ZIRAT-6 Annual Report

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1   INTRODUCTION

The objective of the Annual Review of Zirconium Alloy Technology (ZIRAT) is to review and evaluate the latest developments in zirconium alloy technology as they apply to nuclear fuel design and performance. The large amount of data presented at technical meetings, published in the literature, and provided through discussion with the vendors in the year 2001 are evaluated and discussed to highlight the significance of the results and their potential effect on fuel performance and reliability. The review is a joint effort of Aquarius Services Corporation (Aquarius) and Advanced Nuclear Technology Sweden AB (ANT).

The primary issues addressed in the review and this report are zirconium alloy research and development, fabrication, component design, ex- and in-reactor performance including:

- Regulatory bodies and utility perspectives related to fuel performance issues, fuel vendor developments of new fuel design to meet the fuel performance issues

- Fabrication and quality control of zirconium manufacturing, zirconium alloy systems

- Mechanical properties and their test methods (that are not covered in any other section in the report)

- Dimensional stability (growth and creep)

- Primary coolant chemistry and its effect on zirconium alloy component performance

- Corrosion and hydriding mechanisms and performance of commercial alloys

- Cladding primary failures

- Post-failure degradation of failed fuel

- Cladding performance in postulated accidents (LOCA, RIA)

- Dry and wet storage

- Potential burnup limitations

- Current uncertainties and issues needing solution are identified throughout the report.

Background data from prior periods have been included wherever needed. All the data are from non-proprietary sources; however, their compilation, evaluations, and conclusions in the report are proprietary to Aquarius, ANT and ZIRAT members as noted on the title page.

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The work reported herein will be presented in two Seminars: one in Rome on January 15-17, 2002, and one in Cocoa Beach, Florida, on January 29-31, 2002.

The term of ZIRAT-6 is January 1 to December 31, 2001.

2 UTILITY PERSPECTIVE/EXPERIENCE, REGULATORY BODIES, AND FUEL VENDOR BURNUP EXPERIENCE
Peter Rudling

2.1 INTRODUCTION

2.1.1 Regulatory perspective

The historical background for the burnup limits and the fuel design criteria used by the US industry today. By the mid 1980s, NRC had de-emphasized its work on reactor fuels and shifted most of those resources to severe accidents in the wake of the accident at Three Mile Island. Yet this was a time when the industry was moving into new operating regimes with fuel and developing new fuel designs. In response to a request from NRC, all five fuel vendors in the U.S. submitted licensing topical reports requesting approval to apply their safety analysis methods to high-burnup fuel. These reports were given NRC approval in 1985 and 1986 for various burnup levels up to 60 GWd/t average burnup for the peak rod. As a result of the NRC review of these reports, the regulatory burnup limit was later raised to 62 GWd/t. The embrittlement criteria used for analysing a loss-of-coolant accident (17% cladding oxidation and 2200°F peak cladding temperature) were believed to be unaffected by extended burnup operation because beginning-of-life conditions were expected to be limiting.

The following example results generated during the 15 years since those approvals were given indicate that the design criteria related to postulated accidents may not be conservative:

- Cladding oxidation can approach the 17% LOCA limit during normal operation in this burnup range and that the associated hydride accumulation can embrittle the underlying metal.

- Recent testing under reactivity accident conditions of fuel with burnups around 60 GWd/the has demonstrated that cladding failure and fuel dispersal can occur well below the 280 cal/g level, thus calling into question the fuel enthalpy criteria being used for these events.

- Changes in pellet properties at high burnup, along with the rim structure that develops, also alter the thermal performance of fuel rods.

- NRC has licensed two new cladding materials so far. In 1990, Westinghouse submitted a licensing topical report on their Vantage+ fuel assembly in which they introduced ZIRLO cladding.

In 1997, Framatome ANP submitted a licensing topical report on their cladding and structural material, M5. The NRC safety evaluation of this report and its conclusions were quite similar to those for ZIRLO according to Meyer, 2000, with the following exceptions.

- The Zircaloy embrittlement criteria used for analysing a loss-of-coolant accident (17% cladding oxidation and 2200°F peak cladding temperature) were concluded to be applicable to M5 based on quench tests with M5 cladding. Although the test specimens were unirradiated and were not subjected to ballooning deformation, the criteria were thought to be acceptable up to currently approved burnup levels (i.e., 62 GWd/t).

- The safety evaluation report noted that recent testing on the fuel enthalpy limit (280 cal/g) for reactivity accidents has indicated that fuel expulsion and fuel failure may occur before the 280 cal/g limit and the onset of departure from nucleate boiling (DNB), respectively. It was concluded that further testing and evaluation are needed and the limits may decrease in the future, but the current limits will continue to be accepted. It was also concluded that there would be little impact on the use of M5 cladding on fuel expulsion and failure (compared to the use of Zircaloy-4) as long as the cladding remains ductile.

- Fuel assembly structural analysis methodology for seismic and LOCA loads was unchanged, but would require strength values for M5, should M5 be used for guide tubes or thimble tubes.

The OECD/CSNI/PWG2 group has formed a Task Force on Fuel Safety Criteria, TFFSC. The objective of this group consisting of staff from different regulatory agencies from US and western Europe are to assess if the development of new fuel and core designs, and the replacement of statistical instead of deterministic analysis methods have decreased the fuel safety margins. The different safety related criteria being reviewed are listed in Table 2-1.

Table 2-1: **Fuel safety criteria, and task force review basis, the numbers designate the following: I – normal operation, II – anticipated operational occurrences, III and IV – accidents.**

Safety Related Criteria	"New" Elements Affecting Criteria									
	New Fuel Design	New Core Design	New Cladding Materials	New Manu Methods	Longer Fuel Cycles	Uprated Power	High Burnup	MOX Mixed Core	Water Chem Changes	Current/New Operating Pract
CPR/DNBR	I-IV	I-IV			I-IV	I-IV	I-IV		I-IV	
Reactivity Coefficient		II-IV			II-IV	II-IV	II-IV	II-IV	II-IV	
Shutdown Margin	I-IV	I-IV			I-IV	I-IV	I-IV	I-IV		I-IV
Enrichment	I-IV	I-IV			I-IV					
CRUD deposition	I	I	I	I	I		I		I	
Strain Level	I-II						I-II	I-II		
Oxidation			I-IV	I-IV			I-IV	I-IV	I-IV	
Hydride Concentration			I-IV	I-IV			I-IV	I-IV	I-IV	
Internal Gas Pressure	I-IV				I-IV	I-IV	I-IV	I-IV		
PCMI	I-II		I-II	I-II			I-II			
PCI	I-IV	I-IV	I-IV	I-IV		I-IV	I-IV	I-IV		I-IV
Fragmentation (RIA)							III-IV	III-IV		
Fuel Failue (RIA)	III-IV		III-IV	III-IV			III-IV	III-IV		
Peak Cladding Temperature			III-IV	III-IV			III-IV	III-IV		
Cladding Embrittlement/Oxidation			III-IV	III-IV			III-IV	III-IV		
Blowdown/Seismic Load			III-IV				III-IV			
Assembly Holddown Force	I-IV									I-IV
Coolant Activity					I-IV	I-IV	I-IV	I-IV		
Gap Activity					III-IV	III-IV	III-IV	III-IV		
Source Term					III-IV	III-IV	III-IV	III-IV		

The NRC regulatory branch has discussed the requirements to be met to increase the burnup levels beyond those approved today and focused on a series of fuel issues related to normal operation (class I) and anticipated operational occurrences (class II) that were reported during the last years, as follows:

- oxidation levels higher than predicted,

- excessive internal gas pressure in burnable poison rods,

- incomplete control rod insertion (IRI) events,

- large axial offsets or axial offset anomalies (AOA),

- fuel failures due to high fuel duty,

- adverse effects of water chemistry,

- high crud buildup, and

- accelerated growth of rods and assemblies.

All of these problems have been at least partially associated with high burnup fuel or aggressive fuel duty according to NRC Regulatory Branch. To date, these events have not raised safety questions serious enough to require regulatory action such as a restriction of a burnup limit. However, to allow a burnup extension from that existing today in US of 62 GWd/T rod average the industry must address all of the above mentioned issues argues the NRC Regulatory Branch. The nuclear industry in the United States has already indicated that it would like to increase the peak rod average of 75 GWd/T for pressurized-water reactors (PWRs) and 70GWd/T for boiling-water reactors (BWRs) and that this will be done with an enrichment of 5 weight percent U-235 or less.

2.1.2 Utility perspective

One of the major present challenges to nuclear energy lies in its competitiveness. To stay competitive the industry needs to reduce maintenance and fuel cycle costs, while enhancing safety features. To meet these objectives, utilities are looking into the possibilities to:

A. Increase the burnup

B. Longer fuel cycles

C. To use high duty fuel management schemes

D. Reactor power uprating

E. Achieve clean cores, i.e., absence of defects and low system radiation buildup (may be obtained by water chemistry changes such as high pH in PWRs and Zn additions in BWRs and PWRs)

F. Decrease tendency for stress corrosion cracking in reactor components such as piping (may be achieved by water chemistry changes such as Hydrogen Water Chemistry and Noble Metal Addition in BWRs and Zn additions in PWRs)

G. Increase operating flexibility (load following)

H. Continuously improve operating safety margins.

The situation for the nuclear industry is complicated since implementing A. through G. in the list above would normally decrease the operating safety margins (item H.). However, possibilities to implement all items A. through H. may be given by introduction of new advanced fuel designs (with new materials) as well as better models to predict fuel behaviour during class I through IV operation. However, other situations that may make it more difficult to improve operating safety margins are the following:

- There is a drainage of competence in the nuclear industry, very knowledgeable experts are retiring while very few "bright" young people are attracted to the nuclear industry

- Fuel vendors have less resources to solve fuel related issues for utility customers. This situation is driven by the deregulation of the electrical market that has reduced the price for nuclear fuel.

2.1.3 Vendor perspective

As mentioned above, the fuel vendor needs to develop new more advanced fuel designs that can withstand the more severe fuel duty. This section2.1.3 gives a summary of new fuel designs previously presented in the ZIRAT-5 report.

2.1.3.1 *Framatome ANP GmbH*

Framatome ANP GmbH PWR program has yielded the materials shown in Table 2-2

Table 2-2:   Material development for PWR fuel assemblies.

Boundary Condition	Cladding (year of introduction of full reloads)	Structural Parts
Meet ASTM specification[*)]	**Optimization of Zry-4**	
	Low-Sn Zry-4 (1987) Optimized Zry-4 (1989)	Low-Sn Zry-4 (1989)
In total, meet ASTM specification[*)]	**Implementation of the duplex concept (DX) for cladding**	
	Duplex Extra Low-Sn (0.5-0.8%) /2/ (DX ELS, DX D4) (1989/90)	
Keep close to ASTM specification[*)]	**Insertion of an advanced Zry-type alloy**	
	Modified Zry-4 (1995)	Modified Zry-4 (1997)
High burnup, severe fuel duty, alloy composition not limited to ASTM specification	**Development of high performance alloys (since 1985)**	
	Zr1Nb, DX Zr2.5Nb Zr0.6SnFeV (HPA-4, DX and throughwall)	Zr0.6SnFeV(HPA-4)

*) licensing reasons

The burnup experience on the PWR DUPLEX alloys are shown in Table 2-3. The first commercial DUPLEX ELS material used for reloads was DX ELS 0.8a inserted in 1991. This cladding material had a Sn content of 0.8%, the other alloying not very different from Zry-4. A new DUPLEX ELS cladding material with an enhanced Fe+Cr content (0.45%) was subsequently developed, DX ELS 0.8b, and used for reloads from 1992 onward. In parallel, the development of a DX ELS material with 0.5% Sn and 0.7% Fe+Cr was pursued.

Table 2-3:   Irradiation of DUPLEX ELS reloads.

PWR	First use	Material	Total no. fuel rods	Max FA burnup MWd/kgU	No. of rods PIE pool	PIE HC
A(15*15)	1991	DX ELS0.8a	14,437	56	153	1
B(16*16)	1992	DX ELS0.8a	13,452	60	12	
D(16*16)	1993	DX ELS0.8a	3,776	55	0	
A(15*15)	1992	DX ELS0.8b	56,440	59	376	13
B(16*16)	1994	DX ELS0.8b	46,256	49		
C(16*16)	1995	DX ELS0.8b	30,208	47	70	
D(16*16)	1993	DX ELS0.8b	50,964	56		
E(16*16)	1993	DX ELS0.8b	55,224	41	10	
F(16*16)	1994	DX ELS0.8b	45,312	51	?	
G(18*18)	1993	DX ELS0.8b	94,800	48	87	
H(18*18)	1996	DX ELS0.8b	34,800	47	30	
I(18*18)	1992	DX ELS0.8b	85,879	53	181	3
J(17*17)	1998	DX ELS0.8b	716	24	50	
L(14*14)	1998	DX ELS0.8b	448	7		
H(18*18)	1992	DX D4	18,000	41	12	
I(18*18)	1992	DX D4	2,400	52	48	4
M(17*17)	1994	DX D4	1,056	52	265	

The Framatome ANP GmbH BWR development program is shown in

Table 2-4. For high burn-up applications the intermetallic particle size was increased in the Zircaloy PGP+ variant compared to that of Zry-2 LTP. Also a Zr1SnNb alloy without any intermetallic particles have been tested and may be used for very high burnup levels.

Table 2-4:   Material development for BWR fuel assemblies

Objective	Measure	Introduction
Avoid severe nodular corrosion	ß -quenching	early 70s
Avoid PCI	Zr-liner cladding	1985
Minimize nodular and late accelerated uniform corrosion	Low Temperature Processed (LTP) cladding and sheet material with optimized A-parameter	1987
Avoid PCI and long splits in defective rods	Fe-enhanced Zr-liner cladding	1993
Provide PCI resistant throughwall cladding with improved corrosion and hydrogen pickup behavior	Fine grain size, highly texturized throughwall Zry-type-cladding (Fe, Cr, outside ASTM), alloy used also for structural parts (Zry-BWR)	1995*
Provide resistance against late accelerated uniform corrosion at high burnup	Particle size adjusted to high burnup needs for structural materials and cladding (Zry PGP+)	1998
High burnup, high strength alloy for thin-walled parts with irradiation resistant microstructure	Zr1SnNb for BWR application	1999*

*   lead fuel assemblies, pathfinder rods

The Framatome ANP GmbH BWR ATRIUM 10 fuel design is provided in Figure 2-1. The ULTRAFLOW spacer was introduced along with eight part length fuel rods and a small hole design lower tie plate. As of August 1999, more than 2000 ATRIUM 10 fuel assemblies have been inserted in 17 different boiling water reactors. The maximum fuel assembly average discharge burnup was 59 MWd/kgU in the year 2000. The FUELGUARD lower tie plate, is now introduced with increasing frequency in ATRIUM 10 reloads to reduce the risk of debris fretting failures.

