

Annual Report

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Acronyms and expressions

ACRS	Advisory Committee on Reactor Safety
ADOPT	Advanced Doped Pellet Technology
AFR	Away-From-Reactor
ANSI	American National Standards Institute
ANSYS	a finite element code
ANT	Advanced Nuclear Technology
AOA	Axial Offset Anomaly
AOO	Anticipated Operational Occurrences
APSR	Axial Power Shaping Rods
AR	Annual Report
ASLB	Atomic Safety and Licensing Board
ASTM	American Society for Testing and Materials
B&W	Babcock & Willcox
BOC	Beginning Of Cycle
BT	Burst Test
BTP	Bottom Tie Plate
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium
CB	Control Blade
CCC	Control Cell Core
CE	Combustion Engineering
CEA	Commissariat à l'Energie Atomique
CFD	Computational Fluid Dynamic
CFM	Cell Friction Metric
CFR	Code of Federal Regulations
CILC	CRUD Induced Localised Corrosion
CIPS	CRUD Induced Power Shifts
CPNPP	Comanche Peak
CPSES	Comanche Peak Steam Electric Station
CRSS	Critical Resolved Shear Stresses
CRUD	Chalk River Unidentified Deposits
CSED	Critical Strain Energy Density
CW	Cold Worked
CWSR	Cold Work and Stress Relieved
DBA	Design Base Accident
DCP	Distinctive <i>CRUD</i> Pattern
DCSS	Dry Cask Storage System
DH	Dissolved Hydrogen
DHC	Delayed Hydride Cracking
DOE	Department Of Energy's
DX	Duplex
DZO	Depleted Zinc Oxide
EC	Eddy Current
ECBE	Effective Control Blade Exposure
ECP	Electrochemical Corrosion Potential
ECR	Equivalent Cladding Reacted
ECT	Eddy Current Testing
EFID	Effective Full Insertion Days
EFFPD	Effective Full Power Days
EIS	Electrochemical Impedance Spectroscopy
ELS	Extra-Low Sn
EMAR	Electromagnetic Acoustic Resonance Technique
EOC	End Of Cycle
EPRI	Electric Power Research Institute

ESSC	Enhanced Spacer Shadow Corrosion
ETC	Effective Thermal Conductivity
FA	Fuel Assembly
FAC	Flow Accelerated Corrosion
FEA	Finite Element Analysis
FF	Fitting Factor
FFS	Fitness For Service
FIVPET	Flow Induced Vibration and Pressure drop Experimental Test
FME	Foreign Materials Exclusion
FRED	Fuel Reliability Data Base
FRP	Fuel Reliability Program
FSAR	Final Safety Analysis Report
FW	FeedWater
GE	General Electric
GNF	Global Nuclear Fuel
GTRF	Grid-to-Rod Fretting
HANA	High performance Alloy for Nuclear Application
HBS	High Burnup Structure
HBWR	Halden BWR
HCP	Hexagonal Close-Packed
HDCI	High Duty Core Index
HPU	Hydrogen PickUp
HPUF	Hydrogen PickUp Fraction
HWC	Hydrogen Water Chemistry
IAEA	International Atomic Energy Agency
ICM	Internal Conical Mandrel
ID	Inner Diameter
IFBA	Integral Fuel Burnable Absorber
IFM	Intermediate Flow Mixing
IGSCC	InterGranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
IPHT	In Process Heat Treated
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance documents
IZNA	Information on Zirconium Alloys
KAERI	Korea Atomic Energy Research Institute
KKG	KernKraftwerk Gundremmingen
KKL	KernKraftwerk Leibstadt
KKM	KernKraftwerk Mühleberg
LA	License Application
LCC	LWR Coolant Chemistry
LEFM	Linear Elastic Fracture Mechanics
LHGR	Linear Heat Generation Rate
LK	Låg corrosion (Low Corrosion in Swedish)
LLNL	Livermore National Laboratory
LOCA	Loss of Coolant Accident
LTA	Lead Test Assemblies
LTP	Low Temperature Process
LUA	Lead Use Assemblies
LWR	Light Water Reactor
LWT	Legal Weight Truck
MDA	Mitsubishi Developed Alloy
MFDI	Modified Fuel Duty Index
MHI	Mitsubishi Heavy Industries
MIMAS	Micronized Master Blend route
MOX	Mixed OXide
MP	Mid-Pellet

MREX	neutron shielding material
MSLRD	Main Steam Line Radiation Dose
NDA	New Developed Alloy
NFIR	Nuclear Fuel Industry Research
NMC	Noble Metal Chemistry
NMCA	Noble Metal Chemical Addition
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUPEC	NUclear Power Engineering Corporation
NWC	Normal Water Chemistry
OCRWM	Office of Civilian Radioactive Waste Management
OD	Outer Diameter
OLNC	On Line Noble Chem
PCI	Pellet Cladding Interaction
PCMI	Pellet Cladding Mechanical Interaction
PCT	Peak Cladding Temperature
PFS	Private Fuel Storage
PGP	Particle Growth Parameter
PIE	Post-Irradiation Examinations
PLT	Pin Loaded Tensile
PQD	Post-Quench Ductility
pRXA	Partially Recrystallized
PSCC	Primary Stress Corrosion Cracking
PSI	Paul Scherrer Institute
PSU	Penn State saknas det inte ett ord?
PWR	Pressurised Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RBMK	Reaktor Bolshoi Mozhnosti Kanalov (in English Large Boiling Water Channel type reactor)
RIA	Reactivity Initiated Accident
RIL	Research Information Letter
RX	Recrystallised
RXA	Recrystallised Annealed
SCC	Stress Corrosion Cracking
SED	Strain Energy Density
SEM	Scanning Electron Microscopy
SFPO	Spent Fuel Projects Office
SPP	Second Phase Particle
SRA	Stress Relieved Annealed
SRP	Standard Review Plan
STAR	Steady state and Transient Activity Release
STEM	Scanning Transmission Electron Microscopy
STP	Standard Temperature and Pressure
STR	Special Topic Report
TAD	Transportation, Aging and Disposal
TCE	Total Circumferential Elongation
TFGR	Transient Fission Gas Release
TMI	Three Mile Island
TSC	Transportable Storage Canister
TSHT	Tube Shell Heat Treated
UFC	Ultrasonic Fuel Cleaning
USNRC	United States Nuclear Regulatory Commission
WALT	Westinghouse Advanced Loop Tester
VEC	Vallecitos Embedded Charpy
VVER	Voda Voda Energo Reactor (Russian type PWR)
XANES	X-ray Absorption Near-Edge Structure
ZIRAT	ZIRconium Alloy Technology
ZIRLO	ZIRconium Low Oxidation

Unit conversion

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

Radioactivity	
1 Sv	= 100 Rem
1 Ci	= 3.7×10^{10} Bq = 37 GBq
1 Bq	= 1 s^{-1}

MASS	
kg	lbs
0.454	1
1	2.20

DISTANCE	
x (μm)	x (mils)
0.6	0.02
1	0.04
5	0.20
10	0.39
20	0.79
25	0.98
25.4	1.00
100	3.94

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

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

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I Introduction (Peter Rudling)

The objective of the Annual Review of Zirconium Alloy Technology and Information on Zirconium Alloys (ZIRconium Alloy Technology (*ZIRAT*)/Information on Zirconium Alloys (*IZNA*)) is to review and evaluate the latest developments in zirconium alloy technology as they apply to nuclear fuel design and performance.

The objective is met through a review and evaluation of the most recent data on zirconium alloys and to identify the most important new information and discuss its significance in relation to fuel performance now and in the future. Included in the review are topics on materials research and development, fabrication, component design, and in-reactor performance.

Within the *ZIRAT*₁₃/*IZNA*8 Program, the following technical meetings were covered:

- International Conference on Nuclear Fuel, Montpellier, France, May 19-23, 2008.
- Jahrestagung Kerntechnik, Hamburg, May 27-29, 2008.
- Utility conference, USA, 2008 for *ZIRAT*₁₃ only.
- International Conference on Water Chemistry of Nuclear Reactor Systems, Berlin, Germany, September 15-18, 2008.
- Water Reactor Fuel Performance Meeting, Seoul-Korea, October 19-22, 2008.

The extensive, continuous flow of journal publications is being monitored by several literature searches of world-wide publications and the important papers are summarised and critically evaluated. This includes the following journals:

- Journal of Nuclear Materials,
- Nuclear Engineering and Design,
- Kerntechnik,
- Metallurgical and Materials Transactions A,
- Journal of Alloys and Compounds,
- Canadian Metallurgical Quarterly,
- Journal de Physique IV,
- Journal of Nuclear Science and Technology,
- Nuclear Science & Engineering,
- Nuclear Technology.

The primary issues addressed in the review and this report is 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 (Loss of Coolant Accident (*LOCA*), Reactivity Initiated Accident (*RIA*)).
- Dry 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. The data published in this Report is only from non-proprietary sources; however, their compilation, evaluations, and conclusions in the report are proprietary to *ANT* International and *ZIRAT/IZNA* members as noted on the title page.

The authors of the report are Dr. Ron Adamson, Brian Cox, Professor Emeritus, University of Toronto; Mr. Al Strasser, President of Aquarius and, Mr. Peter Rudling, President of *ANT* International, and Mr. Friedrich Garzarolli.

The work reported herein will be presented in three Seminars: one in Clearwater Beach, Florida, on February 3- February 5, 2009 one in Dresden, Germany on March 16-18, 2009 and one in Japan in 2009.

The Term of *ZIRAT*₁₃/*IZNA*₈ started on February 1, 2008 and ends on March 31, 2009.

2 Burnup achievements and fuel performance issues (Alfred Strasser)

2.1 Trends in fuel operating conditions

2.1.1 General trends

The incentive for the continued improvements in operating economics and reliable fuel performance has resulted in significant advances in materials technology, software for modelling fuel performance and sophisticated instrumentation and methods for post-irradiation examinations. These advances have been used to increase the demands on fuel performance levels and to put pressure on the regulatory bodies to license operations to increased burnup levels. The types of changes in Light Water Reactor (*LWR*) operating methods to achieve improved safety and economics have not changed in the past year and still include:

- Annual fuel cycles extended to 18 and 24 months.
- Discharge burnups increased from mid-30 to 58 GWD/MT batch average exposures by higher enrichments, increased number of burnable absorbers in the assemblies and in Pressurised Water Reactor (*PWRs*) higher Li and B levels in the coolant.
- Plant power uprates that range from 5 to 20%.
- More aggressive fuel management methods with increased enrichment levels and peaking factors.
- Reduced activity transport by Zn injection into the coolant.
- Improved water chemistry controls and increased monitoring.
- Component life extension with *HWC* and Noble Metal Chemistry (*NMC*) in Boiling Water Reactor (*BWRs*).

2.1.2 Fuel cycle lengths

The trend for increased fuel cycle lengths has come to a near “equilibrium” in the US with *PWRs* operating at an average of 500 Effective Full Power Days (*EFFPD*) per cycle and *BWRs* an average of 620 *EFFPD* per cycle. Nearly all the US *BWRs* are trending toward 24 month cycles. The older, lower power density *PWRs* have implemented the 24 month cycles, but fuel management limitations, specifically reload batch sizes required, have limited implementation of 24 month cycles in the high power density plants. The economics of 24 month cycles tend to become plant specific since they depend on the balance of a variety of plant specific parameters. The potential economic gains for cycle extension have decreased in the US since the downtimes for reloading and maintenance procedures have been significantly reduced.

Other countries that historically have had only one peak power demand per year in the winter compared to the two summer and winter power peaks in the US are also trending toward longer cycles as a result of changes in economics, maintenance practices and licensing procedures. *PWRs* are trending toward 18 month cycles in France, the UK (Sizewell B) and Germany. Japan is planning to increase cycle lengths in several steps, first to 15 months which does not require re-licensing, then to 18 and 24 months which will require re-licensing.

2.1.3 Burnup extension

The major incentive for extended burnups is the potentially improved fuel cycle economics, although other incentives might exist. The improved economics depend in part on the decreased amount of spent fuel assemblies to be purchased and handled. This is balanced by the increased amount of uranium and enrichment services required. The economics of decreased assemblies could be impacted by the much longer cooling times required in spent fuel pools prior to on-site dry storage or transport to a storage facility as noted later.

The average batch burnups in US *PWRs* are generally in the range of 45-54 GWD/MT and in US *BWRs* in the range of 45-50 GWD/MT.

