zirconium Alloys

5 Zr Alloy performance during DBAs, LOCA and RIA

This topic is covered more extensively in the following ANT International Reports:

- ZIRAT9/IZNA4 STR Loss of Coolant Accidents, LOCA, and Reactivity Initiated Accidents, RIA, in BWRs and PWRs and,
- ZIRAT15/IZNA10 STR Processes going on in Nonfailed Rod during Accident Conditions (LOCA and RIA).

5.1 Introduction

5.1.1 Reactor safety

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. The main objective of 10CFR[37]50 and 10CFR100 is to limit radioactive impact on the environment:

- In 10CFR50, the General Design Criterias (GDCs) are specified and interpreted in the Standard Review Plan, 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.

  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 – Large Break Loss of Coolant Accident (LBLOCA) and the calculated dose to the environment must be below the 10CFR100 dose limit, [Shoop, 2004]. The dose calculations for all the other DBAs 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 PWRs) and CPR (in BWRs) have failed. For the dose calculations, only the source term generated by these assumed failed rods need to be taken into account. Historically, the Nuclear Regulatory Commission (NRC) has defined that during a RIA, the dose must be ≤ 25% of the 10CFR100 dose limits. For other DBAs (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 DBAs is that in the case of a LOCA, for example, there is some delay in the radioactivity leaking out to the environment while for other accidents, such as steam generator tube rupture (with lower maximum allowable doses), 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 eliminates 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, code calculations must show, that less than 10% of the fuel has failed during a LOCA. 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.

[37] Code of Federal Regulations

5.1.2 LOCA

The LOCA event starts by the decrease and then the loss of coolant flow due to a break in a coolant pipe while, at the same time, the reactor is depressurized, scrammed and shut-down. The fuel starts heating up due to its decay heat until the Emergency Core Cooling Systems (ECCSs) are activated and cools the fuel. Hypothetical LOCA events are analyzed for each reactor to assure that the safety criteria, defined by the regulators, for the reactor system and the fuel, are met. The DBAs relevant to LOCA fall into two general categories. The LBLOCA, assumes a double ended break of a primary coolant cold leg of a PWR or a break in the recirculation pump intake line of a BWR either of which could cause the loss of all the coolant from the core. The small break, or Small Break LOCA (SBLOCA), assumes a break in one of the smaller primary circuit lines that will cause less coolant loss than the LBLOCA.

The effect of a LOCA cycle on the fuel is shown schematically in Figure 5-1, which plots the fuel and cladding temperatures as a function of time in the accident. The loss of coolant flow and reactor pressure at the initiation of the accident will decrease heat transfer and allow the fuel and cladding to heat up until the reactor scrams. The fuel will then cool down somewhat partly due to cooling by the steam-water mixture that is formed, but the cladding temperature will continue to rise until the reflood and quench stage.

During and after the LOCA it must be ensured that:

- The core remains coolable (which means that the maximum allowable coolant blockage is limited) and,
- No fuel dispersal occurs (which means that cladding rupture is not allowed; it is assumed that the cladding burst is so small that only fission gases are released).

Table 5-1 Summarises the impact of Zr alloy properties on margins towards the LOCA design criteria.