A special version, ATRIUM 10P, was developed with about 15 % lower pressure drop by applying e.g. Inconel spacers, larger inside channel width and increased number of part length fuel rods. Lead assemblies of ATRIUM 10P have been inserted in a German BWR in 1996.

An "ATRIUM 10C" fuel assembly design is being developed to increase MCPR margin and thermal hydraulic stability by optimising number and axial positions of spacers as well as number and radial position of part length fuel rods.
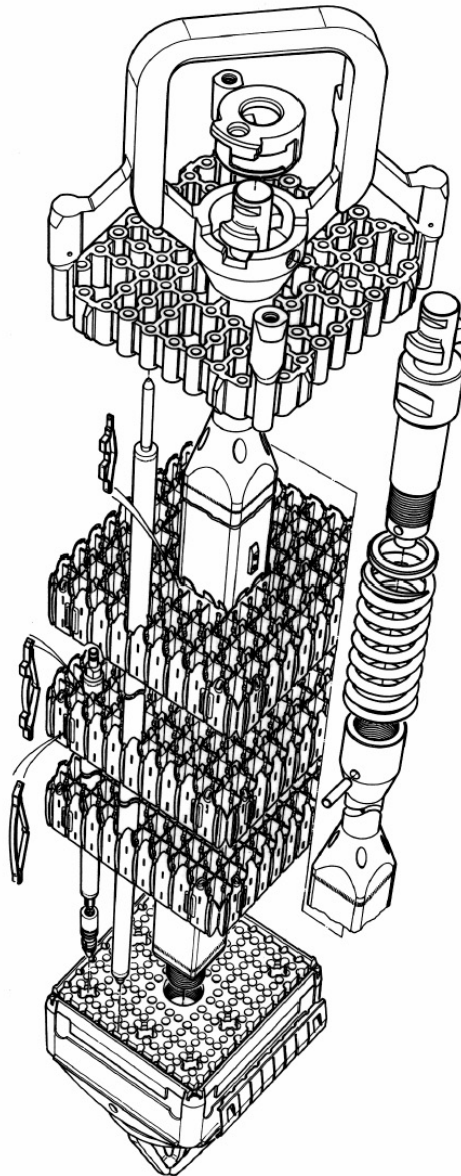
Figure 2-1:   The ATRIUM 10 fuel assembly.

2.1.3.2   *Mitsubishi*

Mitsubishi has developed the advanced fuel cladding material *MDA* intended for a discharged assembly burnup of 55GWd/t in collaboration with Japanese PWR utilities, Table 2-5.

Table 2-5: **Typical chemical composition of cladding, MDA-Mitsubishi Developed Alloy.**

Material	Sn	Fe	Cr	Nb	Zr
MDA	0.8	0.2	0.1	0.5	bal.
Low Sn Zr'y-4	1.3	0.2	0.1	-	bal.
con. Zr'y-4	1.5	0.2	0.1	-	bal.

2.1.3.3 NFI

In Japan, BWR fuel burnups have been extended in a stepwise manner with design changes from the design maximum burnup of 40GWd/t for Step I, to 50GWd/t for Step II, and to 55GWd/t for Step III designs, Figure 2-2.

Step I fuel is characterised by the 8X8 lattice bundle including 2 water rods and introduction of Zr-liner with a high corrosion resistance cladding tube, Figure 2-3.

Step II fuel also has a 8X8 lattice configuration of 60 fuel rods and a large diameter central water rod. Ferrule-type spacers were introduced in place of the former eggcrate ones to improve the thermal hydraulic characteristics, Figure 2-3.

In Step III fuel, the main design change was introduction of a 9X9 lattice configuration and two large diameter central water rods (type-A) or a water channel (type-B), which results in an optimized distribution of water and fuel, Figure 2-4. A small number of lead use assemblies (LUAs) of this design were loaded in commercial BWRs in 1996.

The next step design, Step III+, is currently under development. It also has a 9X9 lattice design and aims at an increased average discharge burnup up to around 50GWd/t keeping the same design maximum bundle burnup of 55GWd/t as the Step III design. The expected improvements are reduction of spent fuel amount by about 10% and fuel cycle cost by about 4%.

In order to respond to future demands for further improvements, high burnup 10X10 fuel is also a subject of development with the aim of an average discharge burnup higher than 60GWd/t.
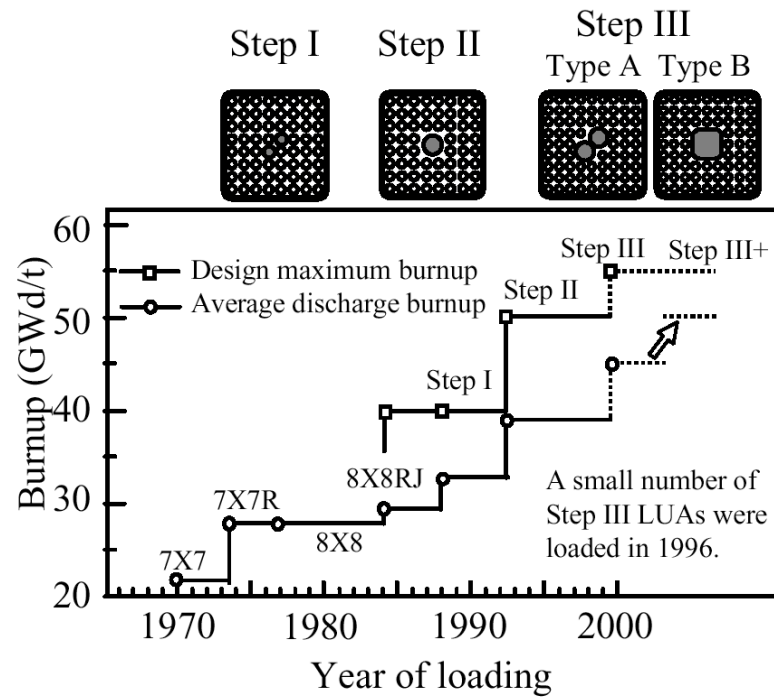
Figure 2-2:   Burnup extension trend in Japanese BWR fuel.
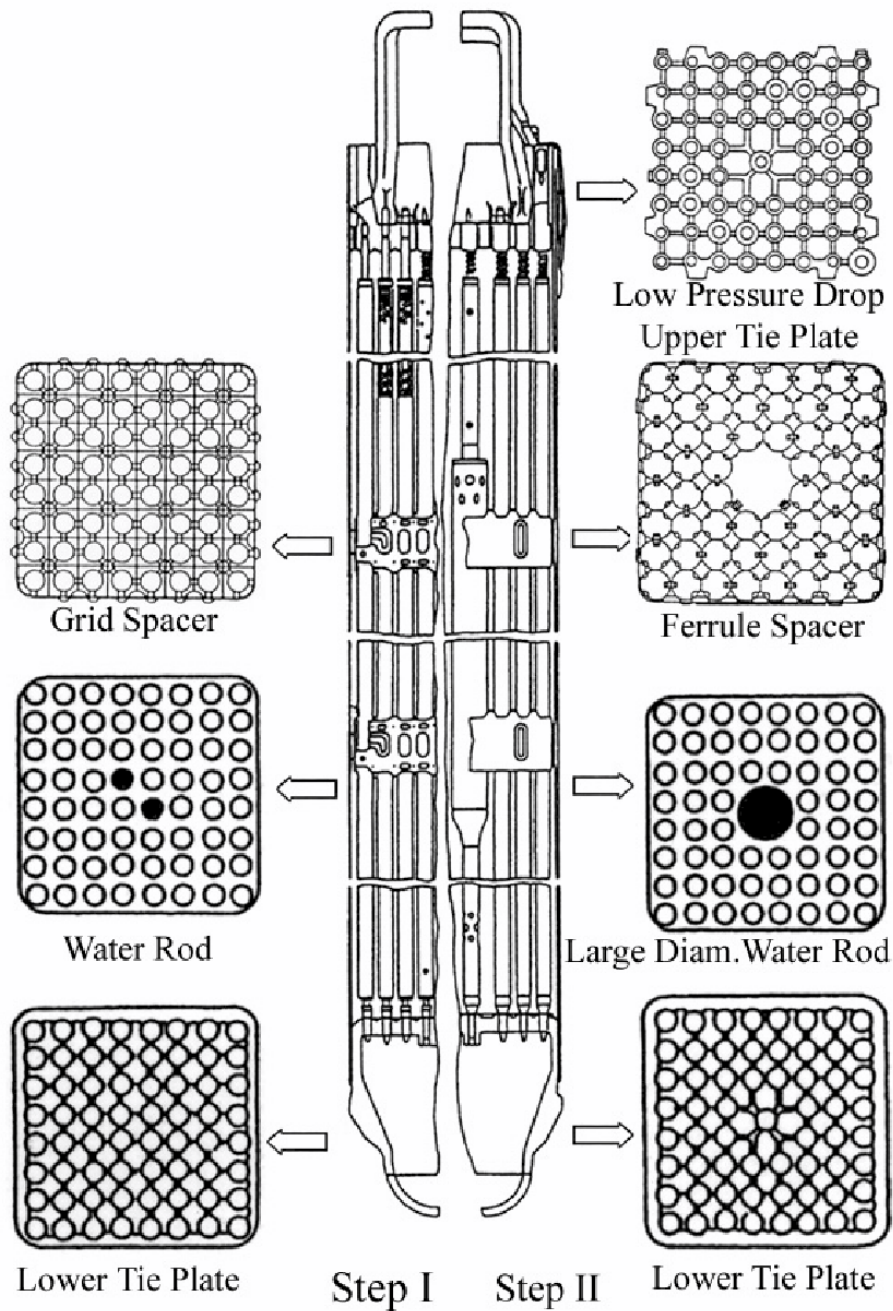
Figure 2-3:   Design Features of BWR Assembly.
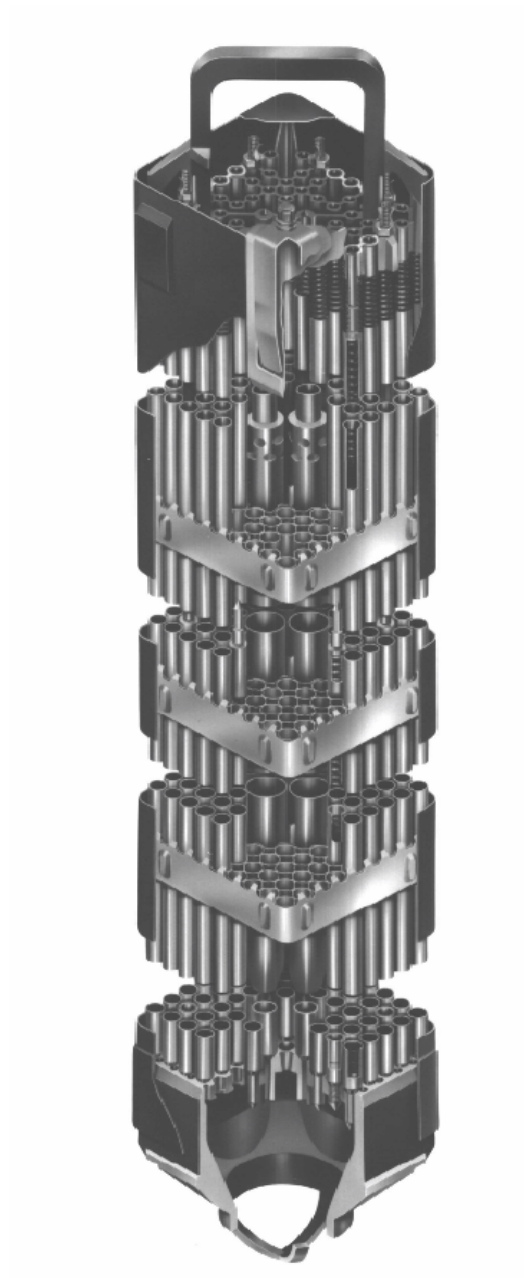
Figure 2-4:   Step III (Type A) fuel assembly.

2.1.3.4   Framatome ANP

The alloy M5 cladding is now fully licensed by the NRC for use in US PWR reactors and the irradiation experience is shown in Table 2-6.

Table 2-6: Irradiation experience of M5 cladding.

Number Of M5 Fuel Rods Irradiated	Number Of Reactors	Number Of Utilities	Rod Matrix	Maximum Burnup Achieved
>65,000	24	12	14x14 to 18x18	63 GWd/mtU

Location	Europe					USA	
Rod Matrix	14	15	16	17	18	15	17
Number Of Plants	1	1	4	13	2	1	2
Core Average Linear Power (W/cm)	220	190	207 to 211	170 to 186	166	190	178
Maximum Burnup GWd/mtU	51	38	48	63	51	45	40
T_{inlet} (°F)	545	543	556	549 to 552	556	556	558
T_{outlet} (°F)	601	612	622	612 to 615	621	606	622
Max. Coolant Temp (°F)	615	-	650	637	647	-	637
Max. Heat Flux (W/cm^2)	84	-	91	78	82	-	78

2.2 NEW RESULTS

2.2.1 Regulatory perspective

Chatterton, Ref. 2-10, reported that The U.S. Nuclear Regulatory Commission (NRC) is working with the industry to develop Guidelines for Lead Test Assemblies. The intention is to develop a set of guidelines which provides a structured process for regulating lead test assemblies (LTAs) while maintaining safety. Following such guidelines will help ensure uniformity in data collection and make evaluation of new properties or limits more predictable. These guidelines will be consistent with the NRC performance goals which are: maintain safety, increase public confidence, improve efficiency and effectiveness of regulation and reduce unnecessary regulatory burden. Chatterton argued that some fuel issues are in many cases related to that Lead Test Assemblies, LTAs, were not used or that the batch fuel that encountered some performance problems was not irradiated in the same manner as the LTAs.

The NRC staff has now conclude that LTAs should be prototypical in order to be of maximum value, Ref. 2-10. In the past, the NRC staff has restricted the LTAs to non-limiting locations. This has led to burnup histories that were not aggressive and in many cases not typical. Because of the large number of variables that affect fuel behaviour, it will be necessary to have a sufficient number of LTAs to cover the range of operation expected by a large number of plants. Establishing LTA guidelines will also reduce unnecessary regulatory burden since prior approval and identification of the program requirements will enable licensees and fuel vendors to factor the necessary requirements into planning without additional NRC staff involvement.

It is anticipated that most of the LTAs will reach peak rod burnups between the current licensed burnup limit of 62GWD/MTU to 68 GWD/MTU with a few rods obtaining higher burnups in the range of 68 to 75 GWD/MTU, Ref. 2-10. The types of data to be obtained include oxidation behaviour, growth behaviour, hydriding behaviour and fission gas behaviour.

LTAs and LTRs will be characterized prior to under-going exposure in the "test" cycle, Ref. 2-10. Pre-characterization will consist of at least visual examinations, oxidation measurements, and growth measurements. These will be compared with predicted performance to ensure that the fuel behavior has been as expected in prior cycles.