Some European plants operated in the 50-58 GWD/MT batch burnup range and have designed to go to 62 GWD/MT in their current cycles in both *PWRs* and *BWRs*. This is feasible, in part, due to their greater margin to regulatory limits. The maximum burnup Lead Use Assemblies (*LUAs*) are in the range of 67 – 75 GWD/MT for both reactor types.

The Voda Voda Energo Reactor (*VVER-440*) fuel is up to 57 GWD/MT assembly burnup with plans to get to 65 GWD/MT. The *VVER-1000* fuel is up to 55 GWD/MT assembly burnup with plans to go to 68 GWD/MT, Molchanov et al., 2007. Rod burnups of 73 GWD/MT burnup have been achieved in both *VVER-440* and *VVER-1000* fuel, Markov et al., 2008.

The Japanese are planning to raise current *BWR* 45 GWD/MT batch average burnup to 50 GWD/MT without exceeding the 55 GWD/MT peak assembly burnup regulatory limit for 9x9 “Step 2” fuel, Itagaki & Murata, 2007. This will be done in conjunction with cycle extension from their current 13 month to future 16 or 19 month cycles. The *PWR* assembly burnups will also be raised to 55 GWD/MT for all rod arrays. A peak rod burnup of 73 GWD/MT has been reached by a fuel rod with Mitsubishi Developed Alloy (*MDA*) alloy cladding, Watanabe et al., 2008.

The burnups are currently limited by the regulatory agencies more than by technical limitations, except for *LUAs* and rods. Some of the current regulatory limits are summarized in Table 2-1.

Table 2-1: Regulatory burnup limits in GWD/MT.

USA	62.5 peak rod	Korea	60 peak rod
Belgium	55 UO ₂ assembly	Netherlands	60 peak rod
	50 MOX assembly	Sweden	60 assembly
Finland	45 assembly		64 rod
France	52 assembly	Switzerland	75 peak pellet
Germany	65 assembly	Taiwan	60 peak rod (<i>PWR</i>)
Japan	55 UO ₂ assembly		54 peak assembly (<i>BWR</i>)
	(66 peak rod)	UK	55 peak pellet
	45 MOX assembly		

Economic analyses reported in past *ZIRAT* reports indicated that economic incentives for extending burnups beyond the 60-70 GWD/MT batch average range will disappear and that other incentives will be needed to go beyond this level. The biggest block to increasing burnup is the essentially universal regulatory limit on 5.00% initial uranium enrichment. Analyses reported in *ZIRAT11/IZNA6* for various reactors, including *PWRs* and *BWRs*, showed that a 5% enrichment will provide up to 55 GWD/MT in a *VVER* and about 63 GWD/MT in a General Electric (*GE*) *BWR assembly* average burnup. Advanced, sophisticated fuel management methods can probably raise this close to 70 GWD/MT, but are unlikely to exceed this level. The cost of equipping and licensing the industry to handle enrichments greater than 5% would be very great and time consuming; and for that reason such a move is not being considered for the foreseeable future.

2.1.4 Power uprates

The power of the majority of the operating plants has been uprated, the maximum being 20%. While the fuel performance limits remain the same for the uprated conditions, the number of fuel assemblies operating at higher power change and the margin to the limits might be reduced. The effect of higher flow rates on hydraulic and water chemistry effects on the interaction with the fuel must also be considered. These changes have not affected the failure rate or apparently the fuel performance; however, it does increase the statistical probability of increasing power and burnup related factors on the fuel performance.

The most recent power uprates in the US are:

Approved (since 2005 with % of thermal power increase)

<i>PWRs</i>		<i>BWRs</i>	
Ginna	16.8	Vermont Yankee	20.0
Beaver Valley 1 & 2	8.0	Hoop Creek	15.0
Waterford 3	8.0	Susquehanna 1 & 2	13.0
Millstone 3	7.0	Browns Ferry 1	5.0
Indian Point 3	4.9	Cooper	1.6
Comanche Peak 1 & 2	4.5		
Palo Verde 1, 3	2.9	Applications under review (Oct. 08)	
Seabrook	1.7	Browns Ferry 1, 2 & 3	15.0
Vogtle 1 & 2	1.7		
Davis Besse	1.6		
Crystal River	1.6		
Fort Calhoun	1.5		
Calvert Cliffs 1 & 2	1.3		

The effect on fuel power distribution was discussed in ZIRAT12/IZNA7 and the effect on water chemistry was discussed in the Special Tropical Report “Consequences of Power Up-rating” to the LWR Coolant Chemistry (LCC3) Seminar. A recent example of the effect of uprating on HWC and its potential effectiveness to prevent InterGranular Stress Corrosion Cracking (IGSCC) of core components is summarized below.

Investigators in Taiwan (Wang et al., 2007) modelled the water chemistry and Electrochemical Corrosion Potential (ECP) of a BWR with hydrogen levels from zero to 2 ppm at power uprates that ranged up to 120%. The modelling points were at four different locations in the reactor as shown in Figure 2-1 the upper plenum (3), the upper downcomer (7), the recirculation system (9) and the bottom lower plenum (11). The analysis was done with the DEMACE computer code. Variables analyzed included the coolant flow velocity, the neutron and gamma dose rates related to the power levels. Taiwan Power Co. operates two BWR 4s and two BWR 6s; however, the BWR modelled was not identified.

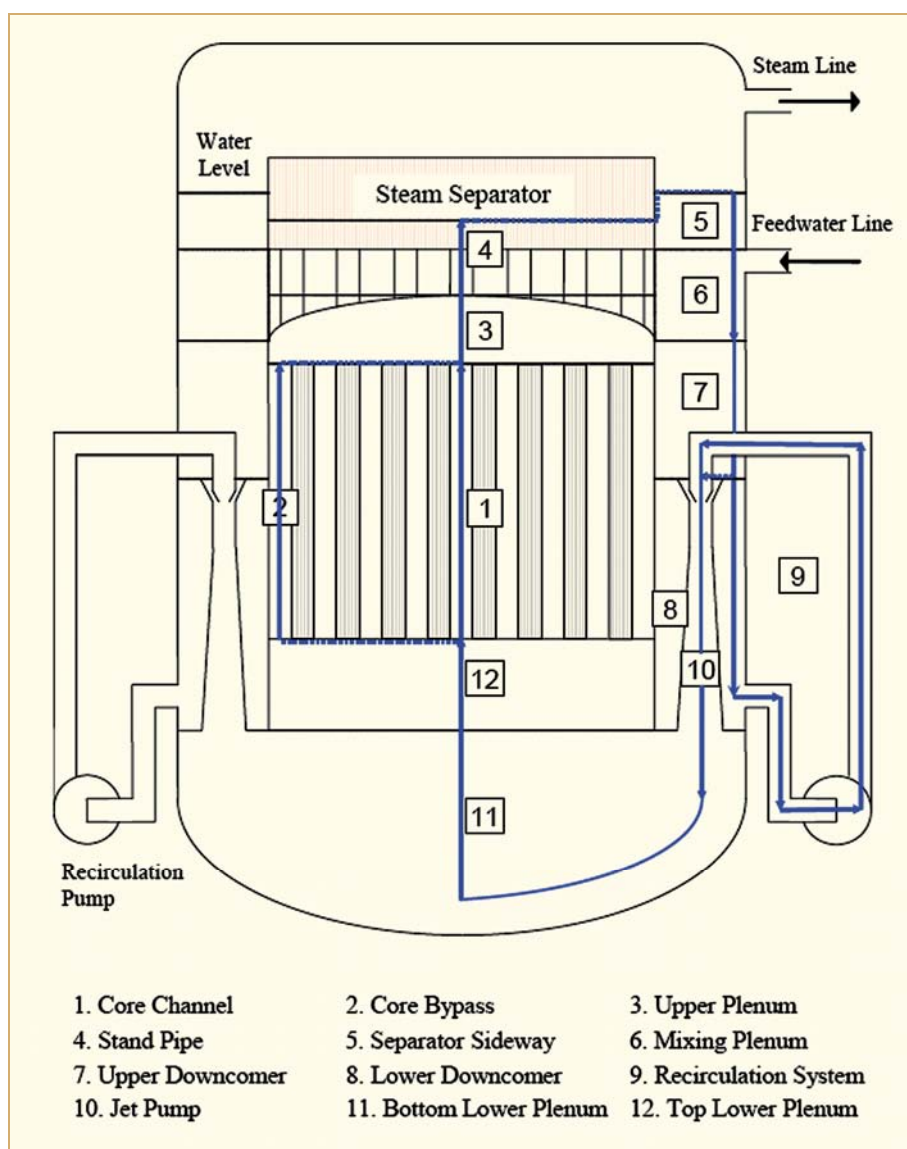


Figure 2-1: ECP monitoring points in a typical BWR primary coolant circuit, Wang et al., 2007.

The hydrogen peroxide (H_2O_2) levels did not change consistently with increasing power and hydrogen levels, but peaked at about 104% power. One example showed that the H_2O_2 level remained at >10 ppm at 104% power with hydrogen at 2 ppm while the H_2O_2 levels at other powers, up to 120%, went down with 1.9 ppm hydrogen. This effect is believed to be the unique combination of enhanced water radiolysis and reduced coolant residence time that produces a more oxidizing atmosphere. A similar effect was noted in the ECP modelling.

The ECP in the coolant is determined by the concentration of all the redox species, hydrogen, oxygen and H_2O_2 . The results showed there were no significant differences in ECP due to power uprates when the hydrogen levels were either below or above the critical level of $-0.23 V_{\text{SHE}}$, required to protect stainless steel from IGSCC. A particular uprate level, however, such as 104%, can induce a more oxidizing atmosphere at most locations. A slightly beneficial effect of HWC could exist at the upper plenum and upper downcomer at certain power levels, Figure 2-2.

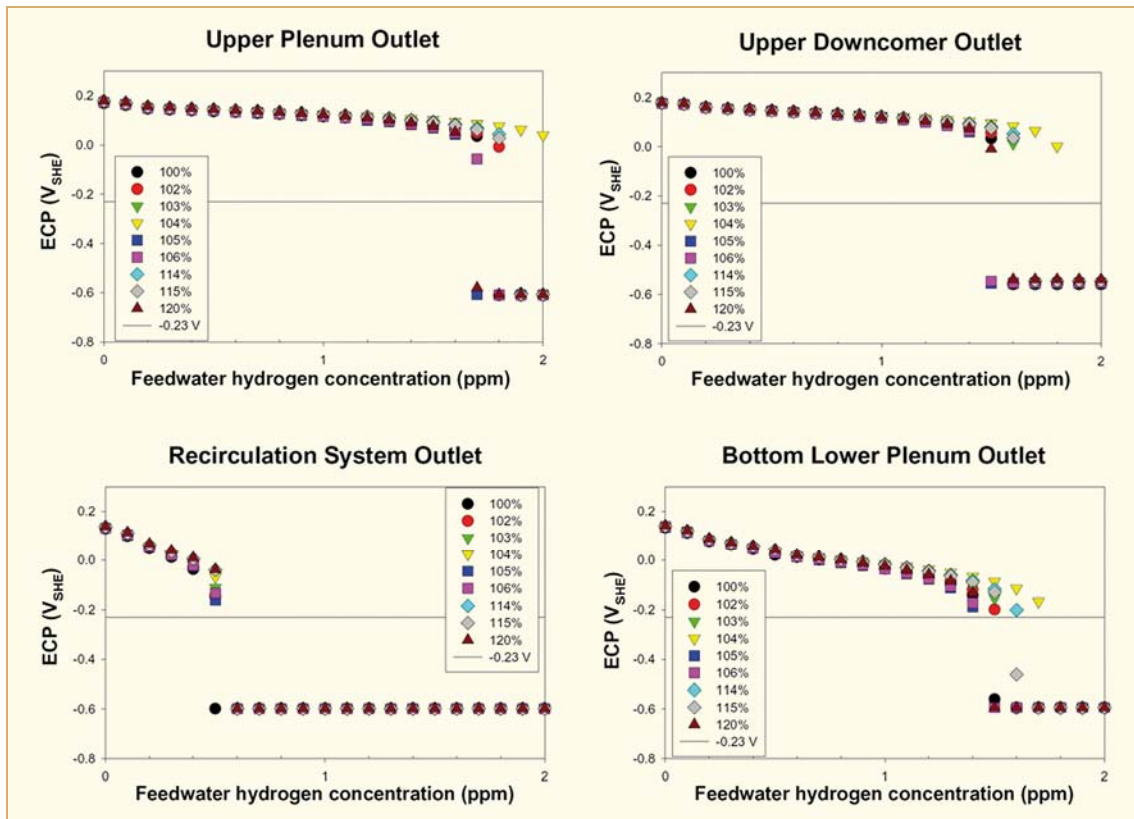


Figure 2-2: Variations in *ECP* as a function of $[H_2]_{FW}$ at four selected locations of a *BWR* with operating power levels ranging from 100% to 120% of the rated power, Wang et al., 2007.