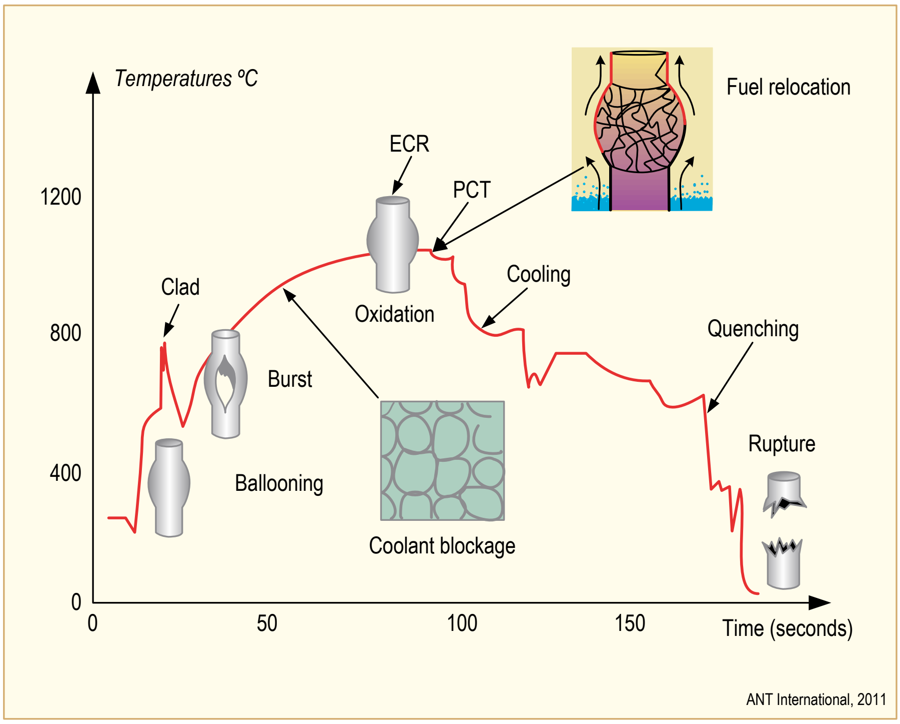

Figure 5-1: Typical LOCA in a PWR.

Table 5-1: LOCA Design Criteria and relation to Zr alloy issues and performance limitations.

Criterion	Failure mechanism	Consequences	Material parameters increasing margins to failure	Read more
LOCA – maximum 10% of the fuel rods in core may burst (Germany)	FR burst		Reduce the fuel clad hydrogen content	e.g. FDRH[38], 2010 and ZIRAT15/IZNA10 STR Vol. II, 2010
LOCA – retaining fuel clad ductility	Fuel cladding fracture (and fuel dispersal)	Non-coolable fuel geometry	Increase fuel clad ductility by: 1) reducing hydrogen clad pickup from pre-LOCA irradiation as well as during LOCA HT oxidation, 2) reduce the HT LOCA clad oxidation	e.g. FDRH, 2010 and ZIRAT15/IZNA10 STR Vol. II, 2010
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In the following subsections, the effects of Zr alloy properties on LOCA fuel performance are described.

5.1.2.1 Ballooning

The loss of coolant flow decreases heat transfer from the fuel, increases the fuel temperature and causes a significant temperature rise of the cladding. The decrease in system pressure causes a pressure drop across and a hoop stress in the cladding. The result is the creep deformation, or *ballooning* of the cladding. Depending on the temperature, the cladding ductility and the rod internal pressure, the cladding will either stay intact or *burst*. Ballooning of the FRs may result in *blockage* of the coolant sub-channels that in turn may impact the fuel coolability. If large fuel clad burst strains occur at the same axial elevation leading to *co-planar deformation* in or among FAs, coolability may be significantly degraded. The extent of the ballooning is dependent on:

- Creep strength of the cladding which depends on:
  - Alloy composition
  - *Fuel clad hydrogen content (picked up during the water-zirconium alloy corrosion reaction during reactor operation prior to LOCA), see Figure 5-2 and Figure 5-3*
  - Microstructure
  - Texture
  - Oxidation during the LOCA

  The elevated LOCA temperatures will anneal radiation damage and any CW remaining from the fabrication process
- Stress and corresponding strain rate the cladding is subjected to,
- Temperature and the rate of temperature rise assumed for the DBA.

Figure 5-2 shows that hydrogen decreases the $\alpha/\alpha+\beta$ phase transformation temperature, which means that increasing the hydrogen content in the fuel cladding will lower its ductility and result in more FR ruptures during a LOCA. Figure 5-3 shows that fuel clad ductility in thermal creep tests is lower for prehydrided[39] Zry-4 than that of Zry-4 with low hydrogen content.