Post-irradiation characterization of the LTA or LTR will consist of at least visual examinations, oxidation measurements, and growth measurements. The guidelines being developed will specify when additional pool-side measurements should be taken and when and which hot-cell examinations are indicated, Ref. 2-10.

The number of LTAs within a given core must be limited. The maximum number is under discussion. For PWRs, eight might be a reasonable number assuming adequate justification is provided, Ref. 2-10. LTAs should be placed such that the data obtained is prototypical in terms of power density, temperature and duty. However, they should not be in the absolute limiting location with respect to these parameters.

2.2.1.1 *Utility perspective*

2.2.1.1.1 *Utility perspective*

The deregulated market in Spain has resulted in a greater emphasis on utility cost efficiency, Ref. 2-3. In Spain the electricity produced by nuclear power is very competitive compared to other energy sources, Figure 2-5. Mergers of nuclear utilities during the last decade in Spain have decreased the current number of utilities to four, two big and two small utilities. Partly due to these mergers the nuclear fuel costs has decreased, Figure 2-6.
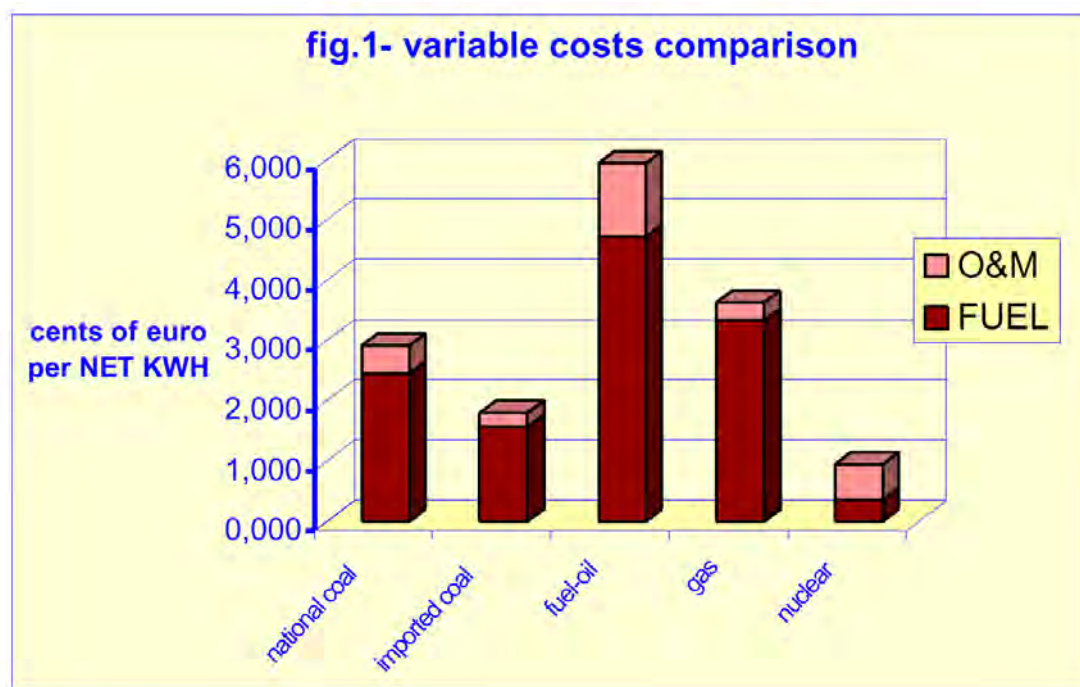
Figure 2-5:   Cost comparison between different energy sources in Spain, Ref. 2-3.

FIG.2 - NUCLEAR FUEL COST IN SPANISH PWR-W
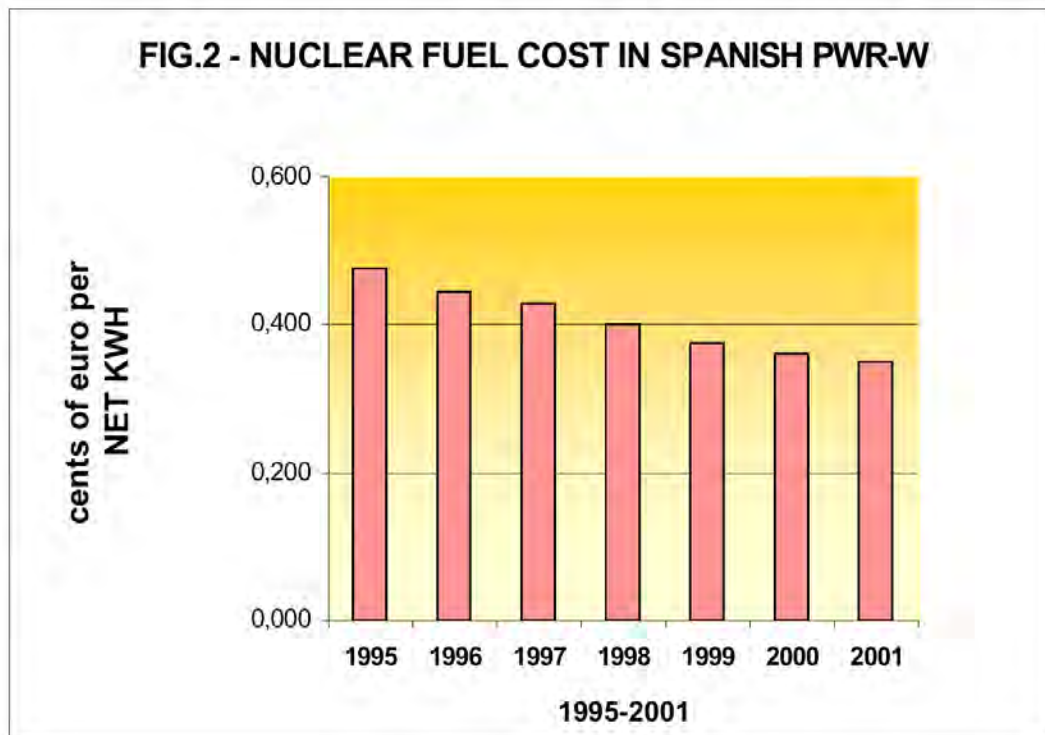
Figure 2-6:   Nuclear fuel costs in Spain during the years 1995-2001, Ref. 2-3.

2.2.1.1.2   TVO

TVO's fuel costs and overall production costs are competitive through an efficient cooperation with competent fuel suppliers and authorities and through the following arrangements, Ref. 2-4:

- A robust fuel with large operating margins are many times preferable compared to that of a fuel design optimised in many respects. The most expensive fuel is that which causes unscheduled shutdowns due to fuel failures or licensing problems. This is why TVO e.g. for many years used the 9x9-1 fuel instead of the modern 9x9 fuels, and switched directly into the 10x10 fuels when enough experience existed. On the other hand, TVO are also open to adopt quite new fuel types when it is deemed necessary such as e.g. when power uprating necessitated a change in fuel design.

- Annual fuel cycle still gives the lowest overall power production costs for TVO, provided that the annual reloading and maintenance is managed in about 2 weeks.

- TVO is keeping a stock of fresh fuel assemblies at the power plant for securing the plant operation against transport and delivery disruptions. The minimum stock by law in Finland is for 7 months of consumption. With this fuel stock TVO can load a core that can produce power for nearly one year with a long coast down operation. There is also a wet interim storage facility for spent fuel at the TVO site and consequently TVO is not facing any spent fuel transportation problems, which could jeopardise the plant operation.

- TVO has some uranium concentrates in stock, mainly for commercial purposes. The uranium is purchased when uranium prices are low. The uranium stock is mainly at a European converter for the security of supply.

- TVO makes medium-term contracts with qualified suppliers for uranium, conversion, enrichment and fuel manufacturing. Very long-term contracts are difficult, since TVO prefer fixed prices. Reliable transport arrangements, all now outsourced to suppliers, are also appreciated.

- TVO has in-house resources, including specialists in reactor physics, safety, materials, chemistry and quality assurance.

In the future, also TVO will increase burnup since this will result in fuel savings even if TVO's back-end costs are low. Extensive burn up studies are now underway with the suppliers of TVO.

Lunabba et al, 2001, also reported that additional nuclear power capacity would be competitive against any other type of power plants except existing hydro and coal plants.

2.2.1.1.3   Fortum

Teräsvirta et al., Ref. 2-11, stated that in March 1996 BNFL signed a contract with Fortum Power and Heat (formerly Imatran Voima Oy) and PAKS Nuclear Power Plant to deliver 5 Lead Test Assemblies (LTAs) of a new type of fuel to Fortum's VVER-440 reactor at Loviisa in Finland. These 5 assemblies was loaded into the core of Unit 2 in September of 1998.

Notable differences in the fuel design compared to the standard Russian VVER-440 fuel design are:

- use of Zircaloy 4 as fuel rod cladding and hexagonal shroud material instead of Zr1%Nb or Zr2,5%Nb alloy respectively,

- solid pellets instead of hollow pellets and

- thinner rod diameter, 8.9 mm instead of 9.1 mm, which somewhat increases the reactivity of the assemblies via better moderation.

The irradiation of the LTAs is planned such that two of the fixed fuel assemblies will reach as high burnup as possible and the two other fixed assemblies will experience as high as possible increase in the power level between the first and second cycle, Figure 2-7. The outlet temperatures are shown in Figure 2-8.
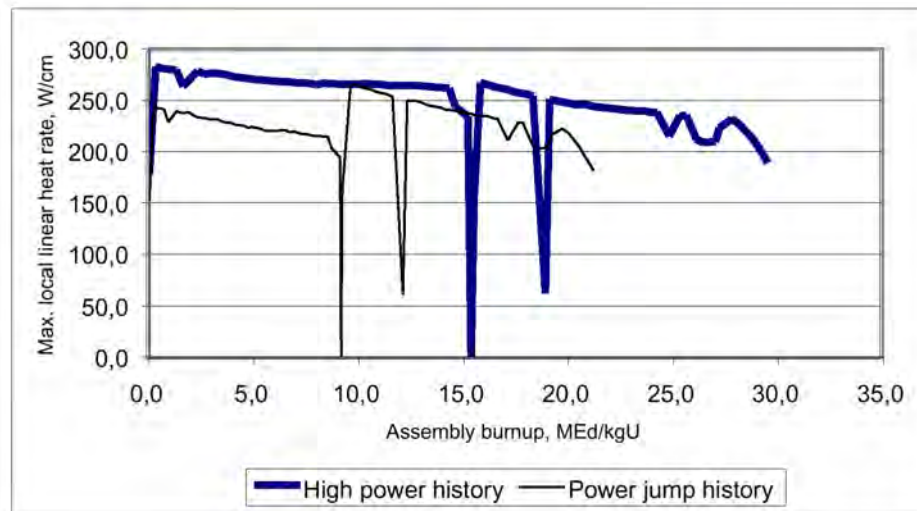
Figure 2-7   Two power history types for the LTAs during first two cycles, Ref. 2-11
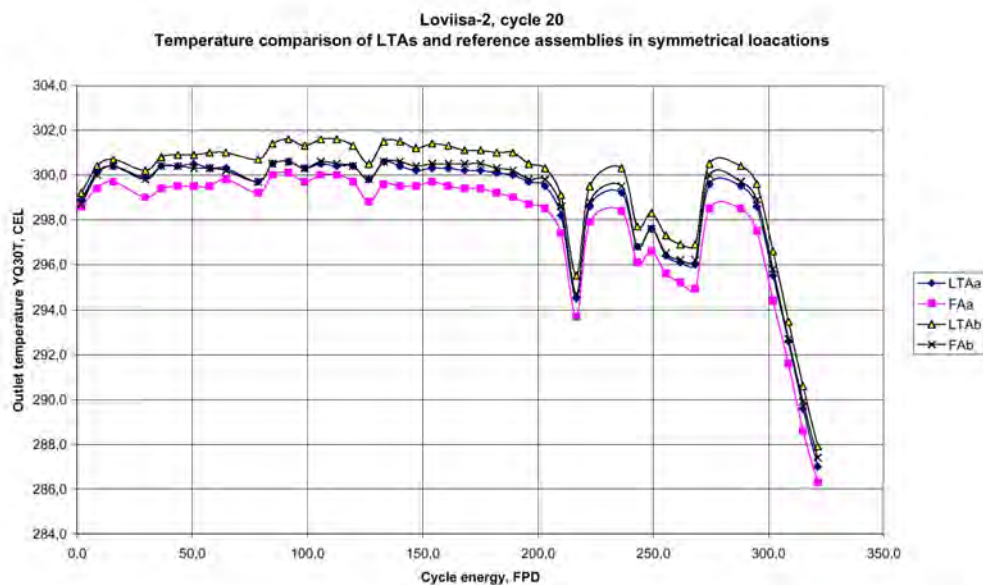
Figure 2-8   Outlet temperatures of two LTAs and two reference fuel assemblies in a symmetrical core location, Loviisa-2, 20 th cycle, Ref. 2-11

General visual appearance of the LTAs after one and two cycles was good. Figure 2-9 shows a top nozzle of LTA2 after 2 cycles. Examples of oxidation patterns on a fuel rod are seen in Figure 2-10. In LTA2 there was also a distinct difference between the two sides of the bundle, one side having clearly less oxide spots than the other. This is demonstrated in Figure 2-11. The difference in the oxidation correlates with the power tilt that the assembly experienced, in particular during its first cycle.
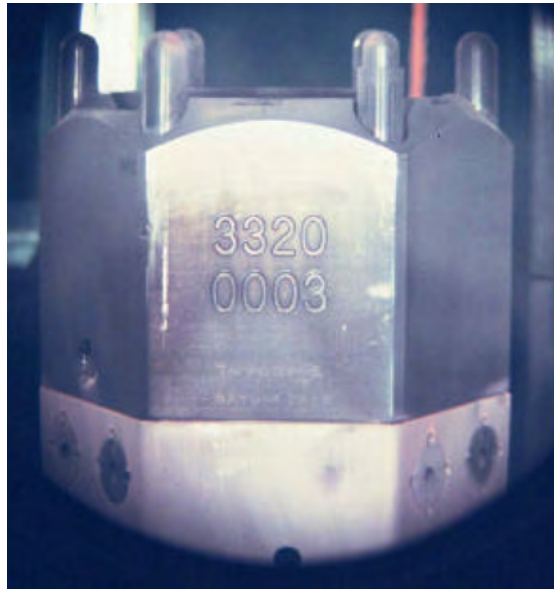
Figure 2-9   Top nozzle of the LTA after two cycles, Ref. 2-11

Figure 2-10   Typical oxide spots on a fuel rod, Ref. 2-11

**Figure 2-11   Difference in the amount of oxide spots on the left and right hand
                side ofthe bundle, Ref. 2-11**