The *conclusion* of the work was that the effect of power uprates on *HWC* efficiency varied from location to location within the coolant system, but that no consistent trend could be found as a function power level. Both degradation and slight improvement of *HWC* could be identified, however, it should be noted that these are specific to the *BWR* system analyzed and are probably not directly applicable to another system. *Nevertheless it would be prudent to re-analyze the potential effectiveness of the HWC level when contemplating a plant uprate.*

2.1.5 Water chemistry

2.1.5.1 High lithium operation

Increasing burnup, cycle lengths and enrichments levels, combined with the recommended high, 7.4 pH operation has increased the required lithium (Li) levels above the previously used maximum 3.5 ppm concentrations. The potentially increased zirconium alloy corrosion levels due to increased Li levels has created a cautious approach to increasing the Li levels; the concentration of Li in high Chalk River Unidentified Deposits (*CRUD*) levels and/or in combination with nucleate boiling is of particular concern.

Comanche Peak

The highest Li level used has been at the Comanche Peak Steam Electric Station Unit 2 (CPSES 2) operating with ZIRconium Low Oxidation (ZIRLO) clad fuel:

Cycle	Li (ppm)	pH	Days >3.5 ppm Li
7	<5	7.3	66
8	<6	7.4	334
9	<6	7.4	468

The cycles were 18 months long and the fuel accumulated 42-50 GWD/MT assembly burnup after 3 cycles. The CPSES 1 unit operated at <3.5 ppm Li and pH of 7.1-7.2 as a reference plant. Results from the first two cycles were reported in ZIRAT11/IZNA6 and the results from the third cycle (Cycle 9) were reported in ZIRAT12/IZNA7. The most recent report (Stevens, 2008) provides all the results and is summarized here.

A total of 8 assemblies were examined from Unit 2 and 8 from Unit 1. The assemblies operated with nucleate boiling and their thermal duty is summarized for the Unit 2 fuel rods examined, Table 2-2. No localized or accelerated corrosion or oxide spalling were observed during the high visual magnification examination. No visual differences were observed between the fuel from Unit 2 and the Unit 1 fuel that operated with standard Li content.

Table 2-2: Fuel duty parameters in Comanche Peak Unit 2, Stevens, 2008.

	<i>HDCI</i>	Max steam rate kg/hr-m ²	Integrated boiling 1E09 kg	Max core boiling area m ²
U2C6	156	2494	3.5	324
U2C7	152	2191	2.0	242
U2C8	155	3035	2.2	207
U2C9	154	2474	3.1	248

An increasing trend in localized *CRUD* deposits was observed on Unit 2 fuel from Cycle 6 to 8 which peaked in Cycle 8 and decreased in Cycle 9. The variations in *CRUD* levels can not be attributed to the pH alone, since there were changes in fuel duty and boiling conditions that could have contributed to the variability. Oxide measurements are shown in Figure 2-3 as a function of burnup that range up to 60 μ , observed on the upper span of some of the rods. A comparison of axial oxide measurements on 2 cycle rods from Unit 1 and Unit 2 are shown on Figure 2-4. The slightly higher oxide levels in Unit 2 are believed to be due to higher fuel duty and some metallurgical differences in the cladding alloy. A cautionary note by the author states that the oxide measurements include both a *CRUD* and oxide component and are not separated by the measurement technique.

Based on these results the utility has considered 7.4 pH operation fully qualified and adopted this operation to both their units. Unit 1 is in its 3rd high Li cycle and Unit 2 in its fourth cycle.

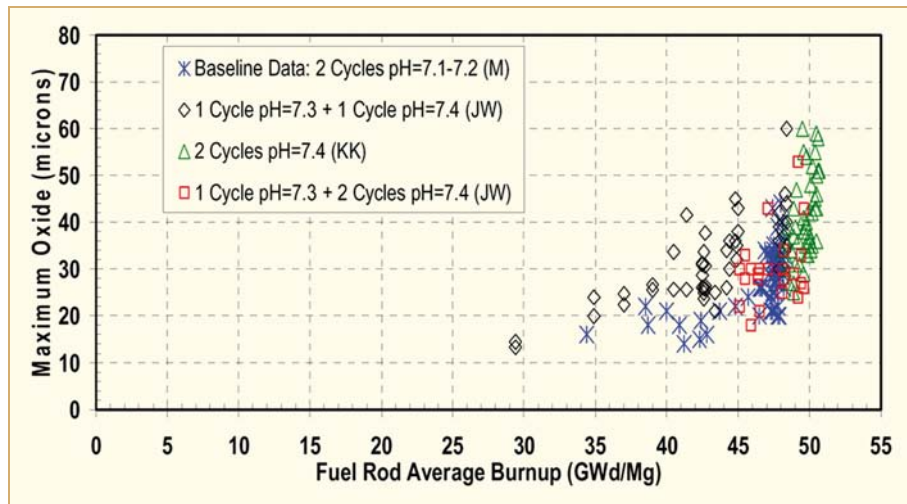


Figure 2-3: Oxide measurement results on ZIRLO in high Li program, Stevens, 2008.

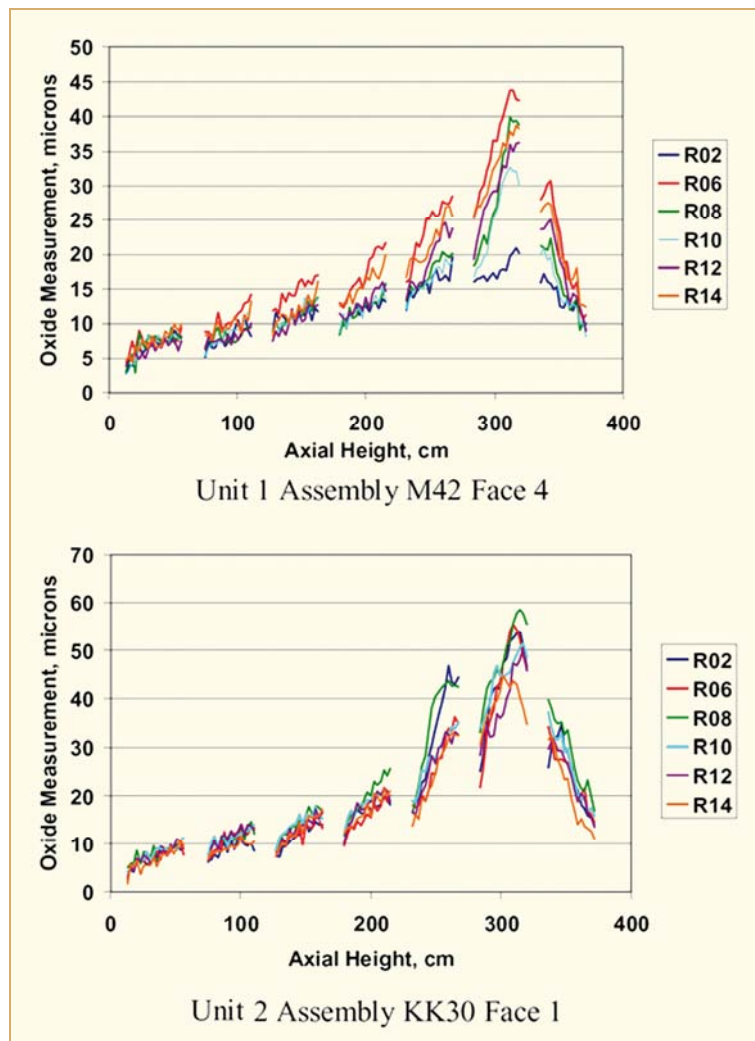


Figure 2-4: Comparison of oxide measurements on two cycle fuel rods from Comanche Peak Units 1 & 2.

Ringhals

A high pH, high Li program has been in progress at the Ringhals 2 and 3 Units for several years, Bengtsson et al., 2008. The practice has been to maintain a pH of 7.4 with Li up to a level of 5 ppm, Figure 2-5. The technical specification was recently raised to 6 ppm Li and the intent is to start all units at pH >7.1 and maintain the appropriate level of Li until a pH of 7.4 is reached. Ringhals Unit 4 has not adopted this operating mode since it still has Alloy 600 steam generators sensitive to corrosion, while Units 2 and 3 have changed their steam generators to Alloy 690 over 10 years ago.

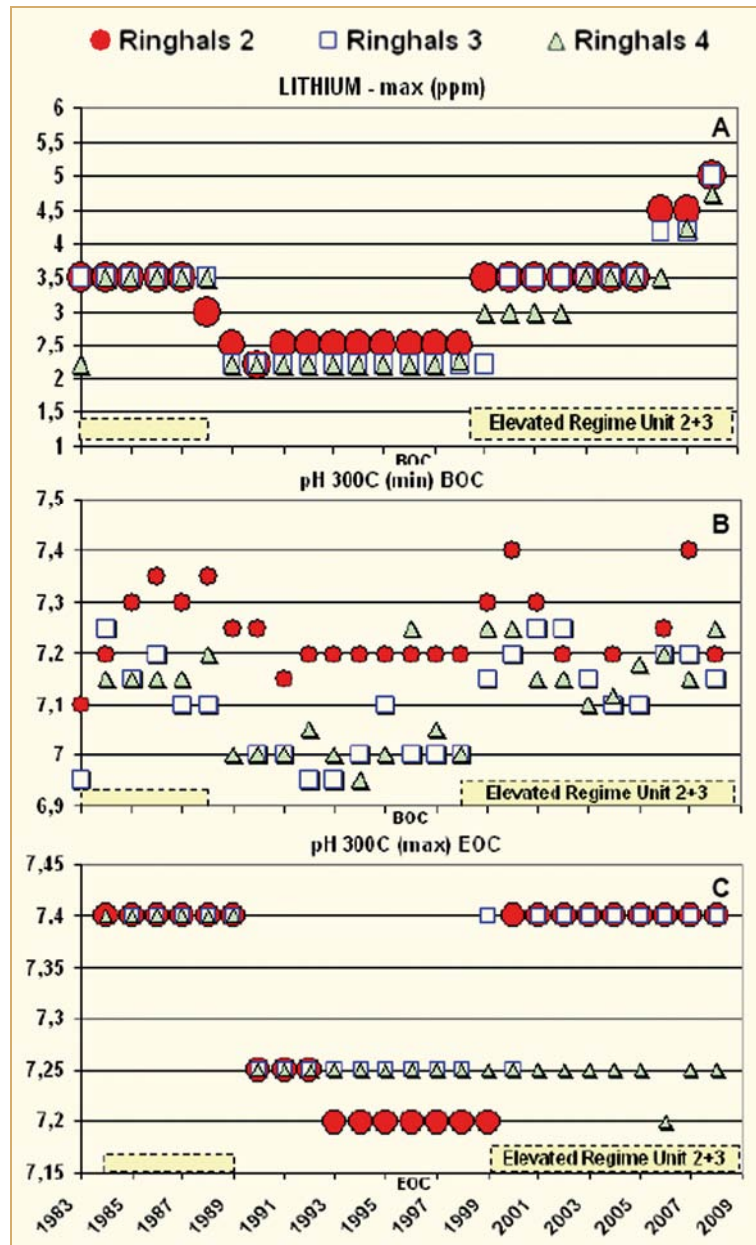


Figure 2-5: Elevated pH/Li-program in Ringhals PWRs 1983-2008, Bengtsson et al., 2008.

All three Units are currently operating with M5 fuel cladding and “oxide thickness measurements have all been within the vendor experience data base”. Unfortunately no data are given in the publication. No related cladding leakages have occurred.

CRUD deposits on the fuel have been 5 to 10 times lower in Units 2 and 3 than Unit 4 operating at a constant 7.25 pH; however, this is primarily due to the lower corrosion of the Alloy 690 steam generator tubes in Units 2 and 3, although the higher pH operation may play a role in this. As a result the Ni deposits on the fuel have been reduced to 100-200 grams from 2-3 kg in Units 2 and 3.