[38] Fuel Design Review Handbook
[39] Prehydriding is being done to simulate the HPU during the base irradiation prior to the hypothetical LOCA event.

6 Intermediate dry storage

6.1 Introduction

As of mid-2010, about 225 000 tons of Spent Nuclear Fuel (SNF) is stored around world [Sokolov, 2010]. This fuel is being stored in wet and dry conditions in reactor pools and interim storage facilities without major incidents. Storage is, however, only a temporary step in the overall fuel cycle; the fundamental issue of whether the SNF is a resource to be recycled or waste to be disposed of after ageing remains unresolved.

Fuel cladding is considered the primary confinement barrier and needs to maintain a high degree of integrity during normal storage and handling. Similarly, the FA is a structural entity that ensures the configuration of SNF in a storage container and enables handling and retrieval. FAs need to survive storage and handling without substantial alteration under normal conditions. Under accident conditions during storage or subsequent transportation, the fuel must remain subcritical and should be recoverable by normal methods.

The two key fuel clad failure modes during interim dry storage is:

1) *Creep rupture during storage and,*

2) *Brittle fracture of hydride fuel cladding during a cask drop accident*

Controversy exists regarding the mechanism and the conditions under which hydrogen and hydrides can contribute to cracking of cladding during dry storage (Hydrogen assisted cracking or DHC of fuel cladding). Kim has proposed a cracking mechanism, which is predicted to lead to failure when temperatures fall below 180°C [Kim, 2008]. This mechanism is based on work by Kim and co-workers at the Korean Atomic Energy Research Institute (KAERI) over the past ~10 years. Opposing arguments have been offered by Puls based on work which originated at Atomic Energy of Canada, Ltd. (AECL) in the mid-1970s and has been advanced by research at AECL and other laboratories since that time [Puls 2009] and [Strasser et al, 2008/2009a and b]. At present, there is no definitive resolution of the issue raised by potential FR failure during dry storage as postulated by Kim. Simple calculations using the data presented by Kim argue against such failures, however. That is, spent PWR was examined after 15 years of dry storage and found to be free of breaches or non-penetrating cladding cracks [Einziger et al, 2002]. The absence of failures in either the dry storage characterization project or in normal wet storage suggests that the theory proposed by [Kim, 2008] is either not valid or not applicable to commercial LWR fuel. See Section 9 in ZIRAT14 AR by C. Patterson for more details.

The extended storage time that is now being projected will result in smaller margins towards creep rupture and lower fuel clad temperature when the cask is transported to the final repository or reprocessing facility. The lower fuel clad temperature will reduce fuel clad ductility/fracture toughness, thus reducing the margins towards fuel clad fracture during a cask drop accident.

The Zr alloy properties having an impact on margins towards clad creep rupture and fracture during accident conditions are summarised in Table 6-1 and reviewed in the sections that follow.

Table 6-1: Interim dry storage design criteria and relation to Zr alloy issues and performance limitations.

Criterion	Failure Mechanism	Consequences	Material parameters increasing margins to failure	Read More
Creep Strain limitation	Creep rupture	Fuel dispersal	Increase fuel clad creep strength	e.g. ZIRAT5-15/ IZNA1-10 AR
No fuel failures during cask drop accident	Fuel clad fracture	Fuel dispersal	Decrease hydrogen fuel clad content, tangential hydrides	e.g. ZIRAT5-15/ IZNA1-10 AR
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6.2 Creep rupture

Creep-rupture is a potential cladding failure mode during dry storage. The parameters that determine the possibility of creep rupture are:

- the cladding temperature during cask drying, depending on the drying procedure,
- the evolution of cladding temperature during long time storage, depending on burnup and the previous cooling time,
- the cladding stresses due to the internal gas pressures depending on cladding wall, fuel duty and burnup,
- the creep strength of the cladding material, which depends on its condition and composition.