Figure 2-12 shows the irradiation growth of LTA1 and LTA2. Both measurement results
fall below the MATPRO prediction. There were almost no changes in the bow of LTA1,
which was located in the middle of the core. Some changes were observed in the bow of
LTA2, which was in the periphery of the core, and subsequently in a power tilt, during
the first cycle. During its second cycle LTA2 was also in a power tilt in the same
direction but with less tilt than during the first cycle. All in all the bow of the assemblies
was quite small, maximum value being 1.4 mm.
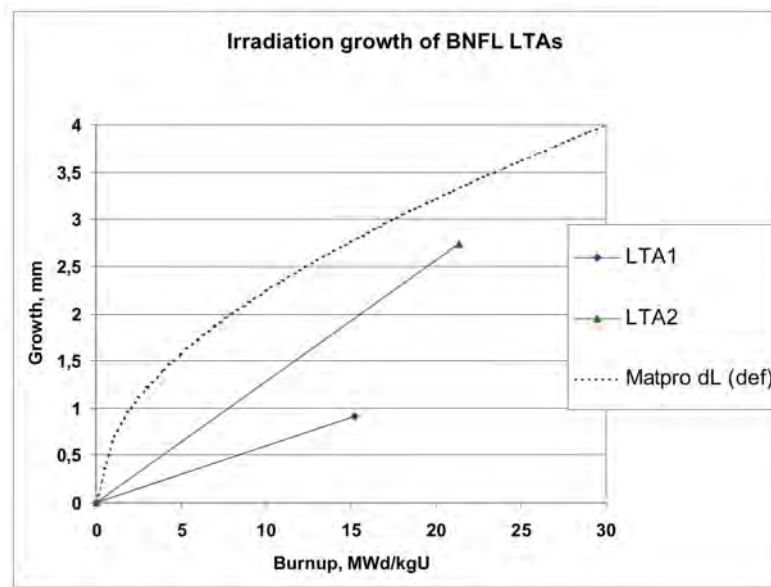
Figure 2-12   Irradiation groth of the LTAs after one and two cycles Assembly bow, Ref. 2-11

Fuel rod diameters were measured for the six corner rods of LTA1 and LTA2. Using the fresh cladding diameter data from the pre-characterisation cladding creep down for each rod versus the axial position was estimated. Figure 7 shows the average cladding creepdown of the corner rods in LTA1 and LTA2.
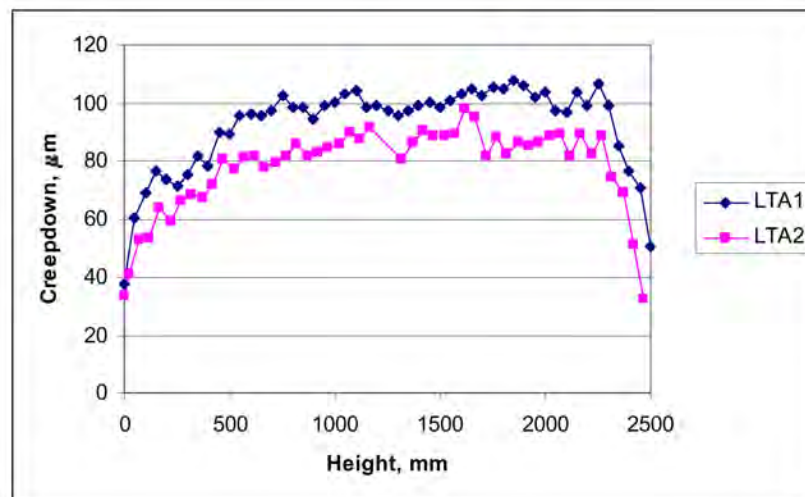
Figure 2-13   Cladding creep down of corner rods of LTA1 (after one cycle) and LTA2 (after two cycles), Ref. 2-11

For LTA2 the lengths of a total of 80 fuel rods were measured with a special tool attached to the poolside inspection equipment. The fuel rod growth values versus rod average burnup are presented in Figure 2-14. Most of the rods show growth of about 6 to 8 mm in the low end of the burnup range, increasing to about 10 to 12 mm in the highest burnup rods. The measured data are compared with estimates due to irradiation growth based on the MATPRO correlation. The solid line was calculated assuming a texture factor of 0.05 and cold work value of 0.2, whereas for the dashed line respectively 0.01 and 0.4 were used, Ref. 2-11. The authors argue that the discrepancy between the MATPRO correlation and the actual data can be due to uncertainties in the irradiation growth correlation. It is also likely that, there is already contact between the pellet and the cladding. Thus the data show not only rod growth due to irradiation growth of the cladding, but also a component from PCMI.
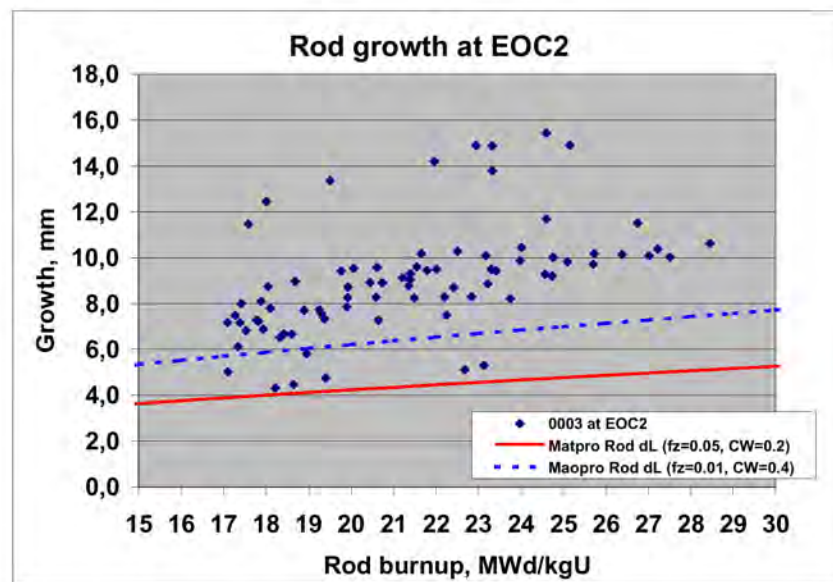
Figure 2-14Fuel growth of 80 fuel rods in LTA2 after two cycles, Ref. 2-11

Figure 2-15 shows the average oxide thickness on one face of LTA1 and LTA2 as a function of assembly height.

3 ZIRCONIUM ALLOY SYSTEMS
Peter Rudling

3.1 INTRODUCTION

The material performance in-reactor is a function of the reactor environment as well as the material microstructure. The microstructure depends on the chemical composition and the manufacturing process of the alloy. Characterisation of the material microstructure in relation to the material performance may elucidate the mechanisms behind the material performance. This knowledge may be used by fuel vendors to modify the manufacturing process to get optimum material performance in-reactor.

3.2 ZIRCALOY

3.2.1 Oxide characterisation

To get a better mechanistic understanding of the corrosion process, it is important to study the microstructural changes that occurs in the oxide. It may be that there is a specific change in the oxide microstructure that e.g. results in the transition in corrosion rate at 2-3 microns.

Pecheur et al., 2001, studied the chemical states of tin and iron atoms in 0.3 to 6 μm-thick oxide films grown in an autoclave, at 633 K, in water containing 10 ppm Li and 650 ppm B, by Mössbauer spectroscopy. The Zr-1.3% Sn-1% Fe-0.5% Cr-0.12% O material used was enriched with the ^{119}Sn and ^{57}Fe isotopes during alloy melting, to improve the signal to noise ratio of the ^{119}Sn and ^{57}Fe Mössbauer spectra. To further enhance the ^{57}Fe Mössbauer spectra the iron content was increased from 0.2 to 1% to and to maintain the same Fe/Cr ratio as in Zircaloy-4, the Cr content was increased to 0.5%. β quenching, hot and cold working was performed after the melting. Final heat treatment was performed at 873 K during 3 hours. The resulting microstructure was α-Zr equiaxed recristallised grains with Zr(Fe, Cr)$_2$ precipitates.

The corrosion results indicated a rate transition at an oxide film thickness close to 2 μm after about 100 days of oxidation.

To quantify the bulk relative concentrations of the different tin species, C_i ($C_i/\Sigma C_i$) were calculated for various oxide film thicknesses. The calculations were performed on spectra recorded in transmission mode at 77 K, fitted with and without assuming β-Sn line. The Sn^{4+}, Sn^{2+} and β-Sn relative concentrations versus the oxide film thickness are plotted in Figure 3-1. For the pre-transition oxide film (1.6 µm-thick), the Sn^{4+}, Sn^{2+} and β-Sn relative concentrations are comparable. However, when the oxide thickness increases, the Sn^{4+} relative concentration increases, while that of Sn^{2+} and β-Sn decreases. For the thickest oxide film (6 µm-thick), the Sn^{4+} relative concentration is significantly larger than that of Sn^{2+}. This indicates that tin atoms are oxidised during the oxide growth.

For the CEMS spectra (surface analysis), similar calculations were conducted. However, for the CEMS spectra, the results does not correspond to a concentrations since the probability of the conversion electrons decreases versus the depth of the oxide layer. The main result of this investigation is that, at the transition point (oxide thickness close to 1.9 µm), the Sn^{4+} relative intensity increases strongly with a corresponding decrease of the Sn^{2+} relative intensity (and the disappearance of β-Sn) (Figure 3-2b). Thus, the transition can be associated with the tin oxidation. In the post-transition period, no significant change in Sn^{4+} and Sn^{2+} relative intensities is observed.

Both the ^{57}Fe Mössbauer and CEMS studies were also able to identify different chemical states of iron: Fe_3O_4, Fe^{2+}, Fe^{3+}.

Most likely, the tin oxidation is a result of increased oxidation rather than causing the increased oxidation.
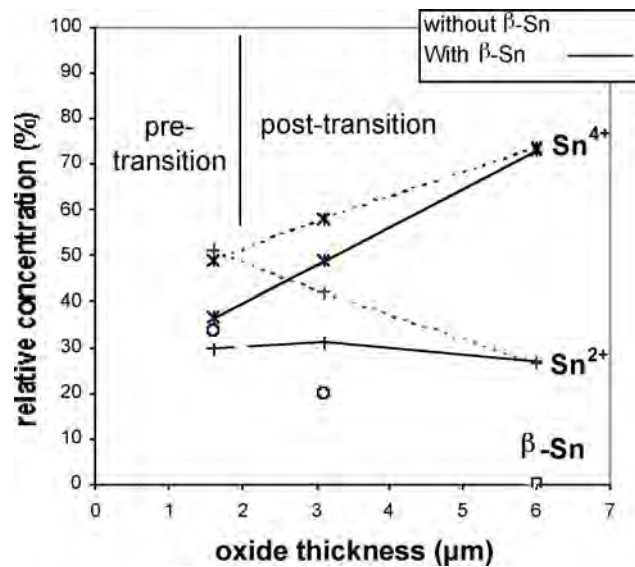
Figure 3-1: Sn^{4+}, Sn^{2+} and β-Sn relative concentration (a) and intensity (b) versus the oxide film thickness, calculated on spectra fitted with and without assuming β-Sn lines. Results based on spectra taken : (a) in transmission mode (bulk characterisation), Ref. 3-2
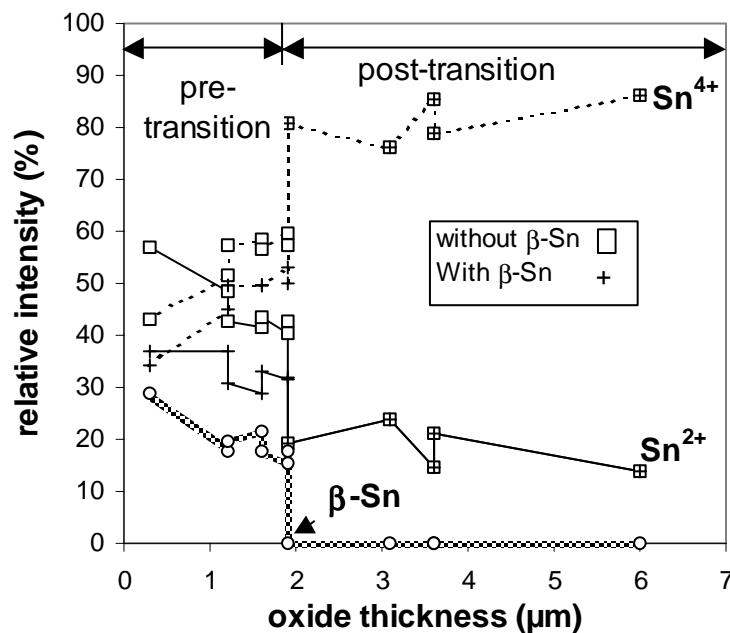
Figure 3-2 Sn^{4+}, Sn^{2+} and β-Sn relative concentration (a) and intensity (b) versus the oxide film thickness, calculated on spectra fitted with and without assuming β-Sn lines. Results based on spectra taken : in CEMS mode (surface characterisation). Ref. 3-2.

4 MECHANICAL PROPERTIES
P. Rudling

4.1 INTRODUCTION

The mechanical properties of essentially two different components are treated in this section. Firstly, it is the *LWR* fuel assembly and, secondly, it is *Pressure tubes*, e.g., in *CANDU* reactors. The difference between these two components is that the fuel is reloaded after some time in-reactor while the *Pressure tube* is a part of the reactor design and must consequently performance satisfactory during the lifetime of the reactor.

Delayed hydride cracking, DHC, is a failure mechanism that may limit the lifetime of *CANDU* pressure tubes, and this mechanism is therefore treated in the pressure tube section.

The mechanical properties of the *LWR* fuel assembly is crucial for its satisfactory performance in-reactor. *Standard Review Plan, SRP,* section 4.2, lists different mechanical failure modes of the *LWR* fuel components and also the corresponding design criterion to ensure that the fuel assembly behaviour is satisfactory. These design criteria are set to ensure that:

- the fuel assembly will not *fail* during normal operation (class I) and anticipated operational occurrences (class II). *Failing* in this sense has a broader meaning, namely that the fuel rod 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 BWR fuel 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 an accident (class III and IV). Class IV design basis accident are *LOCA, RIA* and earthquake. 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.

During class I and II operation, the following mechanical *failure* mechanisms and corresponding design criterion for the fuel assembly, including its components, are listed in *SRP* section 4.2:

- Plastic deformation – the component is regarded as *failed* if it is plastically deformed and the appropriate criterion is that the stresses must be lower than the yield stress. *SRP* section 4.2 also state what type of methodology should be used when calculating these stresses. In these calculations the stress in the assembly location subjected to maximum stresses is calculated. In calculating this stress, all types of stresses are taking into account, such as welding residual stress, thermal stress, stress imposed by rod-system differential pressure, etc. It is interesting to note that the criterion on maximum allowable oxide thickness on fuel rods is related to this criterion. If the oxide thickness becomes too large in a *PWR*, the oxide thickness will increase the cladding temperature due to its lower thermal conductivity and would then increase corrosion rate. The oxide thickness would increase further, raising the clad temperature and corrosion rate, resulting in thermal feedback. Since increasing temperature decreases the yield strength of the material, the material would eventually mechanical fail, i.e. plastically deform, provided that the cladding stresses are large enough.