The ^{58}Co concentration was reduced quite significantly in Unit 3, by nearly a factor of 30, to 50-100 Bq/kg during normal operation, a level below even that of Unit 4 with Alloy 600 tubes. Some, but not nearly similar reduction was observed in Unit 2. Some of these differences are believed by the authors to be due to the poor reliability of the hot leg coolant sampling.

2.1.5.2 Zinc Injection

One of the successful innovations in operating methods has been zinc (Zn) injection into the coolant to reduce activity transport within the primary coolant systems of both PWRs and BWRs, without significantly affecting cladding corrosion, be it Zircaloy-4, ZIRLO or M5. The presence of Zn, however, promotes the deposit of a Zn containing spinel on the cladding probably in the form of $\text{Ni}_x\text{Zn}_y\text{Fe}_z\text{O}_4$, especially if sufficient Fe is present. While these CRUD deposits have not been observed to affect corrosion of the cladding, they are suspected of promoting boron deposits that can result in Axial Offset Anomalies (AOAs) also called CRUD Induced Power Shifts (CIPSs) in plants with subcooled nucleate boiling. A key question has therefore been the effect of Zn injection on cores with nucleate boiling and the experience of such high duty plants with recently applied Zn injection. Nearly all the BWRs have applied Zn injection.

A total of 45 PWRs are applying Zn injection worldwide (17% of all PWRs) and 20 additional plants are in the process of starting to implement Zn injection. The increase in Zn injection application over the past years is shown on Figure 2-6(a). The majority of these plants, 82%, are using the process for dose reduction and the rest for mitigation of Stress Corrosion Cracking (SCC), AOA or CIPSs, Perkins et al., 2008. All of the currently used PWR cladding alloys have been exposed to Zn injection with satisfactory results. ZIRLO has had the most exposure at >5500 Zn ppb – months and Zircaloy-4 and M5 at >350 Zn ppb – months, Figure 2-6(b).

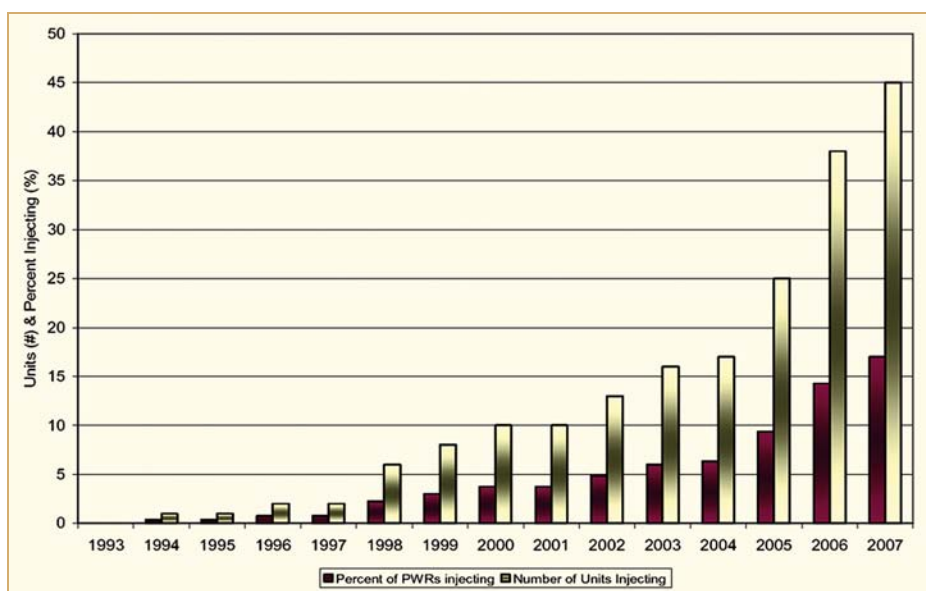


Figure 2-6(a): Zinc injection industry trends, Perkins et al., 2008.

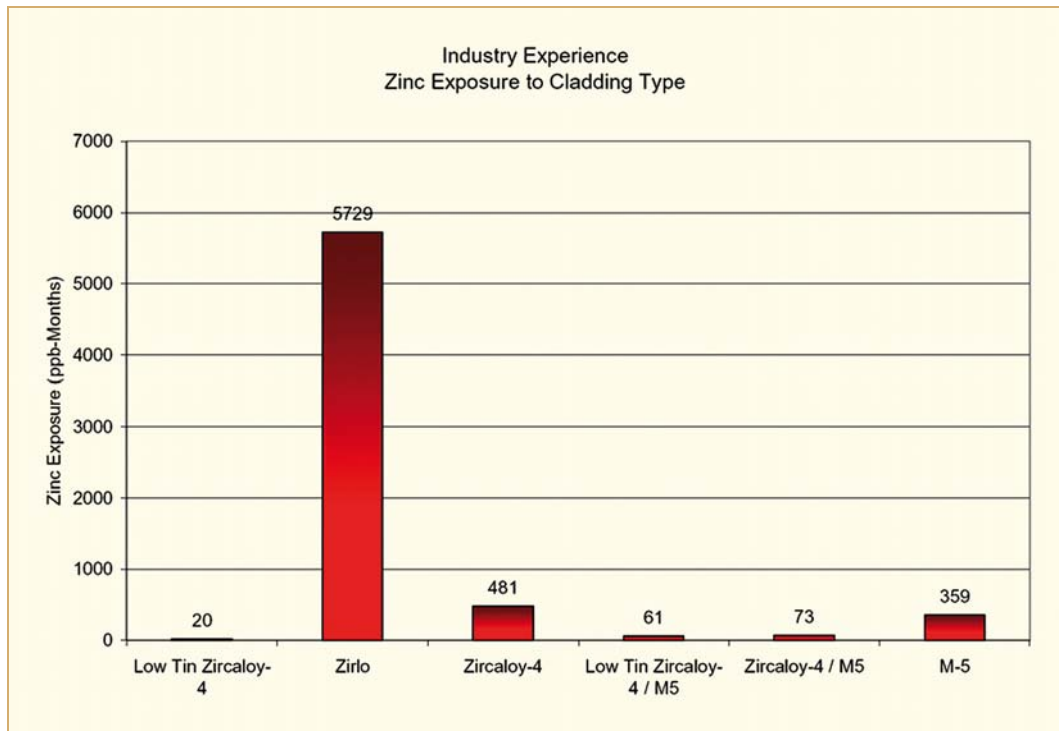


Figure 2-6(b): Zinc exposure by fuel cladding type, Perkins et al., 2008.

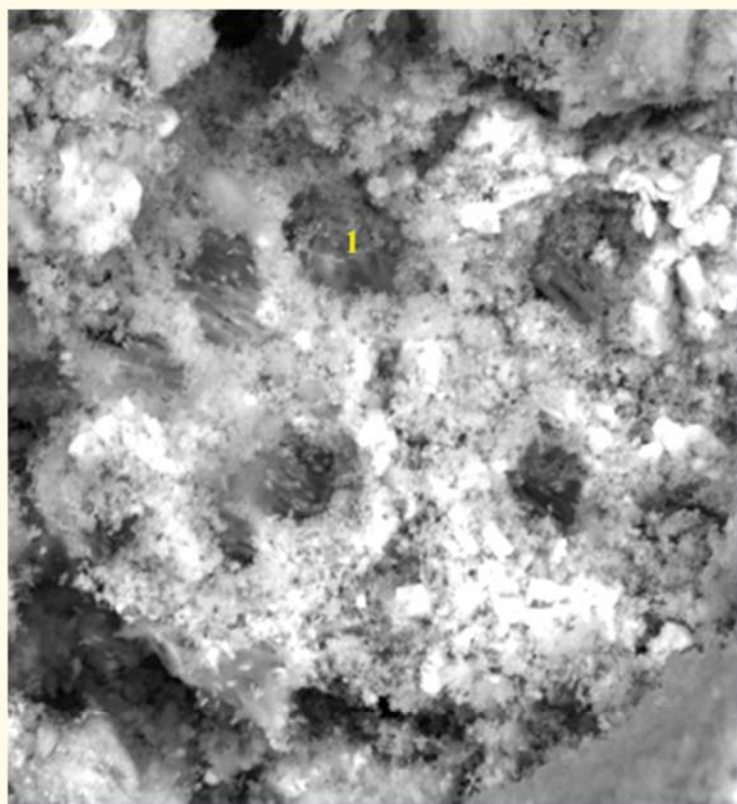
While the effects of Zn injection on cladding in high duty plants has been benign, there are concerns that Zn could liberate corrosion products from the structural materials that would deposit on the cladding or that Zn compounds themselves could be concentrated on the cladding with nucleate boiling. To determine the Zn concentration levels that could affect cladding performance, the potential effects of Zn concentration on cladding surfaces with nucleate boiling are being investigated by a Westinghouse/Electric Power Research Institute (EPRI) project (Byers et al., 2008) using ex-reactor tests. The tests consist of a heated rod with artificial *CRUD* (NiFe_2O_4) in a water loop with increasing amounts of soluble Zn ranging from zero to 60 ppb. The cladding temperatures are measured by thermocouples on the cladding Inner Diameter (ID) to record increasing thermal resistance due to the *CRUD* deposition and potentially the Zn concentration. The operating parameters of the loop and rod are given in Table 2-3.

Table 2-3: Loop operating parameters, Byers et al., 2008.

Parameter	Value
Heated power	7.7 kW
Heated rod current	630 amps
Heated rod length	0.29 m
Heat flux	900 kW/m ² (90 W/cm ²)
Pressure	15.5 MPa (2250 psia)
Flow velocity	4 m/s
Heater rod OD	9.5 mm
Inlet temperature	329 to 332°C
Rod material	ZIRLO™
Boron concentration	1000 ppm
Lithium concentration	2.2 ppm
Hydrogen concentration	40 cc/kg H ₂ O

The publication is limited to reporting the results of one test with the maximum amount, 60 ppb Zn injection subsequent to the deposition of the artificial *CRUD*. The clad *ID* temperature increased abruptly from the original 371°C after each *CRUD* addition until it levelled off at 383°C. Two hours after the Zn injection the temperature started to increase again and levelled off at 392°C indicating that the Zn addition resulted in a thermal resistance by some means.

Scanning Electron Microscopy (*SEM*) analyses showed the structure of the *CRUD*, 50 µ thick, indicated wick boiling chimneys with a white material near the bottom of the chimneys, subsequently identified as a zinc oxide (ZnO) rich phase. The Zn rich crystals grew in fibrous bundles, each about 10 µ long and 1 µ in width and precipitated in the chimneys where the Zn was concentrated, Figure 2-7. Recent measurements determined that the solubility limit of ZnO in water at the rod surface temperatures is about 200 ppm, so that this experiment would indicate a concentration factor of about 3.3.



Top-Down SEM image of deposit at 1500X magnification. EDS point analysis location marked with a "1".

EDS analysis of deposit								
Elemental concentration (wt%, no oxygen)								
	Al	Si	Ca	Fe	Ni	Cu	Zn	Zr
Image Average	2.7	0.1	0.8	45.4	17.0	13.5	20.4	0.2
Point 1	2.4	0.0	1.7	21.8	8.3	8.8	57.0	0.0

Figure 2-7: *CRUD* deposit post 60ppb Zn injection in EY-reactor loop, Byers et al., 2008.

The results of this experiment are not substantiated by in-reactor experience as noted subsequently. Neither one of the highest duty plants with Zn injection, Callaway and Vandellos 2, formed any high Zn compounds, all had spinels primarily in the form of $\text{Zn}_{0.2}\text{Ni}_{0.8}\text{Cr}_{0.8}\text{Fe}_{1.2}\text{O}_4$. *CRUD* samples from the highest duty US plant, Vogtle 1, indicated < 1% Zn, Byers et al., 2008. Nevertheless the experiment indicates a potential mechanism that is worth monitoring in-reactor.

Zircaloy-4

Zn injection in the range of 5 to 13 ppb was implemented in the **French 900 MWe Bugey plants**, for 27 months (3 cycles) in Unit 2 and 11 months (1 cycle) in Unit 4, Tigras et al., 2008(b). Oxide measurement results after these periods were within the data base of previous measurements and agreed with modelling predictions, Figure 2-8. All the cladding involved was Zircaloy-4, less corrosion resistant than the other alloys exposed to Zn Injection; however, these units are also lower duty than the other plants discussed.