During vacuum drying after loading of the fuel into the cask, temperatures of 360-500°C may exist for hours depending on the drying procedure. During long time storage the temperature will decrease from an initial value of 300-400°C to significantly lower values. The cladding hoop stress will also decease as a consequence of the decreasing temperature of the internal gas during this period. The initial value, which varies among FRs, reactor types and regulatory arenas, may be in a range of 60-120 MPa.

The factors which may affect the creep mechanism under dry storage condition are:

- Thermal creep.
- Effect of hydrogen on thermal creep.
- Effect of irradiation hardening on thermal creep.
- Irradiation damage recovery during the initial drying cycle(s) and during subsequent long-term dry storage.

Estimation of cladding creep strain under long-term dry storage condition for tens of years is in most cases performed by using short-term creep data.

The creep behaviour of unirradiated Zry cladding tubes (out-reactor creep) was investigated in much detail in numerous programs. Based on the different creep deformation mechanisms and experimental data [Chin et al, 1986] has developed a creep deformation map for Zry-4. According to this map, different deformation kinetic regions, namely grain boundary sliding (at stresses of about <120 MPa) and dislocation climb creep (at higher stresses), are dominant. The creep stress exponent n and effect of fast neutron fluence on creep rate is much larger in the dislocation climb range than in the grain boundary sliding range. Furthermore, at <120 MPa, grain size might in addition play an important role for grain boundary sliding creep. For dry storage, grain boundary sliding is the dominant process. Thus, any extrapolation from higher stresses to lower stresses may be misleading.

The total creep strain at stresses below yield stress is usually described by the sum of primary creep strain and the product of time and steady state creep rate. If no information exists on the effect of irradiation hardening on creep rate, the potential effect of recovery, and the effect of hydrogen content on creep strain is usually performed by application of thermal creep data of unirradiated coupons. For the alternate alloys, considered in this report, only very limited information on long time creep behaviour at relevant stresses have been published.

The out reactor creep behaviour during dry storage depends similarly as in-reactor creep on:

- The degree of recrystallization after the final annealing
- The final CW before the last heat treatment
- The intermediate annealing temperature
- The SNO (=Sn+2*sol.Nb+6*O) value

A reasonable correlation for the effect of these parameters exists only for in reactor creep [Adamson et al, 2009a]. For out reactor creep in stress range below 120 MPa and temperatures below 400°C the relative difference in dry storage creep rate of different new Zr alloys may at a first approximation be deduced from Figure 6-1.

The SNO value, material condition and the relative creep rate (at a constant stress and temperature) of the **Commercially Used Alternate Zr Alloys** and the **New Alternate Zr Alloys** are given in Table 6-2 and Table 6-3. The relative creep rate in comparison to SR Zry-4 was estimated for the alloys of interest by the assumptions that the ranking of thermal creep is similar to the ranking of irradiation creep (see Table 4-1 and Table 4-3). From the estimated relative creep behaviour, given in Table 4-1 and, Table 4-3, it can be concluded that all **Commercially Used Alternate Zr Alloys** will have less strain during dry storage than standard Zry-4 claddings. Only the NDA cladding will probably experience somewhat higher strain. (However, this conclusion is uncertain because the actual intermediate and final annealing temperatures as well as the final CW are not known.) The creep resistance benefit is especially large for RX Zry-2, SR ZIRLO, and RX M5. As far as the **New Alternate Zr Alloys** are considered, the creep strain of all alloys under dry storage conditions is expected to be much less than SR Zry-4 due to the fully RX or at least to a large extends PR final condition of the New Zr Alternate Alloys.