- Excessive creep deformation that could either result in creep fracture or too large plastic deformations that could e.g. lead to *dryout* due to excessive outward creep of the fuel cladding diameter that would limit coolant flow. Creep occurs at a stress level lower than the yield stress. The corresponding criterion is very general and just specify that the creep deformation must be limited.

- Fatigue failure – Most fuel assembly components are subjected to fatigue stresses and *SRP* section 4.2 provides the maximum allowable fatigue stress level.

- *PCI* – The criterion to eliminate this type of failure is by limiting the elastic and uniform plastic deformation in the cladding circumference during a class I and II transient to 1%. This value is of course not sufficiently to ensure that *PCI* failures do not occur. However, the fuel vendors are still designing their fuel so this 1 % limit is achieved in their design.

- Hydride embrittlement – The criterion just mentions that the hydrogen content in the material must be limited so the fuel assembly component will not fail.

During accident conditions such as *LOCA* and *RIA*, the mechanical performance of the fuel cladding is crucial to meet the objective that the fuel must remain coolable during these types of accidents. In both situations, it is important that the fuel cladding may not fail in a brittle fashion during the *reflooding*[3] phase during *LOCA* and due to *PCMI* during the *RIA* transient.

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[3] 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

Fuel vendors have developed codes to model the fuel assembly mechanical performance during class I, II, III and IV situations and to be able to do the modeling correctly, data on mechanical performance of the fuel assembly must exist. The data are generated in two types of tests, either separate effect tests and integral tests. The former test studies only the impact of one parameter at a time on the mechanical performance. This could e.g., be the impact of hydrogen content on ductility. The data from these separate effect tests are then used by the fuel vendor to develop adequate models in their fuel performance codes. To then verify that the code comes up with the correct prediction on fuel assembly mechanical performance e.g. during a *LOCA*, the code predictions are benchmarked towards integral tests. In the integral test, the fuel assembly design and environment is as similar as possible as is existing in the situation that is simulated in the test, e.g. a *LOCA*.

Delayed Hydride Cracking, DHC, is a fracture mechanism that may result in CANDU pressure tube failures as well as degradation of failed LWR fuel. A better understanding of the relation of the DHC mechanism to the material properties may e.g. assist the fuel vendors to develop products with enhanced resistance towards DHC.

4.2   NEW RESULTS

4.2.1   Pressure tubes

In a CANDU reactor the cold-worked Zr-2.5Nb pressure tubes operate at temperatures between about 250 and $3\tilde{1}°C$, and at coolant pressures of about 10 MPa corresponding to hoop stresses of about 130 MPa. The maximum flux of fast neutrons from the fuel is about $4\cdot10^{17}$ $n\cdot m^{-2}\cdot s^{-1}$.

The pressure tubes used in a CANDU reactor are made from Zr-2.5Nb. The tubes are extruded at 815°C, cold worked 27%, and stress relieved at 400°C for 24 hours, resulting in a structure consisting of elongated grains of hexagonal-close-packed (hcp) α-Zr, partially surrounded by a thin network of filaments of body-centred-cubic β-Zr. These β-Zr filaments are metastable and initially contain about 20% Nb. The stress-relief treatment results in partial decomposition of the β-Zr filaments with the formation of hexagonal-close-packed ω-phase particles that are low in Nb, surrounded by an Nb-enriched b-Zr matrix. The hcp α-Zr grains are oriented with their unique c-axes aligned in the radial-transverse plane, mostly tilted towards the transverse direction.

4.2.1.1   DHC Tests

Critical temperatures for Delayed Hydride Cracking (DHC) initiation during cooling (T_{RIT}) and crack arrest during heating (T_{DAT}) were studied by Schofield et al., 2001. Prehydrided specimens from a Zircaloy-2 electron beam weld irradiated to a fluence of 3 – 5 x 10^{25} $n\cdot m^{-2}$ (E>1MeV) were used in this study that is more described in the following.

5   DIMENSIONAL INSTABILITY
R. B. Adamson

5.1   BACKGROUND

One of the most unique aspects of material behaviour in a nuclear power plant is the effect of radiation (mainly neutrons) on the dimensional stability of the reactor components. In fast breeder reactors the Fe and Ni-based alloys creep and swell, that is, they change dimensions in response to a stress and change their volume in response to radiation damage. In light water reactors, zirconium alloy structural components creep, do not swell, but do change their dimensions through the well-known constant volume process called irradiation growth. Radiation effects are not unexpected since during the lifetime of a typical component every atom is displaced from its normal lattice position at least 20 times! With the possible exception of elastic properties like Young's Modulus, the properties needed for reliable fuel assembly performance are affected by irradiation. A straightforward summary of such effects is given in [Ref. 5-1].

Practical effects of dimensional instabilities are well known and it is a rare technical conference in the reactor performance field that does not include discussions on the topic. Because of the difference in pressure inside and outside the fuel rod, cladding creeps down on the fuel early in life, and then creeps out again later in life as the fuel begins to swell. A major issue is to have creep strength sufficient to resist outward movement of the cladding if fission gas pressure becomes high at high burnups. PWR guide tubes can creep downward or laterally due to forces imposed by fuel assembly hold down forces or cross flow hydraulic forces – both leading to assembly bow which can interfere with smooth control rod motion. BWR channels can creep out or budge in response to differential water pressures across the channel wall, again leading toward control blade interference. Fuel rods, water rods or boxes, guide tubes, and tie rods can lengthen, possibly leading to bowing problems. (For calibration, a recrystallized (RX) Zircaloy water rod or guide tube could lengthen due to irradiation growth more than 2 cm. during service; a cold worked/stress relieved (SRA) component could lengthen more than 6 cm.) Even RX spacer/grids could widen enough due to irradiation growth (if texture or heat treatment was not optimized) to cause uncomfortable interference with the channel.

In addition, corrosion leading to hydrogen absorption in Zircaloy can contribute to component dimensional instability due to the fact that the volume of zirconium hydride is about 16% larger than zirconium.

The above discussion leads to the concept that understanding the mechanisms of dimensional instability in the aggressive environment of the nuclear core is important for more than just academic reasons. Reliability of materials and structure performance can depend on such understanding.

A review of creep and growth fundamentals is given in Appendix 5A. That review gives the background against which to compare recent results from the open literature.

5.2   YEAR 2000 HIGHLIGHTS

Last year's ZIRAT-5 review highlighted several developments. Irradiation growth of RX Zircaloy-2 at BWR temperature was shown to remain linier with fluence well above projected fuel exposure limits. It was also confirmed that the microstructure evolution, particularly c-dislocation generation, was consistent with the observed linier growth rate. It was also confirmed that for RX Zircaloy and M5, growth decreases and creep increases with increasing temperature in the range 553-623K (280-350ºC), while for Zr2.5Nb growth decreases with increasing temperature in the range 563-583K (290-310ºC).

Many new PWR alloys were introduced in the past few years that appear to have increased corrosion resistance and decreased creep and growth, at least out to a burnup near 50 MWd/kg/U. None of the alloys seemed clearly superior to the others, but M5 and the Russian Alloy 635 appear to be the most thoroughly tested.

It was convincingly shown that hydrogen and hydrides affect dimensional stability of all reactor components, so that decreasing corrosion would, for instance, decrease fuel rod growth. Other ways suggested to decrease rod or guide tube elongation included using Zircaloy-4 with "high Sn", using beta-quenched Zircaloy or using M5 or Zircaloy with high sulfur.

5.3   COMPONENT GROWTH

5.3.1   BWR Channels

Channel performance depends on resistance to corrosion, hydriding, elongation (irradiation growth), bulging (creep) and bow (non-uniform growth and creep). Dimensional stability is also a function of fabrication technique, mainly through management or reduction of residual stresses. As with PWR guide tubes, growth and creep can be reduced by heat treatment: last stage beta-quenching produces random texture and small growth, and recrystallized material has benefits for relatively low burnup. Reduction of corrosion and the subsequent reduction of hydriding reduces length change (see section 5.3.3).

Westinghouse Atom presented data [Ref. 5-2] indicating the advantages of using Zircaloy-2 instead of Zircaloy-4 for channels. Fig. 5-1 shows Zircaloy-2 having lower corrosion than Zircaloy-4. Dahlback et al. [Ref. 5-2] make the reasonable claim that this lowers the hydrogen content of channels, even though the hydrogen pickup fraction of Zircaloy-2 is usually higher than Zircaloy-4. Fig. 5-2 shows Zircaloy-2 having lower elongation than Zircaloy-4. Since it is generally thought that irradiation growths of the two Zircaloys are equivalent, the lower growth is attributed to lower hydrogen.
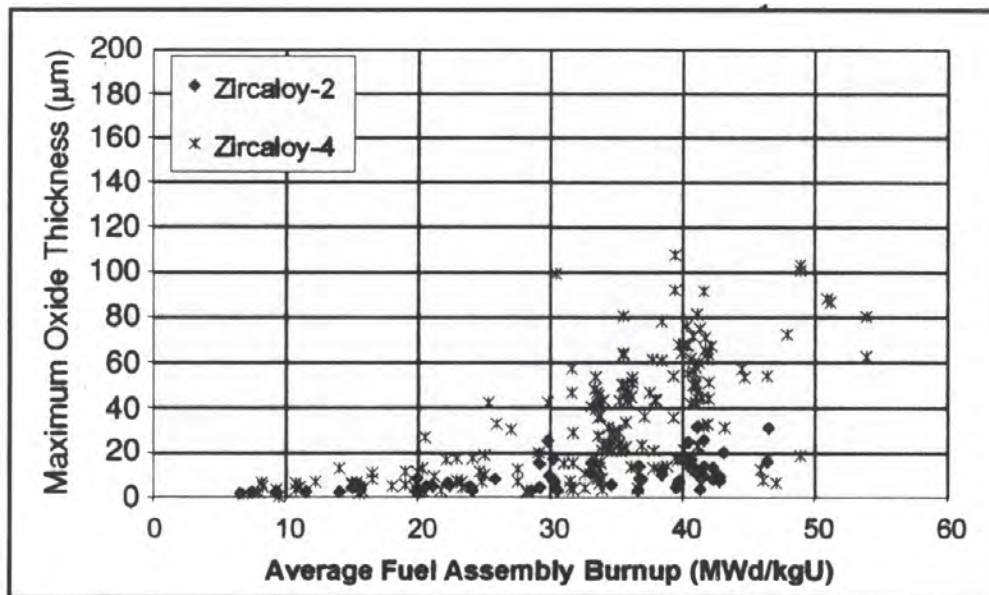
Figure 5-1. **Measured oxide layer thickness versus burnup for Zircaloy-4 and Zircaloy-2 fuel channels.**
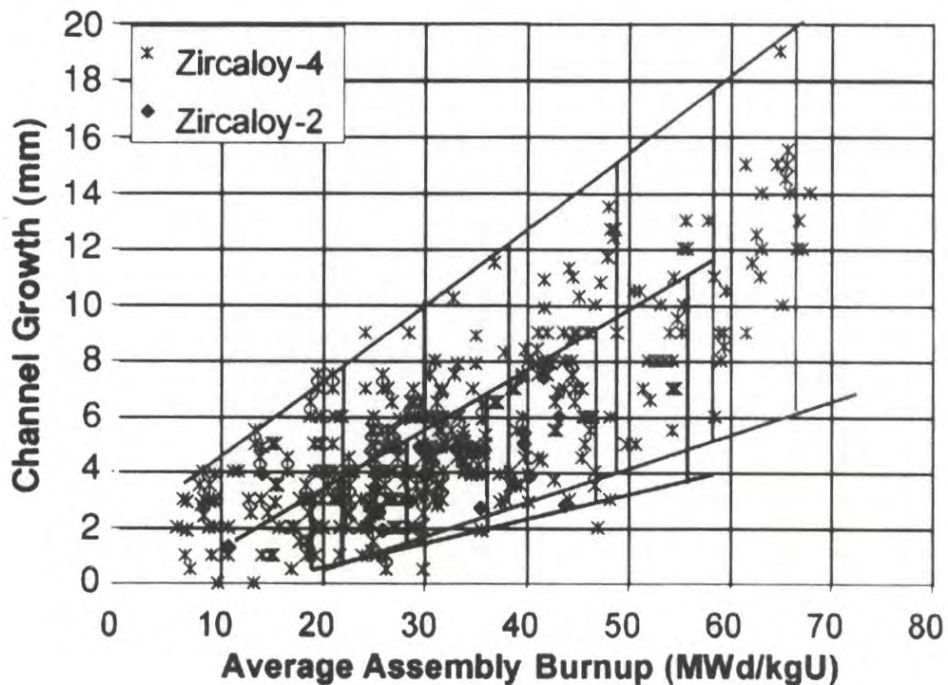
Figure 5-2. **Measured channel growth versus burnup for Zircaloy-4 and Zircaloy-2**

6 CORROSION AND HYDROGEN UPTAKE
B. Cox

6.1 INTRODUCTION

In past reviews the importance of understanding the details of the morphology of the oxide film on the zirconium alloy surface to any understanding of the corrosion and hydrogen uptake behaviour has been regularly emphasized. Despite many years of studying oxide films by a wide range of techniques, there remain a number of features of these films which are difficult to establish with any degree of certainty. We will summarise first these areas of ignorance, before proceeding to examine publications over the past 12 months for any evidence of an improved understanding.

1) The thin oxide film that forms initially on all zirconium alloy surfaces (≤2-3 μm thick) is thought to be a barrier to the migration of molecular species to the oxide/metal interface. Thus, oxygen that reacts to form fresh unit cells of ZrO_2 at this interface and hydrogen (released by the decomposition of water molecules at the oxide/environment interface) would need to diffuse through this barrier in some ionic or atomic form. That this is essentially true for oxygen ions has been well established from measurements of the oxidation kinetics and oxygen ion diffusion rates [Ref. 6-1, Ref. 6-2]. But is this also true for hydrogen species?. Hydrogen absorption rates on bare (i.e. completely oxide free) zirconium alloy surfaces are so rapid that only a few small flaws in the "barrier" oxide would be needed to allow the rates of hydrogen absorption seen during corrosion of zirconium alloys [Ref. 6-3, Ref. 6-4]. Do such flaws exist even in the so-called barrier oxides?. The answer seems to be probably yes, and these flaws may be associated with second phase particles that intersect the initial surfaces of the material, or with sharp changes in surface topography [Ref. 6-5, Ref. 6-6]. Fuel cladding surfaces, however, are usually belt-ground - a surface finish that makes flaw location difficult.