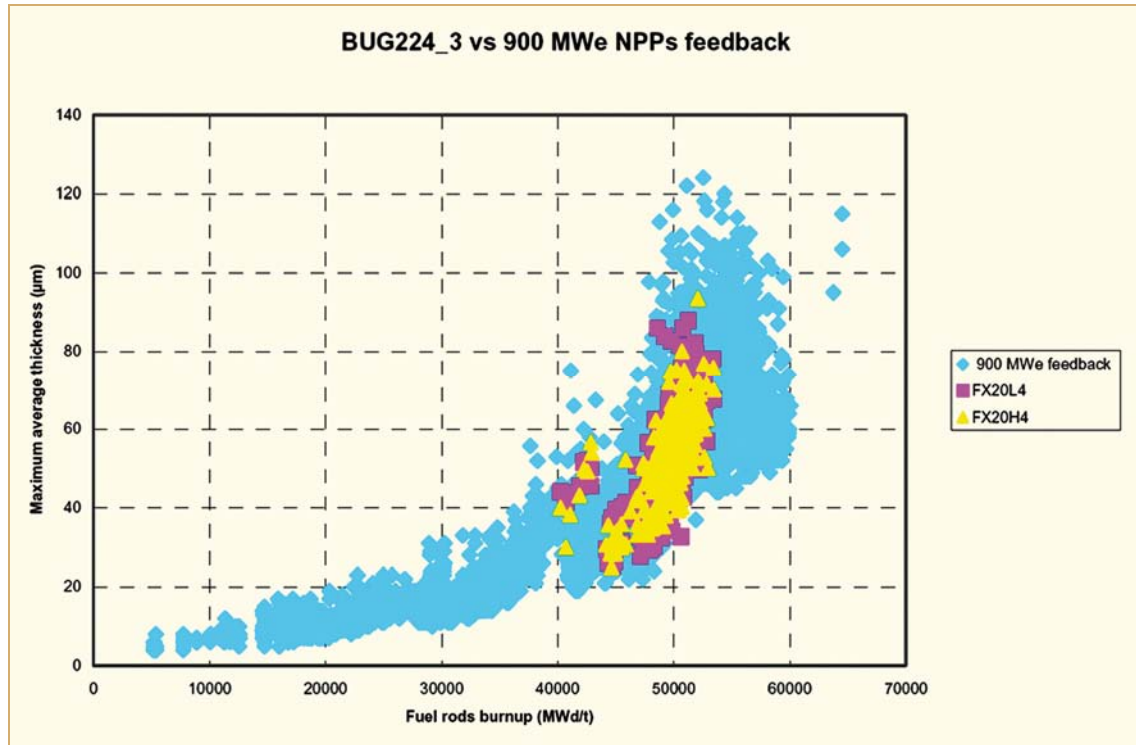


Figure 2-8: Measurement of fuel oxide thickness at Bugey 2 with zinc injection compared with data from 900 MWe units without zinc injection, Tigras et al., 2008(b).

ZIRLO

ZIRLO clad fuel has been exposed to Zn injection in both US and Spanish high duty plants. **Diablo Canyon Unit 2** operated with up to 25 ppb Zn in combination with high Li and examined the fuel after one and two cycles of operation, Iyer et al., 2008. Cycle 13 operated with 4.5 ppm Li and Cycle 14 with 5.1 ppm Li. The fuel duty included nucleate boiling in Cycle 14 with a steaming rate of 316 lbs/hr/ft² units based in the VIPRE-W code. This was a lower power than Comanche Peak 2, which operated with higher Li for longer times. Diablo Canyon had Intermediate Flow Mixing grids (IFMs) which increase turbulence resulting in an increase in the single phase forced heat transfer coefficient, improvement of heat transfer and decrease in corrosion rates.

Oxide measurements were made after both Cycles 13 and 14 after brushing the *CRUD* off. The highest burnup rods were 26-27 GWD/MT after their first Cycle 13 and 51-52 GWD/Mt after their second Cycle 14. Peak oxide measurements on the highest exposure rods ranged between 15 μ and 20 μ . Operation included up to 1.8 ppm silica impurity in the Cycle 14 coolant, which could promote a tendency to form zinc silicates. All of the oxide measurements were within the standard Li data base without Zn injection and well within the levels predicted by the corrosion model, Figure 2-9. The comparison also includes the high Li operation without Zn at Comanche Peak 2.

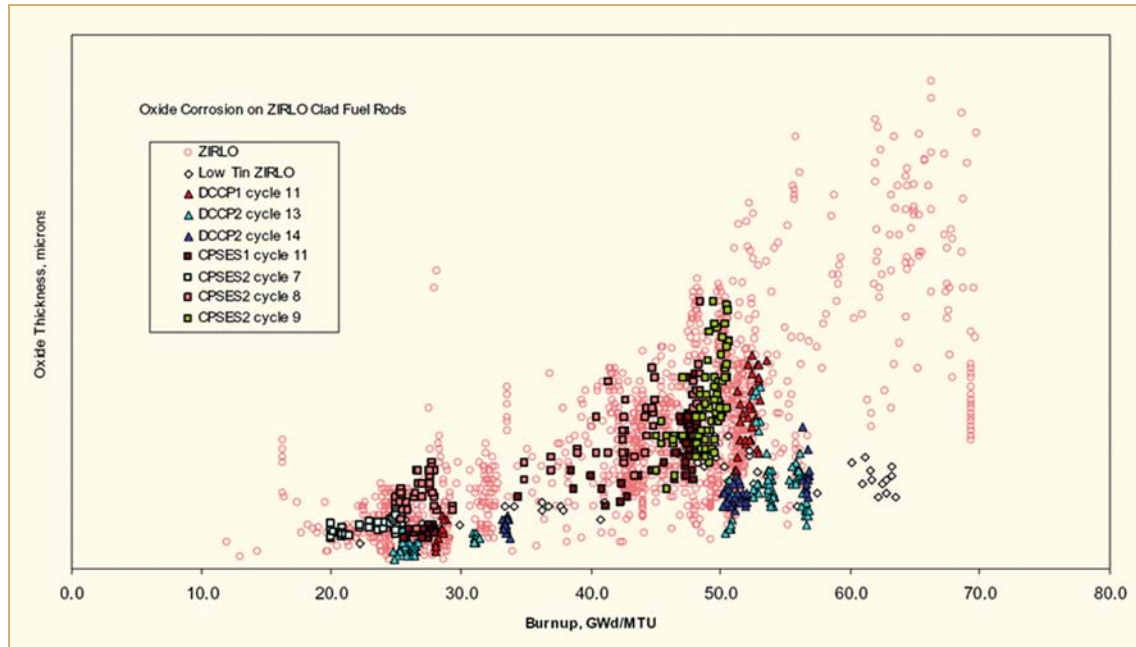


Figure 2-9: Peak measured fuel rod oxide thickness as a function of burnup for Diablo Canyon unit 2 cycles 13 and 14 and Diablo Canyon unit 1 cycle 11 against overall ZIRLO™ database, Iyer et al., 2008.

Three high duty Spanish plants, Vandellos 2, Asco 1 and 2, each applied zinc injection for a half of a cycle, starting at 5 ppb in mid-cycle and raising the level to 10 ppb Zn halfway to the end-of-cycle. The plants operated with normal <3.5 ppm Li at about 7.2 pH. Vandellos 2 is the highest duty plant of the three and the fuel there was covered with more *CRUD* than the Asco assemblies. Vandellos 2 also operates with Alloy 600 steam generators while the Asco plants operate with Alloy 800, which has better general corrosion resistance. The oxide measurement results of Vandellos 2 rods are shown on Figure 2-10 and for Asco-2 on Figure 2-11. Comparison of the maximum oxide thickness data from the Zn injection cycles to the data from pre-Zn injection cycles does not show any significant differences. *Note that the data are limited by the limited half cycle exposure accumulated to date.*

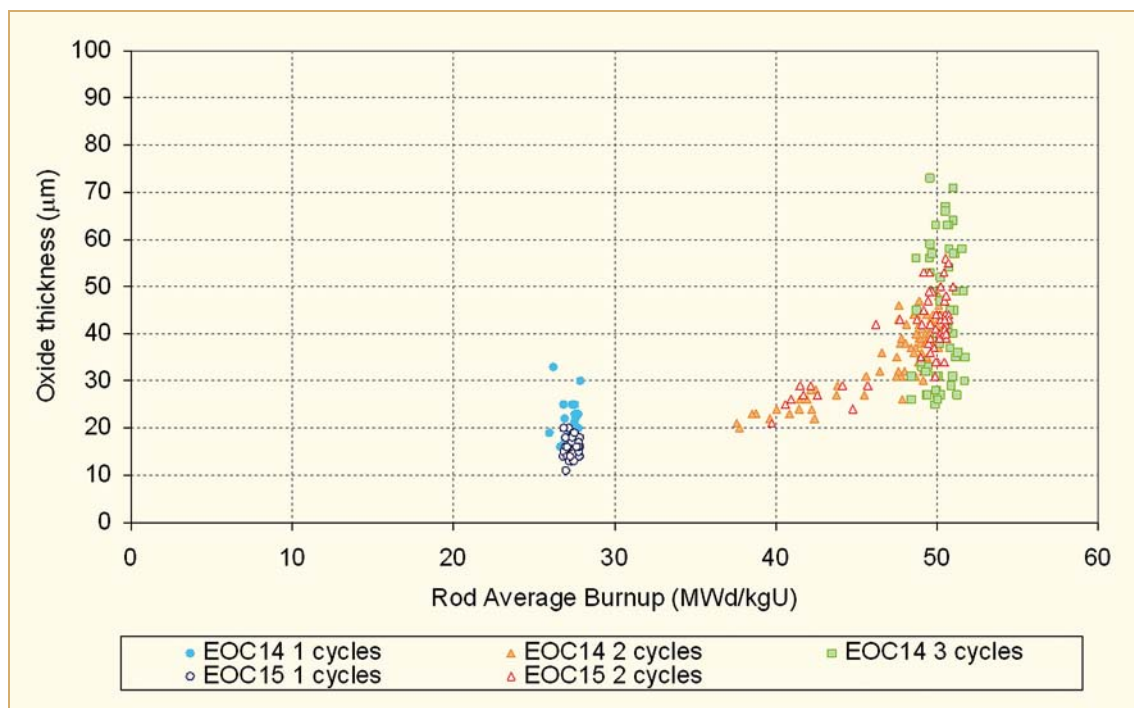


Figure 2-10: Vandellos 2 ZIRLO maximum oxide thickness vs. rod average burnup, Sánchez et al., 2008.

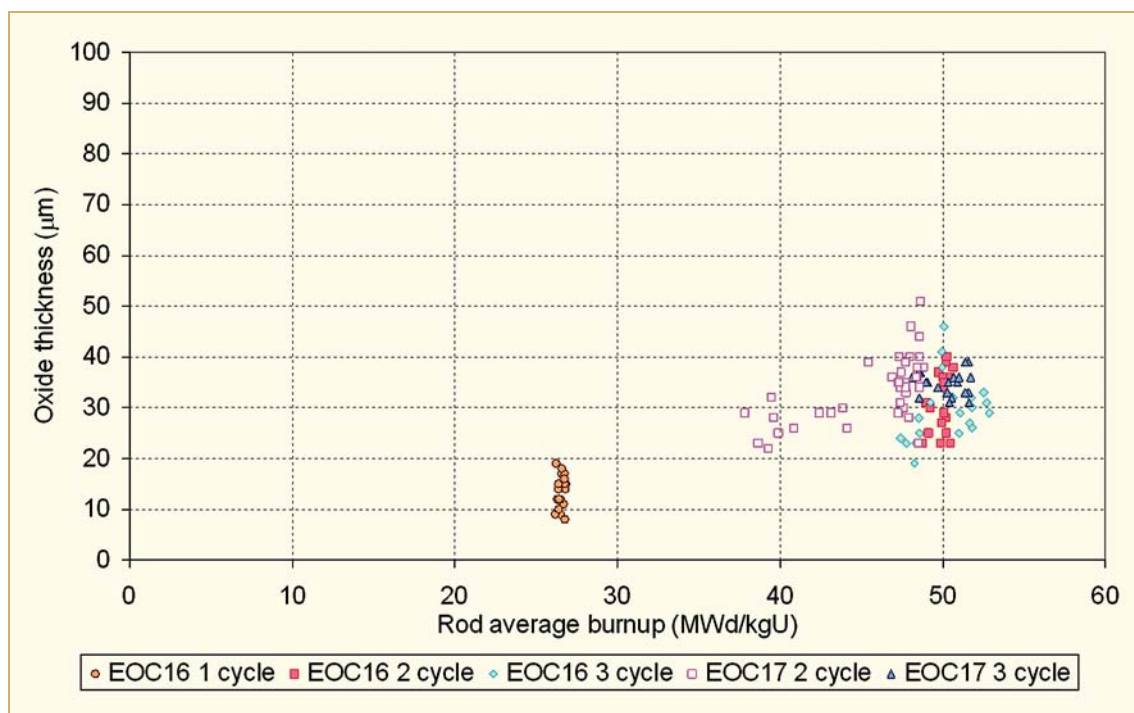


Figure 2-11: Asco 2 ZIRLO maximum oxide thickness vs. rod average burnup, Sánchez et al., 2008.