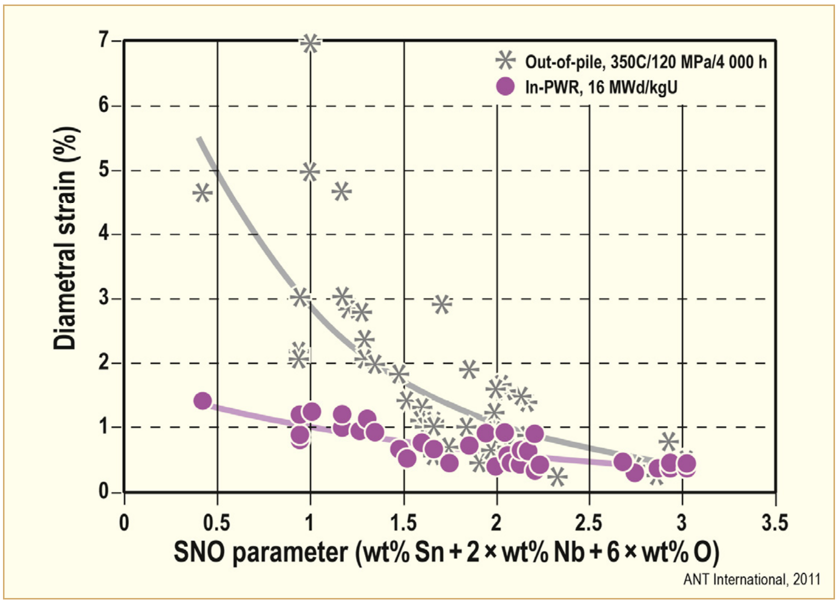

Figure 6-1: Influence of the Sn-, Nb- and O-content on out-of-pile and in-PWR creep, after [Seibold & Garzarolli, 2002].

7 Conclusions for the optimum PWR/VVER cladding and structure material

New Zr alloy materials with increased margins towards the fuel design criteria during normal operation, AOO, DBA as well as dry storage condition are needed. The Zr alloy properties with the smallest margins towards the criteria will become limiting. Thus, it is crucial to develop new Zr alloy materials with properties that increase these margins (Table 7-1).

Table 7-1: Zr alloy properties limiting performance during normal operation, AOO, DBA and intermediate dry storage.

Condition	Failure mechanism	Key Zr alloy material property	Design limit	How to increase margin to failure
Normal operation and AOO	Corrosion	Corrosion rate	Maximum oxide thickness often limited to 100 µm, to ensure that corrosion acceleration due to thermal feedback does not occur	Improve corrosion resistance by alloying and thermomechanical heat treatment during fabrication
	PCI during power ramping	SCC resistance, stress level	Margins assessed by test reactor ramping	Considering the fuel cladding, the best way to improve PCI performance is to add a Zr liner at the clad ID, as being used for BWR fuel cladding. PWR fuel cladding with Zr-liner has been ramp tested with excellent results.
	PCMI during power ramps NB: This failure mechanisms has not yet happened in commercial reactors but was observed for BWR fuel ramped in a test reactor.	HPU =Corrosion rate*HPUF	This failure mechanism may potentially happen for PWR fuel if there exist a solid hydride rim at the fuel clad OD and the ramp height is large enough following extended operation at low power	A prerequisite to form a hydride rim, a large enough heat flux and hydrogen content in the cladding are necessary. Reducing the hydrogen content to <300 ppm may eliminate the formation of a hydride rim at the fuel clad outer surface. An even larger embrittlement effect may be obtained if a hydride blister with significant thickness is formed at the clad outer surface. IN addition to a heat flux and large enough hydrogen content also spalled oxide is necessary for a hydride blister to form. Thus, developing cladding material that forms thin oxides that will not spall will eliminate this embrittlement effect.
	Liftoff	Creep resistance	Rod internal pressure must be limited to avert liftoff that potentially could lead to accelerated rod pressure buildup leading to fuel failure	Improved irradiation creep strength e.g. by replacing SR with RX (or pRX) claddings or by alloying additions
	FA bowing	Elongation rate of GTs	FA bowing (distortion) must be limited to ensure that control rods can be inserted anytime and that thermal margins (DNBR and LOCA) are maintained. Bowing may also cause grid damage during handling.	Tendency for FA bowing is mostly a design issue, i.e., with an appropriate FA design even a large GT elongation rate would not result in any significant FA bowing. However, lower GT elongation rate reduces the GT compressive stresses (as well as other fuel design features may do) and may reduce FA bowing. If GT have a week point or low creep strength, FA bow can occur under the compressive stresses (depending on hold down springs) and cross flow etc.
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Table 7-1: Zr alloy properties limiting performance during normal operation, AOO, DBA and intermediate dry storage. (Cont'd)