2) The protectiveness of this initial oxide film deteriorates once its thickness exceeds ~2μm. In order to identify the flaws that develop it is necessary to do very sophisticated electron microscopy using Fresnel Contrast to identify the smallest cracks and pores. In order to do such studies very thin sections of oxide films must be prepared. This involves successively cutting, grinding and ionmilling the oxides to be examined. Can we be sure that artefacts in the form of small cracks are not produced by these techniques?. The answer to this has to be - No! Nevertheless, very little has been done to establish the extent of such artefact production. Metallographic sections of oxide films show what appears to be layers of cracks often spaced   ~2μm apart in Zircaloys and other alloys. These are visible even in optical micrographs, and when studied by SEM techniques, as well as appearing in TEM micrographs (Figure 6-1) . Were these generated by the stresses applied during the cutting and grinding of the sections. In early study [Ref. 6-7] grinding motions parallel to the oxide/metal interface were applied to one set of specimens, while grinding perpendicular to the oxide/metal interface was applied to another set. Electron microscope replicas of the sections showed an absence of actual cracks in the former, but layers of cracks were present in the latter, and were created in the former when specimens were removed from their mounts and the oxides were fractured (Fig. 6.2). Thus although there are clearly planes of weakness in these oxide films, it seems unlikely that actual lateral cracks are present until the onset of oxide blistering and spalling ensues in very thick oxides [Ref. 6-8]. However, much mileage has been made from poor quality SEM images of sections exposed to uncontrolled grinding [Ref. 6-9], and little or no attempt has been made to establish the presence of cracks running in the radial direction using Fresnel Contrast. Obviously, cracks in this direction are the most important for understanding accelerating corrosion and hydrogen uptake processes. However, the idea that many investigators do not want to find such features, and, therefore, do not do the careful experiments that might reveal them, seems to be one possible conclusion. That such radial cracks must exist in post-transition oxides has long been deduced from pressure change experiments [Ref. 6-1, Ref. 6-2].
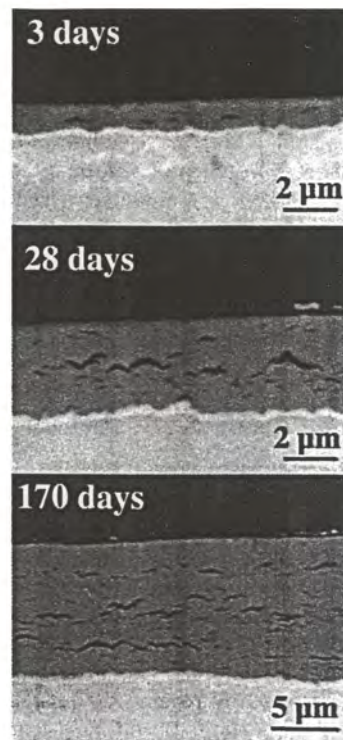
Figure 6-1:   SEM micrographs, on cross sections, in backscattered electron mode.

(a)   **(b)**
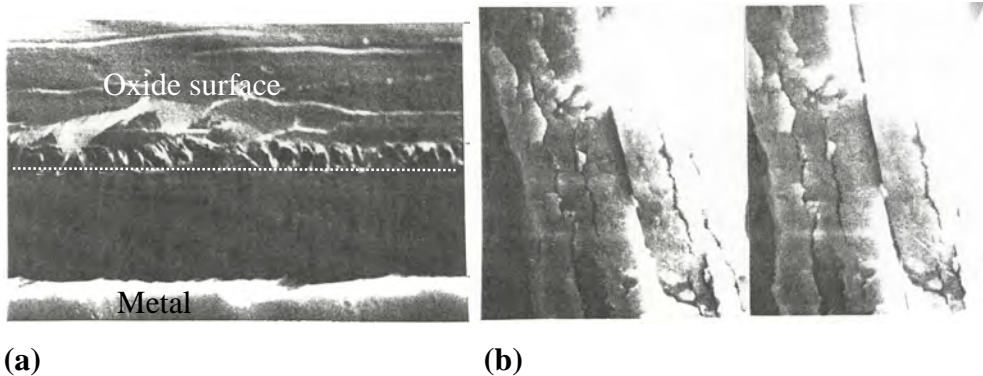
Figure 6-2   (a) SEM micrographs of 8 microns oxide grown on Zry-2 in 350°C water. Specimen was protected with Ta foil during mounting and polishing. All grinding and polishing done parallel to interface; (b) Similar specimen to (a) removed by mount; metal etched back and unprotected oxide fractured; stereo – pair, both sides of fracture shown. SEM micrographs at 0° and 10°.

7 EFFECTS OF WATER CHEMISTRY
Gunnar Wikmark

7.1 INTRODUCTION

One important issue that was discussed last year was the AOA (Axial Offset Anomaly) problem. The year 2000, 20 cores in the US had been affected, but almost none outside of the US. Since the number of cores affected by AOA was still increasing and had caused at least one plant to downgrade, AOA was becoming of vital importance for PWR operation. The crud phenomenon causing AOA had obviously also been resulting in fuel failures and considerable efforts have been started, mainly within the EPRI Robust fuel program, to understand and mitigate the phenomenon. Last year the EPRI recommendations, to operate with a higher constant pH, by increasing the Li concentration, or to avoid the AOA problem by using enriched boric acid, whereby the Li level could be kept lower, had yet no support in experiment and application.

Zinc additions have gained increasing popularity. In the US, virtually every BWR and an increasing number of PWRs are employing this measure, generally with the purpose to have less activity build-up. The experience is mainly good, with decreased dose rates in the primary system. To date, there are no recorded drawbacks, as long as depleted zinc is used (at a very elevated cost). Difficulties in using the Eddy Current technique to assess the real oxide thickness of cladding when the crud thickness is large have, however, been found.

For a number of years, ESSC (enhanced spacer shadow corrosion) has been a topic worrying many plant fuel managers, since the first and only record of the phenomenon in NPP Leibstadt (KKL) in 1997. Last year, extensive results from KKL were presented and discussed. Iron depleted water chemistry and the small SPP (second phase particle) cladding material were reported to be instrumental in leading to the problems. Both water chemistry and cladding material were changed in KKL, and there has been no additional report of any plant having ESSC to the extent in KKL in 1997, i.e. having local oxide thickness in the spacer region >200 µm.

In BWRs, NMT (noble metal technology,) and then especially NMCA (noble metal chemical application), has been gaining increased popularity in the US as means to improve the efficiency of hydrogen injection for SCC mitigation. It had, however, been experienced in several BWRs that the general water chemistry is also affected. These changes in behavior could, for instance, make the detection of fuel failures more difficult and could also cause water chemistry transients of iron and sulfate. The plant chemists consequently have to adapt to the different behavior of the plant chemistry in order to optimize the fuel failure monitoring as well as to be able to apply activity and SCC reduction programs.

Last year, we pointed out that the current trend towards elevated burn-ups and higher power correlated with an increased number of fuel failures and operation disturbances due to water chemistry and crud effect on fuel cladding. The KKL case above and especially the failures in River Bend, where some more detailed data were discussed last year, are modern examples of the situation. This development leads us to provide a review of the basis for crud formation and the recorded fuel failures and operation disturbances due to water chemistry and crud in a ZIRAT special topics report this year [Ref. 7-1]. This fact and the lack of water chemistry conferences 2001 imply that comparably little new water chemistry and crud information is presented here.

7.2 PWRS

7.2.1 Axial Offset Anomalies (AOA)

The axial offset anomaly (AOA) phenomenon is still causing significant problems for the utilities, and it has been reported that some 15 plants and more than 30 cycles have been affected of the US PWRs [Ref. 7-2]. The cost to avoid AOA has been estimated to be 0.5 – 2.0 MUSD, which is probably a very low estimate considering that one plant had to operate at 70% power for 6 months in one cycle [Ref. 7-2]. It has also been stated that AOA is the main obstacle in increasing PWR efficiency currently in the US. The current understanding of the AOA mechanism is shown schematically in Figure 7-1.
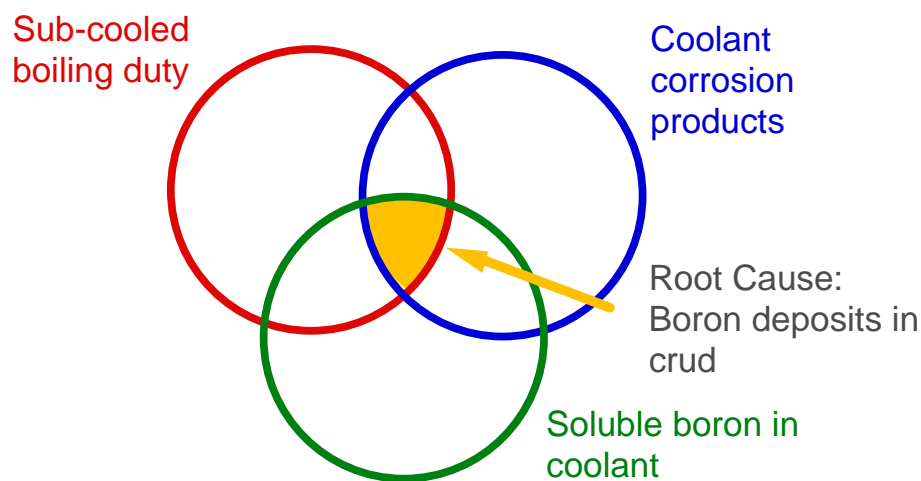
Figure 7-1.   Schematic view of AOA mechanism [after Ref. 7-3]

From this Figure, it should be inferred that three simultaneous factors are needed to cause AOA[6]. It is assumed that the boiling causes abundant crud formation due to high availability of corrosion products as a result of noticeable release from the steam generator tubing mainly. The crud is assumed to be rather porous, allowing trapping of boron to cause the neutron flux deviation. Recently, there has been a hot-cell investigation of crud that supports this model, as shown in Figure 7-2.
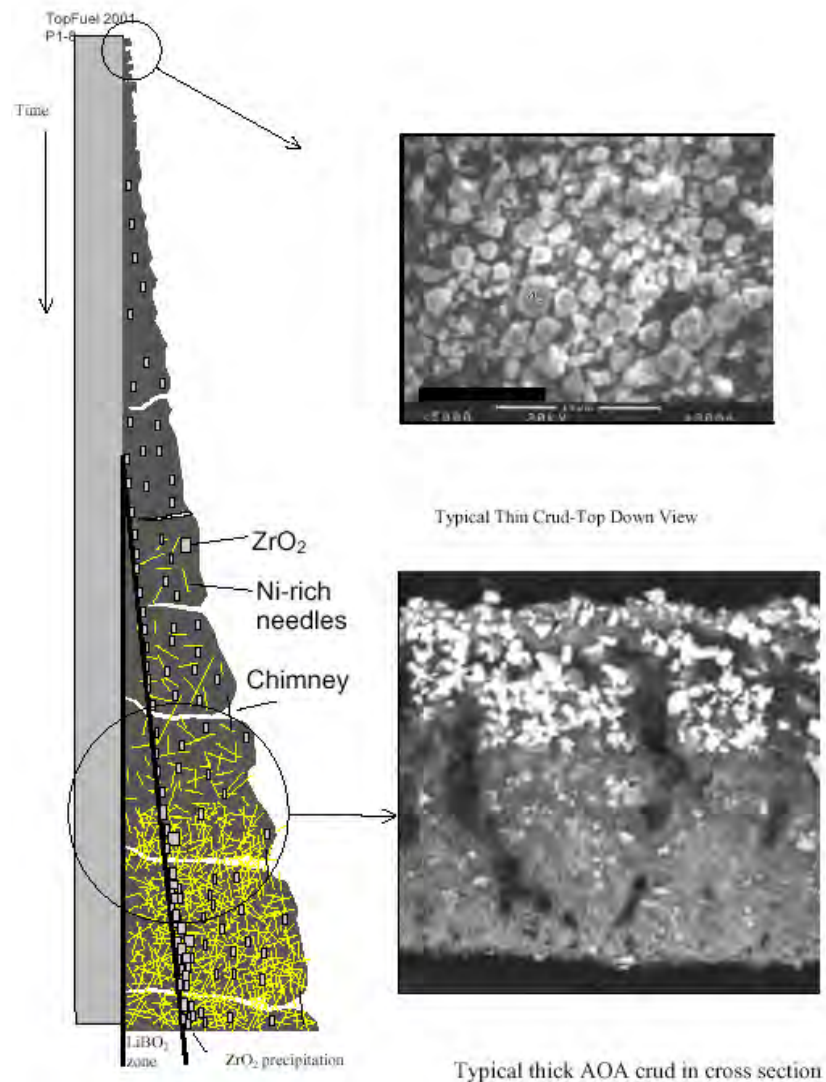
Figure 7-2.   Crud deposits in an AOA plant [Ref. 7-4]

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[6] As pointed out in [Ref. 7-1], this understanding and principal mechanism was reporter already in 1976, based on German experience.

8   ZIRCONIUM ALLOY MANUFACTURING
Peter Rudling

8.1   INTRODUCTION

The development of manufacturing of the zirconium alloys has, during the last years, been characterised by decreased variation in process parameters, to cope with higher demands on the materials; increase efficiency, in order to reduce manufacturing costs; and introduction of new materials, again to comply with higher demands on the cladding.

In all these cases, it is important to be aware of the possible influence of the changes on materials properties not in focus for the change of the process or the new alloy introduced. For instance, a change to improve the corrosion properties by alloying with niobium may change the mechanical properties during power ramp, i. e. that the same PCI rules should not apply.

8.2   NEW RESULTS

8.2.1   Zircaloy

During manufacturing, cold pilgering is a process that may result in cladding defects. Generally, larger degree of cold reduction in each pilgering step will result in a more radial texture that is beneficial in PCI and hydride orientation respect. However, to much cold reduction may result in cladding defects. Thus, a knowledge of the deformation processes during the pilgering process may be used to optimise the microstructure in the cladding without introducing defects.

An experimental evaluation of plastic shearing due to cold rolling of Zry-4 cladding tubes were performed Ref. 8-4. The impact of the following parameters on the shear stress were investigated: (a) the feeding, (b) the frequency of rolling steps, and (c) the type of internal lubricant. To analyse the different deformation modes in each grain, strain texture simulation was performed from a VPSC model[7,8]. The authors argue that with this VPSC model it was possible to show that the prismatic mode was the most active slip system but that the pyramidal $<c+a>$ slip and tensile twinning do have a role that varies with the strain degree.

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[7] R. Lebensohn, C. N. Tomé, Acta Metall. Mater. 41, 1993, p. 2611-.
[8] R. Lebensohn, C. N. Tomé, Mater. Sci. Eng. A 175, 1994, p. 71-

8.2.2 Zr-Nb

8.2.2.1 E110

The deformation of Zr1Nb alloy during the pilgering process was studied by Zavodtchikov et al., 2001, Ref. 8-3. The results of local X-ray measurements of residual stresses and ultrasonic speed measurements made it possible to improve the cladding manufacturing process. The obtained results indicate that the total deformation for any cold reduction step should be below 85 % for the Zr1Nb alloy. Also, the amount of cold-work as well as the Q-ratio[9] (ranging from 1.0 to 3.0) should increase from the first to the last cold rolling step.