The Figures give some interesting data for ZIRLO in non-Zn injection cycles as well up to about 50 GWD/MT. The higher oxide thickness range of 25 to 70 μ in Vandellos 2 compared to the 25 μ to 50 μ range in Asco 2 is probably due in-part to the higher fuel duty level in Vandellos 2.

Analyses of samples taken from the most crudded rods at the end of the Vandellos 2 Zn injection cycle (No. 15) are of particular interest, especially since there is a notable absence of CRUD scrape results in the other publications. The results indicate an average composition of $Zn_{0.1}Ni_{0.8}Cr_{0.4}Fe_{1.7}O_4$ with a Zn percentage of about 5.5%. No Zn compounds were identified, such as noted in the previously described ex-reactor experiment. The composition was similar to that found previously in Callaway. A Zn-Fe-Cr ternary diagram based on Scanning Transmission Electron Microscopy (STEM) data and chart showing the distribution of the major CRUD components is shown in Figure 2-12(a).

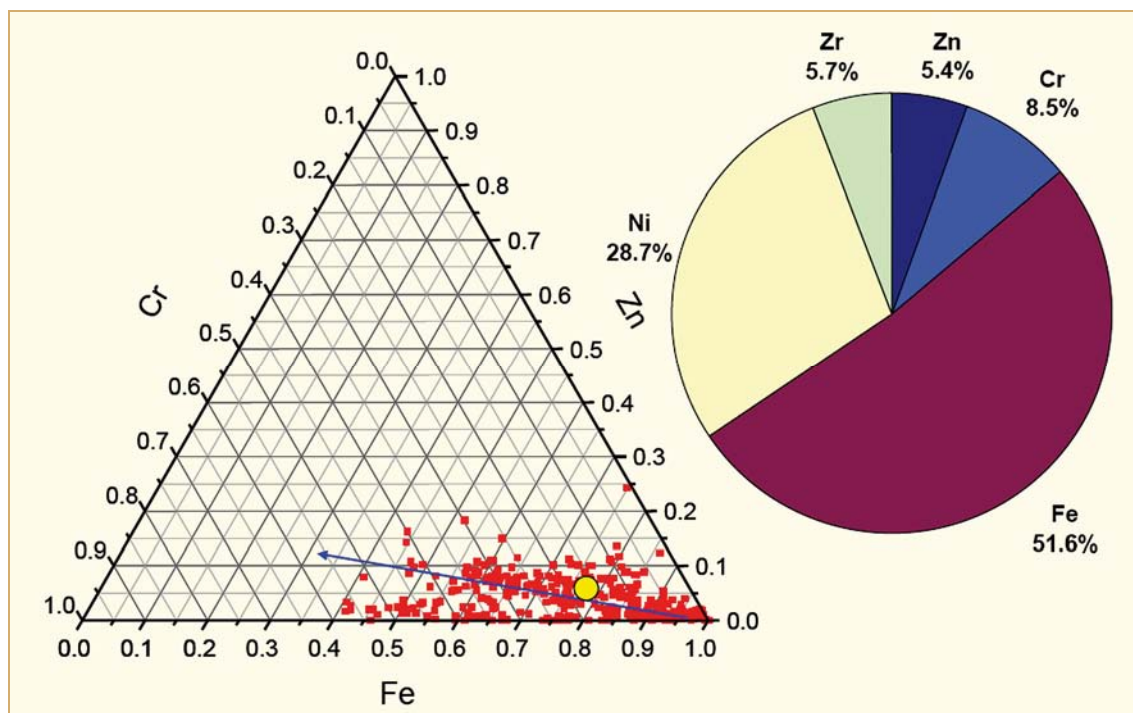


Figure 2-12: CRUD composition after Zn injection cycle 15 in Vandellos 2, Sánchez et al., 2008.

M₅

Assemblies with M₅ cladding were examined after operation for one, two and three 18 month cycles in the Sequoyah 2 PWR to exposures of up to 51.2 GWD/MT, a medium duty plant with nucleate boiling, Guillermier et al., 2008. The target level of 5 ppb Zn varied from about 2 to 7 ppm and operation with a pH of 7.15 and 3.5 ppm max. Li the CRUD levels accumulated after 3 cycles were low and the oxide measurements varied from 5 to 19 μ m on the highest burnup fuel with readings taken on the hottest top span of the assembly. The data fell well within the M₅ corrosion data base without Zn injection, Figure 2-12(b). Water chemistry was controlled within AREVA recommendations of <3ppb Ni and <1.5 ppb SiO₂, with actual values of <1 ppb Ni most of the time and <0.4 ppb SiO₂.

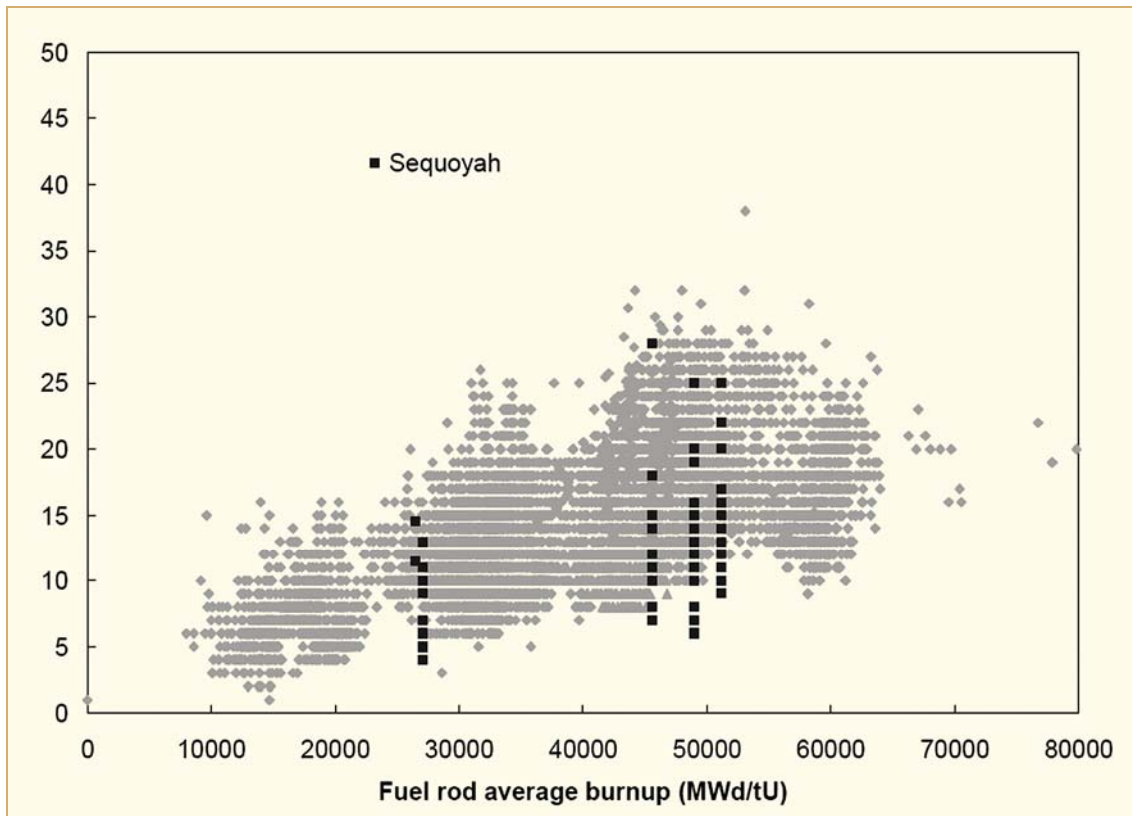


Figure 2-12(b) Oxide thickness after Zn injection cycles in Sequoyah 2 (in black) compared to peak rod oxide thickness without Zn injection (grey) for M5 cladding, Guillermier et al., 2008.

An excellent paper evaluating the basic mechanism by which zinc injection works and its effects on dose reduction, materials and components, *PWR* operation and the related experimental data is by Tigras et al., 2008(a). Some of its conclusions and this authors' are summarized below.

Conclusions applicable to operating conditions to date

- A dose rate reduction has been observed in every plant as a result of Zn injection without any apparent negative impacts on plant operation.
- A relationship between Zn exposure and dose rate reduction can not be generalized due to variables such as plant system and materials designs, the age of the plant, power levels and other factors.
- Zn injection has not affected the oxidation rates of *PWR* alloys Zircaloy-4, ZIRLO or M5.
- *CRUD* deposits while different (thinner and denser), do not appear to be detrimental to heat transfer and may be beneficial for avoiding AOAs.
- *CRUD* scrapes have not identified any morphology or composition that would be harmful to heat transfer.
- Zn injection can reduce Ni input from structural materials and *CRUD* deposition on fuel and a recommendation has been made to apply it prior to steam generator exchanges and as early as possible in a new NPP.
- Zn injection may be beneficial to reduce margin to SCC of steam generator tubing.

2.1.5.3 Noble metal chemistry addition

The reduction of the BWR stainless steel structural components' sensitivity to IGSCC has been accomplished by the application of HWC, the addition of up to 1.8 ppm hydrogen to the coolant which in turn lowers the ECP by the recombination of the oxidants. Crack initiation and propagation will not occur at the target level of $<-230V_{SHE}$. The efficiency of the recombination process was improved by the addition of the noble metals platinum (Pt) and sometimes rhodium (Rh) that catalyze the recombination at significantly lower hydrogen levels (0.15-0.30 ppm), resulting in a potentially more effective function within the cracks and a reduction in the Main Steam Line Radiation Dose (MSLRD) due to ^{16}N promoted by the higher hydrogen contents. This Noble Metal Chemical Addition (NMCA) is a process patented by GE.

Initially the Pt additions, in solution form, have been added batchwise at the End Of Cycle (EOCs), and more recently an On-Line NMCA (OLNC) process has been initiated that injects the noble metal periodically during the operating cycle. Three OLNC applications have been made at KernKraftwerk Mühleberg (KKM) in Switzerland between 2005 and 2007 and the first application in the US was at a BWR2 in 2006 and at Peach Bottom 3 (BWR4) in 2007. All proceeded according to plan as described in the references listed below.

Since a significant fraction of the noble metals deposit on the fuel, the effect on fuel performance has been under investigation. The effect on crack propagation in core structural components, while shown successfully in the laboratory, is still being monitored in plant applications. *No data have been reported on any fuel examinations or in-plant crack propagation in the past two years that were exposed to NMCA or OLNC.* However, there have been several excellent papers published that cover the protective mechanism of the process and the results of plant applications on water chemistry, deposition rates and operations:

- The effect of environmental variables that control the size and distribution of noble metal particles deposited on structural material surfaces and in IGSCC cracks, Kim et al., 2007(a).
- The second and third application of OLNC at KKM are described including the effects on water chemistry, dose rate reduction, Pt concentration, ECP decrease measurements, Hettiarachchi et al., 2008.
- The process and effects of OLNC application to Peach Bottom 3 are described including changes in water chemistry, radiation levels and radiochemistry. A comparison is made between the OLNC effects and the effects of NMCA in Peach Bottom 2, Odell et al., 2008.
- A summary paper describes the basic mechanisms of NMCA and OLNC and the ex-reactor experimental bases supporting the proposed mechanisms, Andresen & Diaz, 2007.

The last ANT International reports on the effect of NMCA on fuel performance were in Section 8.2.1 of the ZIRAT10/IZNA5 Annual Report and Section 6.2 of the LCC1 Annual Report. In summary there have been no fuel failures directly attributable to NMCA. CRUD flaking observed has been attributed to concurrent Zn injection which had an effect on the CRUD morphology. A 4 ppb Zn FeedWater (FW) limit had been recommended by EPRI Guidelines to prevent this effect. Samples of Pt coupled directly to Zircaloy-2 and irradiated early in the program showed that oxidation of the Zircaloy could occur by galvanic corrosion. For this reason noble metal injection is delayed in the cycle until the fresh fuel has had a chance to form a protective oxide coating. An interesting ex-reactor experimental project pursued this issue by investigating the effects of various Pt loadings and distributions on Zr and Zircaloy-2 oxidation, Anghel et al., 2008.