Condition	Failure mechanism	Key Zr alloy material property	Design limit	How to increase margin to failure
LOCA	FR fracture and fuel dispersal	The embrittlement of the fuel cladding depends on 1) the HT oxidation rate during the LOCA and 2) the HPU during reactor operation prior to the LOCA and during the HT oxidation phase during LOCA.	Retain some ductility of the fuel cladding to ensure that fuel dispersal does not occur by making sure that the cladding does not fracture during the LOCA quenching phase or during post-LOCA events. The licensing criteria are currently being revised by USNRC	Reduced HPU during reactor operation before LOCA (through more corrosion resistant materials with less HPUF) and during LOCA HT oxidation (through using Zr sponge as source material) will increase margins toward fuel dispersal (by ensuring some retained clad ductility).
RIA	FR fracture (through PCMI) and fuel dispersal	The PCMI failure tendency increases with: 1) increased HPU during corrosion before the RIA 2) localised hydrides (blisters and rims) 3) increased volume fraction of radial hydrides	In most countries, FR failures are allowed above a certain enthalpy increase level during the RIA. However, in Germany FR failures during RIA are not allowed.	Reduced: 1) HPU (by more Zr alloys with lower corrosion rate and/or lower HPUF) and, 2) fraction of radial hydrides (by using SR instead of pRX or RX materials), formed during reactor operation before RIA (through more corrosion resistant materials with less HPUF) as well as elimination of any localised hydrides such as blister and rims will increase margins toward PCMI failures.
Dry Storage	Creep rupture and fuel dispersal through fuel fracture during cask drop accident	Creep strength Smaller tendency to form large fractions of radial hydrides	In most countries a maximum creep strain of 1% is established to ensure that creep rupture does not occur. The clad maximum temperature and stress are limited to minimise the formation of radial hydrides during cooling. The regulation is established to ensure that fuel dispersal during dry storage and cask drop accident does not occur.	Increased creep strength will increase margins towards creep rupture Reduced fraction of radial hydrides (by using SR instead of pRX or RX materials), as well as elimination of any localised hydrides such as blister and rims formed during reactor operation before the hypothetical cask drop accident will increase margins toward FR fracture. It is noteworthy that adding a liner to the PWR cladding less radial hydrides will form since the soluble hydrogen tends to diffuse to the liner material during cooling instead of precipitating on radial grain boundaries.

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In the following, the above limiting Zr alloy properties during different conditions are discussed.

Future Zr alloy materials should have excellent corrosion resistance and a high creep strength. More specifically, an in-PWR corrosion enhancement factor of <1 and the HPUF less than 15% are needed. The creep resistance should be also as large as possible considering the accelerating fission gas release at increasing burnups and fuel duty. This is specifically true in case of the Westinghouse IFBA which release He that contributes to the pressure due to the fill gas and released fission gases. In many countries FR burnup is limited by the requirement that liftoff should not occur, i.e., the creep out rate of the cladding due to internal overpressure should not be larger than the fuel swelling rate. However, often the optimum corrosion resistance and optimum creep strength are counteracting, as far as chemical composition and material heat treatment condition of alternative Zr alloys are concerned. This means that material changes that increases corrosion resistance, decreases creep strength. Thus, the optimum cladding would be a DX cladding with a thin outer layer that provides good corrosion resistance and a thick inner part that determines the creep behaviour. To reduce the tendency to form radial hydrides during dry storage and thereby increase the cladding ductility during a cask drop accident, a Zr liner should be considered.