8.2.2.2 ZIRLO

To optimise cladding properties a high degree of cold-work is necessary during the pilgering process. However, the lower level of formability of Zr-Nb alloys as compared to Zircaloy may result in cladding defects.

Cook et al., Ref. 8-2, reported that the addition of up to 1000 wtppm (optimum content ranging from 600 to 750 wtppm) of chloride (as magnesium chloride) greatly improved the ZIRLO formability during the pilgering steps from 63.6 to 37.25 mm TREX stage

8.2.2.3 Zr2.5Nb pressure tubes

A study was performed to evaluate the possibility to increase the strength of the calandria tubes in a CANDU reactor to reduce the economical consequences of a hypothetical accident, Ref. 8-1. Up to now, the current manufacturing method has been to form a sheet of Zry-2 into a cylinder and then to weld along the length, according to Theaker and Coleman, 2001. However, the authors state that in fixed-end burst tests such tubes always fracture in the weld and thus, eliminating the weld would increase the strength and ductility of the tube. A number of different manufacturing procedures for seamless tubes were investigated in this study.

Seamless calandria tubes with a strong radial texture resulting in a high biaxial strength, Table 8-1, were obtained by external roll-extrusion, ERE, and production internal roll-extrusion, PIRE, Figure 8-1. The mechanical properties of the calandria tubes are shown in Table 8-2.

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[9] The ratio between deformation in the tube thickness to the tube circumferential direction

Table 8-1: **Development of crystallographic texture for manufacturing route 1 (ERE+PIRE), Ref. 8-1.**

Manufacturing Operation	Specimen Location	Resolved Fraction of Basal Pole Normals		
		F_R	F_T	F_L
As extruded Hollow	Mid Wall	0.50	0.41	0.09
As extruded hollow + anneal at 625°C	Mid Wall	0.45	0.42	0.13
After ERE (≈ 75% CW)	Through Wall	0.65	0.30	0.05
After ERE (≈ 75% CW) + anneal at 700°C	Through Wall	0.68	0.28	0.04
After 1 PIRE pass (≈ 80% CW)	Through Wall	0.71	0.23	0.06
After 2 PIRE passes (≈ 86% CW)	Through wall	0.74	0.20	0.06
After 3 PIRE passes (≈ 95% CW)	Through Wall	0.75	0.21	0.04
After 3 PIRE passes + anneal at 750°C + stress relieve at 500°C	Through Wall	0.78	0.17	0.05
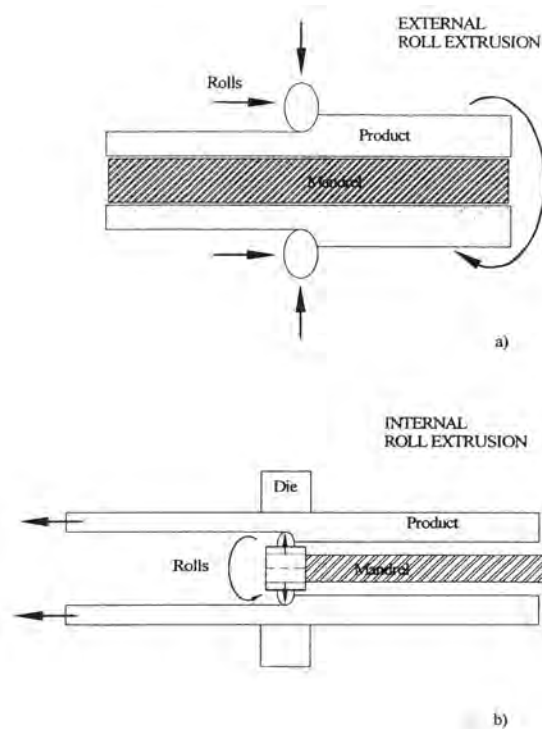
Figure 8-1: **Schematic diagram showing a) external roll extrusion, ERE and b) production internal roll extrusion, PIRE, processes, Ref. 8-1.**

9 PRIMARY FAILURE AND SECONDARY DEGRADATION
P. Rudling

9.1 INTRODUCTION

During reactor operation, the fuel rod may fail due to a primary cause such as fretting, *PCI*, manufacturing defects, corrosion, etc. The more recently experienced primary failures are listed in Table 9-1.

Table 9-1:   Recently Experienced Primary Failure Causes for LWR Fuel, Ref. 9-1.

Primary Failure Cause	Short Description
Debris Fretting	Debris transported by the coolant frets a hole in the cladding. This is the most common primary failure mode in both PWRs and BWRs today.
Excessive Corrosion	An accelerated general or local cladding corrosion causes cladding penetration.
Primary or Reactor Water Chemistry Influenced Corrosion	The water chemistry accelerates the corrosion as found for CILC (Crud Induced Localised Corrosion) and heavy crud deposits. This has by far been most common primary failure mode in BWRs (CILC) until recently.
Dry-out	The coolant flow is too limited, causing the cladding to be overheated. The overheating causes (local) corrosion penetration of the cladding.
PCI	Pellet Clad Interaction, a stress corrosion crack penetrates the cladding. Fission products, mainly iodine, are producing an aggressive chemical environment. The pellet swelling due to a power increase causes a tensile cladding stress.
Grid-Rod Fretting	The spacer grid frets a hole on the cladding (only experienced in PWRs, where it is still the second most common primary failure cause).
Manufacturing Defects	Manufacturing failures can occur in many ways, the most common have been primary hydriding (due to moisture or organic material in the pellets), welding defects around top and bottom end plugs, and cladding tube cracks.

A summary of failures statistics are shown in Table 9-2 and

Table 9-3.

Table 9-2:   BWR Fuel Cladding Failures in the US 1989 – 1999, Ref. 9-1.

Failure Cause	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	Total	
Debris fretting	2	2	17	2	6	4		2	3	5	3	46	*20%*
Grid fretting												0	*0%*
Fabrication failures	3	3	1	1	1	2						11	*5%*
PCI		1			2		2	2	1	1		9	*4%*
Crudding/Corrosion											7	7	*3%*
CILC	52	5	3						3	46		109	*48%*
Unknown/Uninspected		4	3	9	7	9	2	10	1	1	1	47	*21%*
Total	*57*	*15*	*24*	*12*	*16*	*15*	*4*	*14*	*8*	*53*	*11*	*229*	*100%*

Table 9-3:   PWR Fuel Cladding Failures in the US 1989 – 1999, Ref. 9-1.

Failure Cause	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	Total	
Handling failures		6	2			1	1		2			12	*1%*
Debris fretting	146	11	67	20	13	6	10	1	10	3		287	*29%*
Grid fretting	14	18	9	33	36	9	33	52	21	57	5	287	*29%*
Primary hydriding		1		4								5	*1%*
Crudding/Corrosion							4		4			8	*1%*
Cladding creep collapse							1					1	*0%*
Other fabrication failures	1	15	1	5	3	1	15	5			1	47	*5%*
Other hydraulic failures					1							1	*0%*
Unknown					36	36	13	9	10	2	1	107	*11%*
Uninspected	43	58	35	61	14	3	12	3	8		3	240	*24%*
Total	*204*	*109*	*114*	*123*	*103*	*56*	*89*	*70*	*55*	*62*	*10*	*995*	*100%*

A failed LWR fuel rod may degrade either by developing long axial cracks, splits and/or transversal breaks or may not degrade at all. Degradation has historically been more of an issue in BWRs than in PWRs. During the period 1992-1993 six BWR plants in US and in Europe were actually forced into unscheduled outages because of concerns about failed Zr sponge liner fuel degradation leading to excessively high off-gas activities. Both long axial cracks and significant loss of fuel pellet material were observed. In all these cases, the very high off-gas activities resulted from only one or two failed rods. Such a situation may result in very large utility costs. Failed rods in PWRs may also degrade, but the activity release rate is much lower than that in a BWR. The rationale for the less severe behaviour in PWRs is probably related to that the smoother power regulation in a PWR compared to that in a BWR. Also the situation that the coolant in a PWR is more reducing will decrease the tendency for fuel washout.

The secondary degradation scenario is described as follows. Once a fuel rod is perforated, water and steam can enter. The penetration of the fuel clad will subsequently allow water to enter the gap between the cladding and the fuel pellet stack, since the fuel rod pressure normally is much lower than the reactor system pressure. The water vaporises and hydrogen is mainly produced through the oxidation of the fuel cladding inner surface. However, two additional reactions may also contribute to the hydrogen production, namely:

• fuel oxidation and,

• radiolysis of the steam.

The oxide formed at the clad inner surface is uniform and its thickness decreases with increasing distance from the primary defect. This oxide is crucial for resistance against localised hydrogen pickup. Also, due to steam consumption the hydrogen to steam partial pressure will increase with increasing distance from the primary defect. Provided that, at a certain distance from the primary defect, the following conditions are met:

• the hydrogen to steam partial pressure ratio is above a critical ratio and

• the clad inner surface oxide thickness is smaller than a critical thickness,

localised secondary hydriding may occur forming *hydride blisters* or *sunbursts*. Since the specific volume of the hydride is larger than that of the zirconium alloy, a large local stress field will build up in and just outside the *hydride blister.* Also, since the hydride is brittle, many sharp cracks may form within the *hydride blister*. These sharp cracks or other cracks such as *PCI* defects or manufacturing cladding defects may propagate during a large enough power ramp and form an axial split.

If all the secondary hydrides concentrate to only one axial elevation, the massive hydrides may grow throughout the cladding thickness along the cladding whole circumference. This hydride location will be very brittle and easily fracture during reactor operation and it appears that a transversal break in this location may result even without a power ramp. It may be that the stresses generated by the hydrides, due to its larger specific volume, are sufficient to cause a secondary failure. It is not known today what parameters impacts the tendency for hydriding only at a single axial elevation.

10   CLADDING PERFORMANCE UNDER ACCIDENT CONDITIONS
Peter Rudling

10.1   INTRODUCTION

The objectives of the Loss Of Coolant Accident (LOCA) criteria are to maintain core coolability and preserve heat transfer area and coolant flow geometry during the quench phase and post-quench phase of a LOCA.

The existing LOCA criteria were established in 1973 for Zircaloy-clad fuel. These criteria were based on the oxidation kinetics and the post-quench residual ductility of unirradiated cladding material (Zircaloy-2 and -4).

Figure 10-1 is a schematics showing the microstructural changes in Zry-4 containing hydrogen from the $\alpha+\beta$ phase and during the cooling down phase down to room temperature.

(1) After long term annealing at temperature T_0:
(~equilibrium state)

α
enriched with Sn+ O

β
enriched with H, Fe and Cr

$T_{\alpha/\alpha+\beta} < \underline{T_0} < T_{\alpha+\beta/\beta}$:

⇨ *All elements (H, O, Sn, Fe, Cr) are in solid solution within the α and/or β phase*

C O O L I N G

(2) Beginning of cooling (T_1):

α
enriched with Sn+ O

α α α α α α
(*) (*)
β

$T_{\alpha/\alpha+\beta} < \underline{T_1} < T_0 < T_{\alpha+\beta/\beta}$:

(*) *β-stabilizing elements (H , Fe and Cr) are diffusing into the non transformed β phase. Then, the first coarse α laths to form are depleted in H, Fe and Cr.*

Coarse α laths depleted in Fe, Cr and H

(3) End of cooling :

« primary » α
enriched with Sn and O

$\underline{T_2} < T_{\alpha/\alpha+\beta}$:

α zones enriched with Fe, Cr and H

⇨ *The last β zones transformed into α upon cooling are enriched with Fe, Cr and H. Then, heavy Zr(Fe,Cr)₂ and hydrides precipitation occurs in these zones.*

Figure 10-1: **Scheme of the microstructural evolution and of the associated alloying elements partition in Zry-4 containing hydrogen, during cooling down from the $\alpha+\beta$ temperature range down to room temperature, Ref. 10-10.**

During the LOCA event, the cladding is primarily embrittled by the formation of ZrO_2 and oxygen stabilized α Zr(O) as well as by hydrogen pickup during irradiation and high temperature steam oxidation. Figure 10-2 is a schematic representation of the oxygen distribution across the cladding, and Figure 10-3 shows the results of a test along with a metallographic view of the phases. It is the region of the cladding wall that becomes the β-phase and is later transformed to α during the quench that is thought to be the principal load-bearing material. The prior β-phase layer imparts fracture resistance to the cladding and can prevent fragmentation during reflood. The prior β-phase region has relatively low oxygen content.

Figure 10-2: Oxygen Distribution Across Cladding Wall, Ref. 10-2.

Table 11-1
U.S. Utilities with On-site Dry Storage *(Ref. NEI)*
1/86 – 2/14/01, 228 Casks Loaded

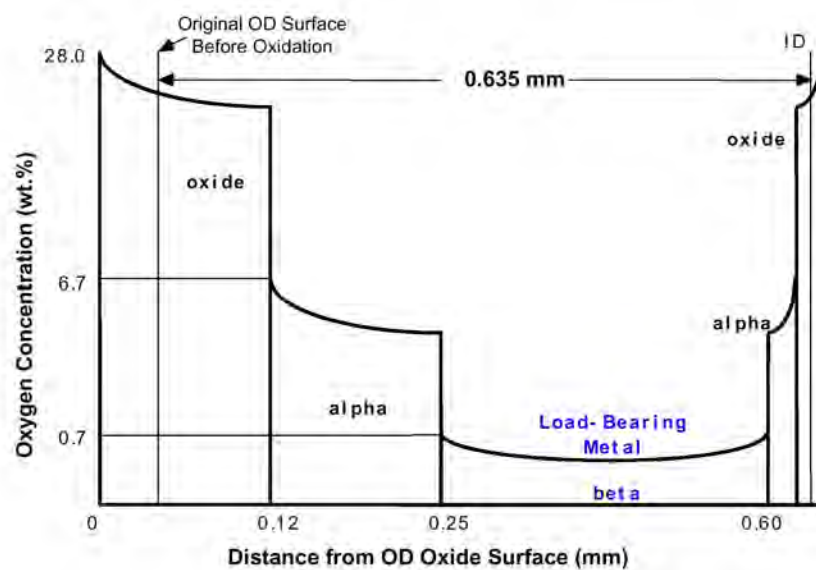| UTILITY	REACTOR	DRY STORAGE TECHNOLOGY	LICENSING METHOD	CASKS LOADED
Dominion (VAPower)	Surry 1 & 2	CASTOR V21 MC10 NAC I28 CASTOR X TN-32	Site Specific	40
Dominion (VA Power)	North Anna 1 & 2	TN-32	Site Specific	9
Entergy	ANO 1 & 2	VSC-24	General License	17
Progress (CP& L)	H.B. Robinson 2	NUHOMS-07P	Site Specific	8
Duke Energy	Oconee 1, 2, & 3	NUHOMS-24P	Site Specific	40
Duke Energy	Oconee 1, 2, & 3	NUHOMS-24P	General License	17
Duke Energy	McGuire 1 & 2	TN-32	General License	1
DOE	Fort St. Vrain	Foster Wheeler MVDS	Site Specific	244[1] (vault storage)
Constellation Energy Group (BG&E)	Calvert Cliffs 1 & 2	NUHOMS-24P	Site Specific	36
Consumers Energy	Palisades	VSC-24	General License	18
Northern States Power Company	Prairie Island 1 & 2	TN-40	Site Specific	12
DOE INEEL	TMI-2 DAMAGED CORE	NUHOMS	Site Specific	1
Wisconsin Electric	Point Beach 1 & 2	VSC-24	General License	12
First Energy	Davis Besse 1	NUHOMS-24P	General License	3
Southern Nuclear	Plant Hatch 1 & 2	Holtec HiStar 100	General License	3
Exelon	Dresden 1	Holtec HiStar 100	General License	1
Exelon	Peach Bottom 2	TN 68	General License	4
PPL	Susquehanna 1 & 2	NUHOMS-52B	General License	8
			TOTAL LOADED CASKS	228

_____
1 not included in cask totals.

An interim dry storage facility has been proposed by the Goshute Indian Tribe on its lands in desolate Skull Valley, Utah. A group of utilities lead by Northern States Power Co. formed Private Fuel Storage LLC (PFS), signed an agreement with the Tribe in 1996 and applied for an NRC license in 1997. The NRC issued a preliminary report in Dec. '99 evaluating PFS compliance with most of the safety related regulations and in Sept. '00 evaluating the remaining safety requirements. The NRC released the Draft Environmental Impact Statement for comment in June '00, the Atomic Safety and Licensing Board (ASLB) held its first Public Hearing (the second is planned at the end of '01) and the NRC is expected to issue the final EIS in August '01. The ASLB and NRC decisions on licensing the facility are expected in the spring of '02 --- encouraging progress.