The experiments reacted a variety of Pt deposit size and distributions with Zr foil and liner type Zircaloy-2 at 400°C in air. The Zr foil samples included hydrogen contents of (ppb) <5, 200, 400 and 600 as well as a sample of Zr foil pre-oxidised with 1 µ thick oxide. The increased oxidation due to the presence of Pt was confirmed for most of the combinations of Pt loading and hydrogen contents in most cases as exemplified in Figure 2-13. However, a decrease in oxidation rate was found for some combinations, such as in the initial hours of the test with 5 µgPt/cm² and 400 ppm hydrogen. This led to a search for the critical combination that will enhance oxidation resistance. An example of this is shown in Figure 2-14 which led the authors to conclude that the critical Pt coverage for Zr with 200 ppm hydrogen is about 4 µg/cm². *While such islands of beneficial effect of Pt on oxidation resistance may exist, there is no practical way to control the hydrogen content and Pt deposit levels in practice.*

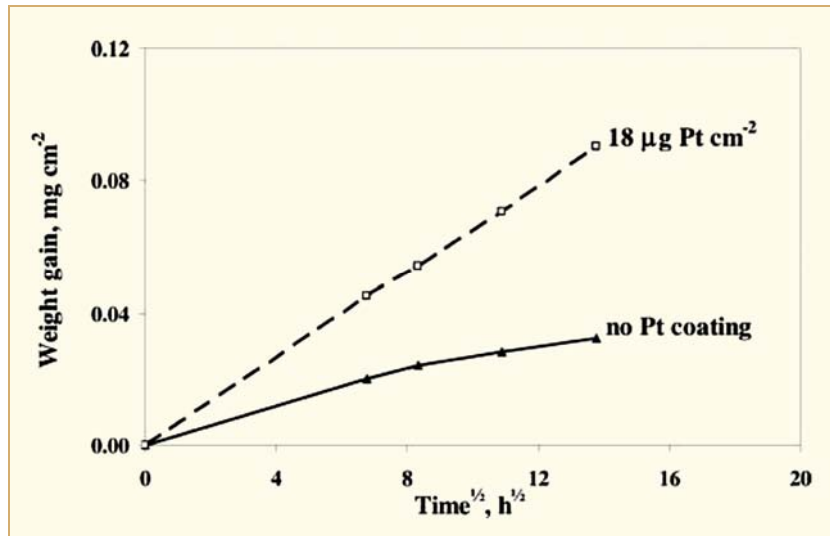


Figure 2-13: Influence of Pt surface coverage (18 µg Pt cm⁻²) on the oxidation of peroxidised (1 µm oxide thickness) Zr (<5 wt ppm H) in 1 atm air at 400°C, Anghel et al., 2008.

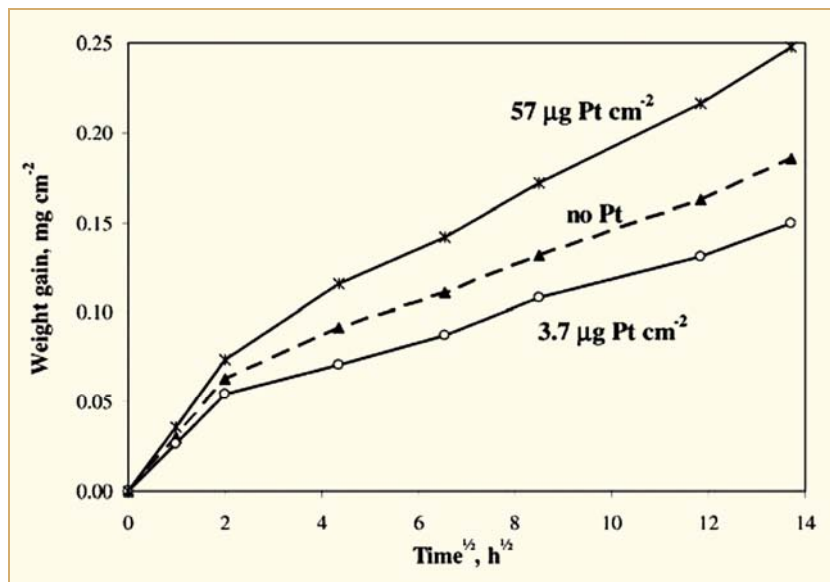


Figure 2-14: Pt influence on the oxide growth on Zr with 200 wt ppm H in 1 atm air at 400°C, Anghel et al., 2008.

3 **Zirconium alloy manufacturing and alloy system**

There were limited number of results related to zirconium alloy manufacturing and alloy system published in the past twelve months. This topic will be covered in greater depth in the *ZIRAT₁₄/IZNA₉* Annual Report (AR).

4 Mechanical properties (Ron Adamson)

4.1 Introduction

The mechanical properties of essentially two different components are normally treated in this section. First, the *LWR* fuel assembly and, second, the *Pressure tubes* in Canadian Deuterium Uranium (*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 perform satisfactorily during the lifetime of the reactor.

Delayed Hydride Cracking (*DHC*), is a failure mechanism that may limit the lifetime of *CANDU* and Reaktor Bolshoi Mozhnosti Kanalov (*RBMK*) pressure tubes, and this mechanism is therefore treated in the pressure tube section. *DHC* is a fracture mechanism that may result in 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*.

In a *CANDU* reactor the cold-worked Zr-2.5Nb pressure tubes operate at temperatures between about 250 and 310°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 $4 \cdot 10^{17} \text{ n m}^{-2} \text{ s}^{-1}$.

The pressure tubes used in a *CANDU* reactor are made from Zr-2.5Nb. The tubes are extruded at 815°C CW 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-centered-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 β -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.

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 (AOO) (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 accidents 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 criteria 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 taken 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 mechanically 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 specifies 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 sufficient 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 reflooding¹ phase during LOCA and due to Pellet Cladding Mechanical Interaction (PCMI)² during a RIA transient.

Fuel vendors have developed codes to model the fuel assembly mechanical performance during class I, II, III and IV situations. To be able to do the modelling correctly, data on mechanical performance of the fuel assembly must exist. The data are generated in two types of tests, either separate effect tests or integral tests. The former test studies only the impact of one parameter at a time on the mechanical performance, see e.g. Adamson & Rudling, 2001. 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.

¹ 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.

² Pellet Cladding Mechanical Interaction, i.e., interaction without the influence of iodine (that would instead result in PCI).

4.1.1 Summary from *ZIRATI1/IZNA6* (Ron Adamson)

Review of the late-2005 to late-2006 literature pertinent to the mechanical properties of zirconium alloys has resulted in the following observations:

- The book *Zirconium in the Nuclear Industry, 14th International Symposium*, American Society for Testing and Materials (ASTM) STP 1467, Peter Rudling and Bruce Kammenzind, editors, ASTM International, West Conshohocken, PA, 2005, was issued in 2006. It contains the final version of papers presented at the Stockholm meeting of June 2004. Most of the papers were reviewed in the *ZIRATI1/IZNA5* Annual Report, so just brief comments were given here to allow the reader to easily look up the papers in the book.
 - Some new evidence of hardening due to irradiation-induced *SPP* dissolution was presented by Russian investigators. This again points to the need to understand microstructural evolution if irradiation effects are to be predicted.
 - *GNF-Ziron* (Zr1.46%Sn.26%Fe.10%Cr.05%Ni) and High Fe-Ni (Zr1.43%Sn.26%Fe.10Ni) are being introduced as alloys to resist corrosion and hydriding at high burnup. These alloys are reported as having mechanical properties very similar to standard Zircaloy.
 - Experimental techniques to obtain fracture toughness values that are appropriate for thin walled Zircaloy are being developed and reported.
 - Several papers gave similar results in terms of effects of hydrides and test temperatures on fracture toughness. The effect of “radially” aligned hydrides at the crack tip in the CT specimen technique does appear to, as expected, have a more pronounced effect on fracture toughness at temperatures below about 127°C (400K). At higher temperatures the zirconium matrix is sufficiently ductile to deform around fractured hydrides, avoiding brittle linked-hydride fracture and resulting in a more stable crack growth and higher fracture toughness.
 - A few basic creep mechanism investigations are in progress which relate to interim dry storage of spent fuel rods. Work conducted at university and private laboratories are making some progress in understanding creep mechanisms in unirradiated Zircaloy.
 - Tensile properties of a de-fuelled cladding tube containing high hydride levels were determined over a wide range of temperatures. It was confirmed that strength decreases with increasing temperature, with a sharp decrease occurring above 400°C (673K) where rapid annealing of radiation damage occurs. Ductility reflected not only the irradiation damage annealing, but also the effect of temperature on hydrided-material ductility.

4.2 Effects of hydrides

4.2.1 Previous work

The definition of a hydride “rim” or “blister” or “lens” is not crisp; however, in all cases the terms refer to a high concentration of hydrides near the surface (normally the outer surface of fuel cladding) of the specimen or component being examined. Blisters and lens are local concentrations, particularly when compared to the gage length of the specimen being tested; a large lens was defined by Hermann et al., 2007 as $>3 \text{ mm}^2$ when projected on the cladding surface, or a few mm long. A “rim” is a high hydrogen concentration over an extended area or length, many mm long. The depth and shape is also important, as is hydride concentration. A low concentration would be several hundred ppm H, while a “solid” hydride would be 10000-16000 ppm. A “rim” is usually a constant thickness, while the others are lenticular in shape.

5 Dimension stability

There were limited number of results related to zirconium alloy manufacturing and alloy system published in the past twelve months. This topic will be covered in greater depth in the *ZIRAT₁₄/IZNA₉ AR*.

6 Corrosion and hydriding

6.1 Zirconium alloy corrosion and cracking (Brian Cox)

6.1.1 Metal and oxide structures

The “rate controlling step” in Zr alloy oxidation and corrosion is the electron conductivity of the oxide film. The highest resistivity oxides are found on Van Arkel (high purity) zirconium oxidised in a dry atmosphere. The presence of impurities and alloying elements that are incorporated in the oxide film decreases the oxide resistivity and increase the oxidation rate. Interaction with high temperature water or steam will degrade the protective oxide film and also increase the corrosion rate. Knowledge of where these species are located in the oxide, and their effects on the oxide properties are, are therefore, important to an understanding of Zr alloy corrosion behaviour. It is also necessary to consider the recoil of uranium and fission products that may penetrate the protective oxide film and affect corrosion and cracking processes. During in-reactor corrosion the interaction of water chemistry and the deposition of *CRUD* on the cladding can also affect cladding corrosion.

Sawicki, 2008 has reported detailed studies of the chemistry of deposits found on fuel cladding following an AOA in the Callaway PWR. He found extensive layers of monoclinic-zirconia in the *CRUD* layers that suggested that they had been formed by zirconia dissolution and reprecipitation, such dissolution processes in the protective ZrO_2 film on the cladding should be considered for in-reactor behaviour.

Garcia-Barriocanal et al., 2008 have drawn attention to another factor that needs to be examined – a possible effect of small particle size in the Zirconia protective film on the electronic resistivity of the zirconia. If the Zirconia particle size is decreased into the nanometer range (as may happen with some thin oxide films) the large increase in electronic conductivity may occur. This could have a corresponding effect of increasing the corrosion rate significantly. Irradiation effects may contribute to this, and a phase difference (e.g. monoclinic/tetragonal) is needed.

6.1.2 Zirconium alloy corrosion out-reactor

New work on corrosion of zirconium alloys published this year can be subdivided according to sub-topics.

6.1.2.1 Effects of water chemistry additives

Yao et al., 2008 studied Zircaloy-4 specimens with different heat treatments exposed at 360°C to 0.01 N, LiOH in a static autoclave in either the as-received (SRA) condition or after a 1020°C β -quench. These samples were compared with two ZIRLO batches (8; 18). The corrosion results are shown in Figure 6-1 and demonstrate the protective effect of a β -quenching treatment on the corrosion resistance in LiOH solutions in which the SRA material behaves very poorly. The authors also examined the fractography of the oxide films produced by partially dissolving the Zr matrix in a HNO_3/HF pickling solution and breaking off the free-standing oxide for SEM examination. The authors blame much of the microporosity seen on the inner oxide surface to attack by the pickling solution. However, they did not examine the outer surfaces of the same oxide samples where similar pitting should be observed if it resulted from the pickling solution. Thus, their conclusions should be treated with caution.