In the following, the aspects that is important for:

- corrosion and HPU
- creep and
- FR elongation, GT elongation

will be discussed.

The most important material parameters for in PWR corrosion are:

- Sn increases corrosion rate at concentrations > 0.3% (Figure 7-1), but decreases the sensitivity of coolant impurities (such as O, C, H_2S, etc) on corrosion rate. The material condition (intermediate annealing temperature, final cold-work, degree of recrystallization, which influence the creep strength) also affects in-PWR corrosion rate at Sn contents >0.3% (in an opposite direction than in-reactor creep) as can be seen from Figure 7-2 and the information reported by [Seibold & Garzarolli, 2002].

- Nb improves corrosion resistance at concentrations >0.5%. The effect of material condition becomes significant at Nb contents >1%, probably mostly via the distribution and composition of the Nb containing precipitates β-Zr so that late heat treatments above the (α+β) temperature should be avoided. On the other hand, Nb increases in-PWR corrosion if added even in small amounts (≥0.1%) to Zr-Sn-FeCr alloys with >0.4% Sn, as can be seen from Figure 7-1.

- The TM content is especially important for the corrosion behaviour of Nb free Zr-Sn alloys with a need of an addition of at least 0.3% or better ≥0.4% (see Figure 7-1 and information reported by [Seibold & Garzarolli, 2002]) for a reasonable low in-PWR corrosion rate. In case of Nb containing alloys, TM additions up to 0.1% are probably always beneficial for corrosion resistance, however, larger additions may increase in-PWR corrosion rate if the Sn content is >0.3% (Figure 7-3). Considering the composition of the **New Alternative Zr Alloys** there seems to be some thoughts that small Cu additions (0.05-0.12%) and potentially also Cr additions may improve corrosion resistance even more than Fe addition, at least for Nb containing alloys.

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Appendix A - Design bases

A.1 Mechanical design bases during normal operation, AOO and DBAs

A.1.1 Introduction

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

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

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

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

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

- Class III events have low probability and the potential for small radioactive release outside the plant site. In these postulated events the core would remain covered with water. Examples of these events are:
  – Small pipe break.
  – Loss-of-flow accident.
- Class IV events have very low probability and are potentially more severe; these are called DBAs. The most severe DBA is considered to be a complete (double-ended) rupture of a large pipe, ranging in diameter from 0.61 to 1.07 m (about 2 to 3.3 feet), in the primary coolant circuit of a PWR/VVER. These DBAs are denoted LBLOCAs. Other DBAs are:
  – Earthquakes, Tornadoes, and Flooding.
  – Control Element Ejection, RIA.
  – Spent-Fuel Handling Accident.

In a severe accident the coolability criteria cannot be maintained, resulting in gross oxidation and core melt.

Table A-1: Probability of different reactor events.

Class	Event types	Acceptance criterion	Probability per year	Examples
I	Normal operation	No fuel failures	1.	Full power operation, refuelling
II	Anticipated transients		$1\text{-}10^{-2}$	Loss of feed-water, pump trip, turbine trip
III	Anticipated transients with additional equipment failures	Fuel failures OK, but fuel should retain coolable geometry	$<10^{-2}$	Break outside containment, small primary break, turbine trip without bypass with scram on second signal
IV	DBAs		$<10^{-4}$	Large break LOCA, Falling control rod
	Severe accidents		$<10^{-6}$	Station blackout, LOCA with large leak from drywell to wetwell
				ANT International, 2011

In the design process, event types, probabilities and acceptance criteria are typically addressed by means of *fuel system*[51] safety reviews. The objectives of such reviews are:

1) The *fuel system* is *not damaged*[52] as a result of normal operation and AOOs.

2) *Fuel system* damage is never so severe as to prevent control rod insertion when it is required.

3) The number of *FR failures*[53] is not underestimated for postulated accidents.