Meanwhile, however, the Governor of Utah, in strong opposition, has passed some punitive laws to stop the project. As an example, anyone assisting the Tribe on this project in Utah will be considered to have committed a criminal act! These laws have been appealed to a Federal Court on the grounds that they are unconstitutional.

The status of USNRC licensing criteria for dry storage and experience with the stored fuel itself were reviewed and evaluated in ZIRAT-5. This Section represents an update of events since last year and the status of the criteria as of late 2001 and specifically to:

- Summarize the latest US licensing criteria that apply to the fuel and discuss potential modifications to the criteria,

- Discuss current differences in opinion between the NRC and industry on the appropriate criteria and the data that support the various opinions,

- Summarize R&D programs to provide additional data,

- Discuss any new information on fuel performance in dry storage.

Applicable regulatory documents and literature published in the past year were reviewed.

Valuable input to this report were provided by the US Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratories (PNL), the Electric Power Research Institute (EPRI), ANATECH, and Electricité de France (EdF).

11.2 STATUS OF REGULATORY REQUIREMENTS AND SCHEDULE FOR THEIR POTENTIAL MODIFICATION

The status and the requirements of licensing documents has not changed since last year (2000). The basic document, *"Standard Review Plan for Dry Cask Storage Systems" (NUREG-1536)* is not changed. The revised version of Section 4, "Thermal Evaluation", that is critical to fuel licensing, has not been issued for comment as originally scheduled in the fall of 2000 and as of September 2001 has apparently still not been completed.

The Interim Staff Guidance Document, *"Transportation and Storage of Spent Fuel Having Burnups in Excess of 45 GWD/MTU", ISG 11, Rev. 1,* still stands, although work is proceeding toward its modification. The essential parts of the ISG are quoted here for reference:

"For spent fuel having average assembly burnups <45 GWD/MTU, there is sufficient experimental data to support the long-term and short-term temperature limits that are specified in the SRPs. Thus the staff has reasonable assurance that spent fuels with average assembly burnups up to 45 GWd/MTU can be transported and stored safely."

"A high burnup fuel assembly (>45 GWd/MTU) containing Zircaloy clad fuel may be treated as intact if both of the following conditions are met:

A1.   No more than 1% of the rods in the assembly have peak cladding oxide thicknesses >80μm.

A2.   No more than 3% of the rods in the assembly have peak cladding oxide thicknesses >70μm

A high burnup fuel assembly should be treated as potentially failed fuel if either of the following conditions are met:

B1.   The fuel assembly does not meet both criteria A1 and A2; or

B2.   The fuel rods with oxide that has become detached or spalled from the cladding."

"--- licensee may use cladding oxidation thickness measurements or predictions based on consideration of reactor operation variables affecting peak cladding oxidation --- In cases where there are no previously documented measurements of the oxide thickness to validate --- predictions, the program may have to incorporate peak cladding oxide measurements."

"For the criticality, thermal and shielding analyses, the applicant should demonstrate that 10CFR71 and 72 requirements are met assuming that the rods with oxide thickness of >80μ in a high burnup fuel assembly are failed (e.g., the fuel is allowed to redistribute in the cask) under normal, off-normal and accident conditions."

"For transportation --- assume in the containment analysis, that the source term for 50% of the rods with peak cladding oxide thicknesses >70μ is available for release in the cask unless justification for a different fraction is presented."

"For fuel --- meeting criteria A1 and A2, the applicant should employ an acceptable methodology (e.g. CSFM methodology) for calculating cladding temperature limits using a 1% creep strain limit."

"Fuel assemblies that meet criteria B1 and B2 should be handled in accordance with ISG-1 "Damaged Fuel". Alternatively, these assemblies may be treated as intact fuel" --- *(options are given).*

The oxidation and spalling limits are of greatest limitation and concern to utilities, especially as they are unduly conservative as discussed later. Double containment is costly: all casks (except HomeTech) have a significantly lower capacity for canned than uncanned fuel and the extra can cost several thousand $ each. Discussions and disagreements between regulators and industry related to these criteria are discussed in Sections 11.4 and 11.5.

12 FUEL RELATED CRITERIA AND FUEL PERFORMANCE DURING WET STORAGE
A. A. Strasser

12.1 INTRODUCTION

The use of water filled pools to store, shield and cool irradiated fuels and materials is an established, worldwide practice since the beginning of nuclear power. Fuel assemblies discharged from the reactor core are moved underwater through a transfer tunnel to a storage pool. The spent fuel pool (SFP), or "wet" storage facility is adjacent to the reactor. Included in the SFP are permanently discharged assemblies, partially burned assemblies that may be returned to the core, and fresh assemblies for future reloads. Control rods and BWR channels are stored there as well. The SFP must also accommodate a full core discharge in case of an emergency or accident.

The reprocessing plants in France and England have large storage pools; however these were not planned for long term storage, but rather for a high rate of throughput.

The demise of reprocessing in some countries and the reduced amount of reprocessing in others is filling the SFPs at the power plants to capacity and additional storage facilities have been designed, licensed and built. These are Away from the Reactor containment and include storage on the Reactor Site, or AFR(RS) or Off the Site AFR(OS). Dry storage is favored for most of these additional sites, although some wet storage sites exist and are planned.

The lack of reprocessing in the US combined with the lack of a final disposal capability is forcing the utilities to store the fuels for longer periods of time. The trend for increasing burnup is also increasing the wet storage times in order to decrease the decay heat and the cladding temperatures that would be reached during subsequent dry storage. While wet storage of fuel has been remarkably successful for over 30 years, the increased storage times that may be required in the future make this a topic of current interest.

The objective of this report is to:

- Summarize the US licensing criteria the fuel has to meet during wet storage,

- Review fuel performance during wet storage, including effects on zirconium alloys and stainless steels,

- Storage and handling of failed fuel,

- Problems and their solutions,

- Past, current and proposed monitoring programs.

The projected cumulative amount of spent fuel generated worldwide by the year 2010 is estimated to be 330,000 tons of which about 115,000 tons will be reprocessed, leaving 215,000 tons for storage. Spent fuel is generated at about 11,000 tons/year.

A total of about 130,000 tons of fuel were stored worldwide at the beginning of 1998. An overview made at that time of the amounts of fuel stored wet and dry by geographic region is given in Table 12-1, storage capacities in Table 12-2, and fuel discharge projections in Table 12-3 [Ref. 12-1]. A detailed table of wet storage experience worldwide as of 1997 is given in Table 12-4 [Ref. 12-2]. A more up-to-date summary of AR wet storage capacities and inventories as of 1999 is given in Table 12-5 [Ref. 12-3].

Table 12-1
Table Status of Spent Fuel Stored at Beginning of 1998 (Ref. 12-1)

[kt HM]

Regions	AR	AFR Wet	AFR Dry	Total
West Europe	13.9	19.3	1.0	34.2
Asia & Africa	11.6	0.2	0.7	12.5
East Europe	7.8	9.9	0.3	18.0
North & South America	59.8	1.5	3.3	64.6
World	93.1	30.9	5.3	129.3

Table 12-2
Spent Fuel Storage Capacity at Beginning of 1998 (Ref. 12-1)

[kt HM]

Regions	In operation				Under construction		
	at NPP	Wet	Dry	Total	Wet	Dry	Total
West Europe	26.1	31.7	9.2	67.0		0.8	0.8
Asia & Africa	20.0	1.9	0.7	22.6	0.7	0.8	1.5
East Europe	14.3	19.6	0.8	34.7	0.8*	1.6	2.4
North & South America	94.9	1.8	10.0	106.7		6.8	6.8
World	155.3	55.0	20.7	231.0	1.5	10.0	11.5

* by reracking AFR storage capacity

Table 12-3
Projections of Spent Fuel Stored (*Ref. 12-1*)

[kt HM]

Regions	1997	2005	2010	2015
West Europe	34.2	40.1	38.9	36.4
Asia & Africa	12.5	27.6	38.6	50.2
East Europe	18.0	31.1	39.4	47.9
North & South America	64.6	91.3	108.4	125.9
World	129.3	190.1	225.3	260.4

13 POTENTIAL BURNUP LIMITATIONS

13.1 INTRODUCTION

The potential fuel assembly burnup limitations related to zirconium alloy components are summarised in this Section. The burnup limitation that have actually been reached, but have been or are being extended, are:

1) Corrosion of Zry-4 in high power PWRs, extended by improved cladding alloys. However, improved corrosion performance by the new alloys may allow the utilities to use the added margins, to modify plant operation e.g., to lower fuel cycle cost. This modified operation will in most cases result in heavier corrosion duty of the zirconium materials. Thus, it is believed that the corrosion will always be limiting for plant operation even with the new type of alloys.

2) Bowing of PWR fuel assemblies contributed in part by irradiation growth and hydriding of Zry-4, extended by improved guide tube materials (i.e., lower irradiation growth and hydriding rates), and reduced assembly holddown forces, but not yet finally eliminated. It is questionable, whether design changes to make the fuel assembly stiffer will resolve the issue. It may be that, these design changes will only increase the incubation time before bowing occurs and at that point it will be even more difficult to e.g. insert control rods.

3) Bowing of BWR channels, extended by improved manufacturing processes, design changes such as variable wall channel thickness with relatively thicker corners, and in-core channel management programs,

4) RIA and LOCA related burnup licensing limits, in the process of being assessed by additional experimental data and analyses. Based upon the on-going test program, it is not clear today if the licensing limits will be extended or not. The limits may also actually be reduced.

The categories of event likely to eventually limit reliably and safely achievable burnup levels are outlined below. The zirconium alloy component most sensitive to the limits and potential methods for extending the limits are noted below.

13.2 CORROSION AND MECHANCIAL PROPERTIES RELATED TO OXIDE THICKNESS AND H PICKUP

1) BWRs: increased uniform and shadow corrosion, oxide thickness spalling.--- due to longer residence time, higher power and water chemistry changes. Current crucial issues are: shadow corrosion mechanisms, CRUD-chemistry-corrosion interaction and specific effects of NMCA.

2) PWRs: increased uniform corrosion, oxide thickness, spalling --- due to longer residence time and higher Li, higher power, more boiling. The introduction of Zr-Nb alloys may also result in accelerated corrosion at the welds, e.g., between the end-plugs and cladding tubes. Welding will always introduce a continuous beta-Zr

phase that will significantly decrease corrosion resistance. Also, welding between dissimilar metals such as e.g. ZIRLO and Zry-4 may result in chemical compositions of the welds that show inferior corrosion resistance. Luckily, the corrosion temperatures at these elevations in the core is significantly lower than the peak temperatures and this may be the reason that no corrosion issues have been reported so far in the welds.

3) Increased H pickup tendency[11]:

- decreased ductility and fracture toughness during any situation (e.g., RIA, PCMI,..)

- increased corrosion due to impact of hydrides at the cladding outer surface

Most sensitive component

Spacer and fuel claddings

Increase margin for PWR

- Improved knowledge of corrosion and hydrogen pickup mechanisms,

- Improved alloys with appropriate fabrication processes: ZIRLO/E635 (Anikuloy), M5/Zr1Nb. Duplex is another alternative that may be necessary to achieve satisfactory mechanical properties

- Change to enriched B soluble shim to reduce Li. There is however a fear that enriched B would increase AOA potential, i.e., more absorption per g. B, even though there may be less B.

- Improved water chemistry and CRUD control

- Increase corrosion resistance of steam generator materials

Increase margin for BWR

- Improved knowledge of corrosion and hydrogen pickup mechanisms,

- Improved alloys under development

- Modification of manufacturing processes (to get optimum sized, more stable second phase particles)

- Improved water chemistry and CRUD control

---

[11] due to the introduction of more corrosion resistant materials. For most materials, the hydrogen pickup fraction increases with decreasing corrosion rate.

13.3   DIMENSIONAL STABILITY

- Increased growth of components and differential growth between them resulting in reduced fuel rod spacing or even rod contact, guide tube bowing, fuel assembly bowing, spacer cell and envelope dimensions, BWR fuel channel and PWR fuel assembly bow may result in

    i.   decreased thermal margins (LOCA and dry-out)

Most sensitive component

Potentially all zirconium alloy components, but currently PWR guide tubes. Also BWR spacers have occasionally increased so much in dimensions that unloading of the assembly from the outer channel was very difficult.

Increase margin for PWR

- Alloys with lower growth and hydriding rates for guide tubes – ZIRLO, M5, M4, E635 (Anikuloy)

- Lower hold-down forces

- Beta-quenched material after the last plastic deformation step during manufacturing. Beta-quenched materials do normally, however, show higher corrosion rate and lower ductility. These properties may be improved by an appropriate final heat-treatment in the alpha-phase.

Increase margin for BWR

- Uniform microstructure and texture throughout the whole fuel outer channel

- Beta-quenched material in as-fabricated step

- Channel management programs

- More corrosion resistant material in channels and spacers is needed to allow irradiation for longer cycles

13.4   PCI IN BWRS

- Increased sensitivity for PCI by alloy diffusion from Zry-2 base material into the liner. However, this may not be a concern with the current Fe-alloyed liner.

Most sensitive component

Fuel cladding