4) *Coolability*[54] is always maintained.

Objective (1) in the above list is formalized in General Design Criterion 10, GDC10 [10CFR Part 50 Appendix A, 1990], see [Strasser et al, 2010a]. The application of GDC10 is described in the SRP [NUREG-800, 2007], see [Strasser et al. 2010a]. The *fuel system*, nuclear, and thermal and hydraulic designs are covered in SRP Sections 4.2, 4.3 and 4.4, respectively. Section 4.2 in SRP identifies a number of *fuel system* failure mechanisms that actually have occurred in commercial reactors, as well as hypothesized *fuel system* failure mechanisms. For each of these *fuel system* failure mechanisms, SRP section 4.2 lists a corresponding design limit intended to accomplish objective (1) in the list above. These design limits are called Specified Acceptable Fuel Design Limits (SAFDLs). Thus, the SRP does not include any design limits to address potential new *fuel system* failure mechanisms related to more recent fuel designs and/or reactor operation strategies.

FR failures must be accounted for in the dose analysis required by 10CFR Part 100[55] for postulated accidents.

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[51] Fuel system consists of assemblies of FRs including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in FRs; holddown spring, connections, spacer grids and springs; end plates; channel boxes; and reactivity control elements that extend from the coupling interface of the control rod drive mechanism in the core.
[52] The avoidance of damaged means not only that the fuel integrity is maintained, i.e., no release of radioactivity, but also that the fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements GDC10 and the design limits that accomplish this are called SAFDLs.
[53] FR failure means that the fuel cladding has been breached and radioactivity from the fuel get access to the coolant.
[54] Coolability means that the FA retains its rod-bundle geometry with adequate coolant to permit removal of residual heat even after a severe accident.

The general requirements to maintain control rod insertability and core coolability appear in the GDC27 and GDC35. Specific coolability requirements for the LOCAs are provided in 10CFR Part 50[56]

The *fuel system* design bases must take the four objectives described on the previous page into account. The SAFDLs discussed below do this. In a few cases the SAFDLs provide the design limit but in most cases it is up to the fuel vendor to recommend a design limit value, taking a specific failure mechanism into account. The fuel vendor must also provide the background data for the design limits (that are specified by the NRC as well as those used by the specific fuel vendor) to ensure that the design limit is both necessary and sufficient. The fuel vendor must also provide data for the specific fuel design that shows that the design limit is met to get their fuel licensed.

Specific failure mechanisms for the *fuel system* (including *the FR*) and licensing criteria related to classes I and II operation, class III and IV events, are listed in the following.

A.1.2 Class I and II design criteria

- *Fuel system damage* - to meet the requirements of GDC10 as it relates to SAFDLs for normal operation, including AOOs, fuel system damage criteria should be given for all known damage mechanisms related to:
  - Stresses and strains due to steady state and transient loads.
  - Fatigue
  - Fretting
  - Oxidation, hydriding, CRUD deposition.
  - Diametral and axial dimensional changes, including FR and channel axial growth.
  - Rod internal pressure as it affects pellet-cladding gap reopening and axial propagation of DNB.
  - Fuel centreline temperature.
  - End plug weld integrity.
  - Hydraulic loads.
  - Loss of control rod reactivity, i.e., control rod reactivity should be maintained.
- *FR failure* - to meet the requirements of GDC10 as it relates to SAFDLs for normal operation, including AOOs, FR failure criteria should be given for all known FR failure mechanisms related to:
  - Internal hydriding.
  - Cladding collapse.
  - Overheating of cladding.
  - Thermal margin criteria DNBR and CPR must not be violated.
  - Overheating of pellets.

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[55] 10CFR Part 100, “Reactor Site Criteria”, NRC– see [Strasser et al, 2010a],
[56] 10CFR Part 50, §50.46, “Acceptance criteria for ECCSs for light water nuclear power reactors”, NRC– see [Strasser et al, 2010a],

Appendix B - References

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