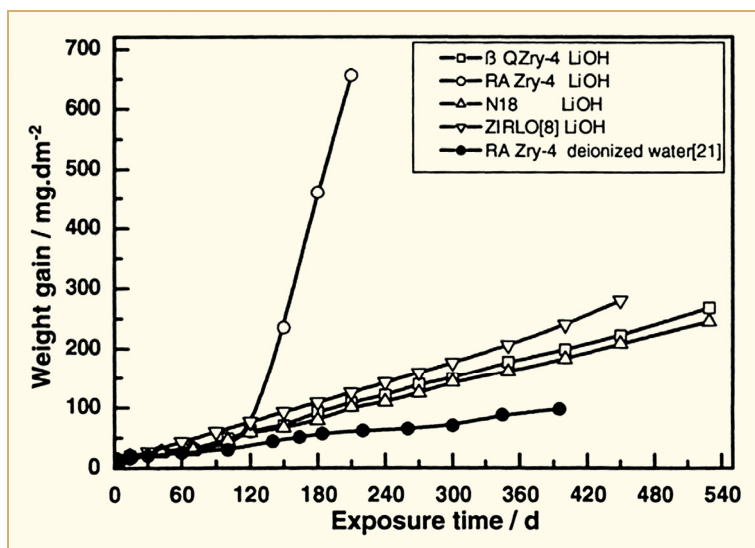


Figure 6-1: Corrosion weight gain vs. exposure time of βQ Zry-4 and RA Zry-4 specimens tested in 0.01 M LiOH aqueous solution at 360°C/18.6 MPa. The weight gain vs. time of ZIRLO [8] and N18 tested in the same condition as this study and RA Zry-4 tested deionised water at 360°C/18.6 MPa [21] is also presented, Yao et al., 2008.

Bojinov et al, 2008 studied the effects of KOH in VVER water on the oxidation of E110 and Zircaloy-4 by Electrochemical Impedance Spectroscopy (EIS). KOH contents of 11, 28 and 56 ppm were compared at 310°C over a 5 day period. Results at the highest KOH concentration (56 ppm) for E110 and Zircaloy-4 are presented as conductivity $|Z|$ and phase-angle in Figure 6-2. Differences in both conductivity and phase angle between E110 and Zircaloy-4 appear to be small but both alloys show increasing corrosion rates with increasing KOH content. The data are all analysed on the basis that the migration of oxygen-ion vacancies in the oxide is the rate determining step, and no consideration of electron conduction is considered. The authors conclude that the effects of increasing KOH concentration are related to increases in the dissolution rates of the oxide films.

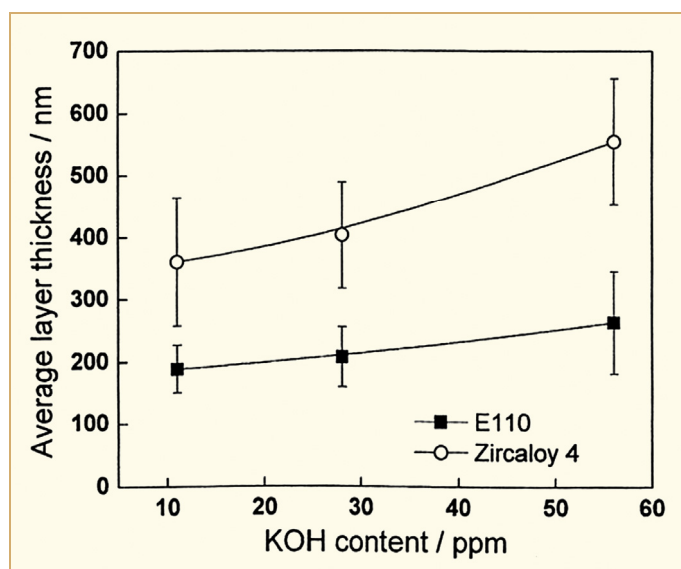


Figure 6-2: Microscopic estimates of the oxide film thickness on E110 and Zircaloy- 4 after 120 h of exposure to simulated VVER water as depending on the KOH content, Bojinov et al., 2008.

6.1.2.2 Effects of niobium additions on corrosion

A numbers of studies of the effects of niobium content on the corrosion of zirconium alloys have been published this year, most of them emanating from South Korea (Kim et al., 2008(a)) studied a range of niobium additions between 0.01 and 2.0 wt%Nb. The corrosion results for static autoclave tests at 360°C/18.9 MPa involved specimens that had been solution treated at 1020°C for 20 minutes, quenched, hot rolled after pre-heating for 20 minutes at either 570 or 640°C, and cold-rolled twice to a final thickness of 0.8 mm. They were pickled in a nitric/hydrofluoric acid solution (N.B. without any sulphuric acid addition). Results are given in Figure 6-3. They show a minimum in the corrosion rates for 0.2-0.39%Nb alloys; severe corrosion of alloys heat treated at 640°C (i.e. above the monotectoid temperature (α - β transformation temperature, and only a small “bump” at 0.8%Nb for alloys annealed at 570°C (just below the monotectoid temperature). Without adding sulphuric acid to the pickling solution it would be difficult to remove niobium oxide film deposits from specimen surfaces. The minimum in the corrosion curves at about 0.2-0.3%Nb agrees with Russian data on similar alloys presented at the 2nd U.N. Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Ambartsumjan et al., 1958.

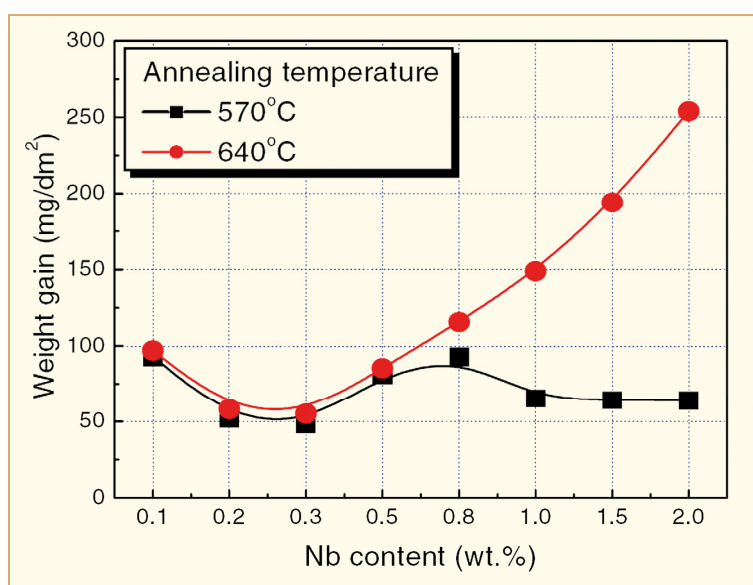


Figure 6-3: Corrosion behaviours of the Zr-xNb alloys with the Nb content and annealing temperature corroded at 360°C for 150 days, Jeong et al., 2003.

Park et al., 2008(c) studied Zr-1.5Nb-0.4Sn-0.2Fe-0.1Cr alloys in 360°C pure water with LiOH additions. This alloy (known as *HANA-4*) has shown good corrosion resistance compared with the Zircalloys. Much of this work has already been reported in *ZIRAT9/IZNA4* & *ZIRAT10/IZNA5* Annual Report (ARs), Sections 7 and *ZIRAT11/IZNA6* & *ZIRAT12/IZNA7* ARs, Sections 6, Adamson et al., 2004, 2005, 2006(b) and 2007. This year's paper seems basically to report on extensions of corrosion tests reported previously to 1050 days for the “pure water” tests and to 850 days for the tests in LiOH (70 ppm Li). Micrographs of specimen sections do not show any unusual behaviour compared with those presented previously. Another Korean paper: Kim et al., 2008(a) presents TEM studies of the oxide films on a Zr-0.4%Nb alloy corroded in 360°C water at 18.9 MPa. They identified a mixture of columnar (monoclinic) and equiaxed (tetragonal) zirconia structures. These phase identifications appear to be the reverse of those obtained by Yilmazbayhan et al., 2006 and *ZIRAT11/IZNA6* AR, Section 7, page 8, Adamson et al., 2006(b). At present there is no explanation for this difference. The dimensions of the two oxide phases also seem to be very different in the two studies.

7 Primary failure and secondary degradation – open literature data (Peter Rudling)

The open literature data are provided in the following sections.

7.1 Introduction

7.1.1 Primary Failures

During reactor operation, the fuel rod may fail due to a primary cause such as fretting, pellet-cladding interaction (*PCI*), manufacturing defects, corrosion, etc., Table 7-1.

Table 7-1: Primary failure causes for *LWR* fuel during normal operation and AOO.

Primary Failure Cause	Short Description
Excessive Corrosion	An accelerated corrosion process results in cladding perforation. This corrosion acceleration can be generated by e.g., <i>CRUD</i> deposition (<i>CILC</i> ¹²), Enhanced Spacer Shadow Corrosion, <i>ESSC</i> , ¹³ (in <i>BWRs</i>), dry-out due to excessive fuel rod bowing.
Manufacturing defects	Non-through-wall cracks in the fuel cladding developed during the cladding manufacturing process. Defects in bottom and/or top end plug welds. Primary hydriding due to moisture in fuel pellets and or contamination of clad inner surface by moisture or organics. Too large a gap between the fuel rod and the spacer grid supports (poor spacer grid manufacturing process) leading to excessive vibrations in <i>PWR</i> fuel causing fretting failures. Chipped pellets may result in <i>PCI</i> failures both in liner and non-liner fuel
<i>PCI</i>	<i>PCI</i> —an iodine assisted <i>SCC</i> phenomenon that may result in fuel failures during rapid power increases in a fuel rod. There are three components that must occur simultaneously to induce <i>PCI</i> and they are: 1) tensile stresses—induced by the power ramp, 2) access to freshly released iodine—occurs during the power ramp, provided that the fuel pellet temperature becomes large enough and 3) a sensitised material—Zircaloy is normally sensitive enough for iodine stress-corrosion cracking even in an unirradiated state.
Cladding collapse	This failure mechanism occurred due to pellet densification. This failure mode has today been eliminated by fuel design changes and improved manufacturing control.
Fretting	This failure mode has occurred due to: Debris fretting in <i>BWR</i> and <i>PWR</i> Grid-rod fretting - Excessive vibrations in the <i>PWR</i> fuel rod causing fuel failures. This situation may occur for example due to different pressure drops in adjacent fuel assemblies causing cross-flow. Baffle jetting failures - Related to unexpectedly high coolant cross-flows close to baffle joints.

¹² Crud Induced Localised Corrosion – an accelerated form of corrosion that has historically resulted in a large number of failures in *BWRs*. Three parameters are involved in this corrosion phenomenon, namely: 1) Large Cu coolant concentrations as a result of e.g., aluminium brass condenser tubes, 2) Low initial fuel rod surface heat flux – occurs in Gd rods and 3) Fuel cladding that shows large initial corrosion rates- occurs in cladding with low resistance towards nodular corrosion.

¹³ This corrosion phenomenon resulted recently in a few failed rods. The mechanism is not clear but seems to be related to galvanic corrosion. This corrosion type may occur on the fuel cladding in contact or adjacent to a dissimilar material such as Inconel. Thus, this accelerated type of corrosion occurred on the fuel cladding material at spacer locations (the spacer springs in alloy *BWR* fuel vendors fuel are made of Inconel). Water chemistry seems also to play a role if the fuel cladding material microstructure is such that the corrosion performance is poor. Specifically coolant chemistry with low Fe/(Ni-Zn) ratio seems to be aggressive (provided that the cladding material shows poor corrosion performance. A fuel cladding material with good corrosion resistance does not result in *ESSC*, enhanced spacer shadow corrosion, even in aggressive water chemistry.

The failure statistics during 1980-2007 in US PWRs and BWRs are shown in Figure 7-1 and the percentage of failures by mechanism during 2000-2007 is provided in Figure 7-2.

Most of the US BWR failure cases are related to CRUD-accelerated corrosion failures. Other US BWR failure cases involved six plants, which experienced PCI-like failures following control rod moves. Debris fretting also remains a problem even after the introduction of debris filters.

In PWRs the primary contributor to failure rates remains grid-to-rod fretting; however, experience with new grid designs appears to be promising. During the last years it was noteworthy that some PCI-suspect failures were also experienced at three B&W-designed PWR plants following the movement of Axial Power Shaping Rods (APSRs) even though their calculated stress levels remained within the permissible range.

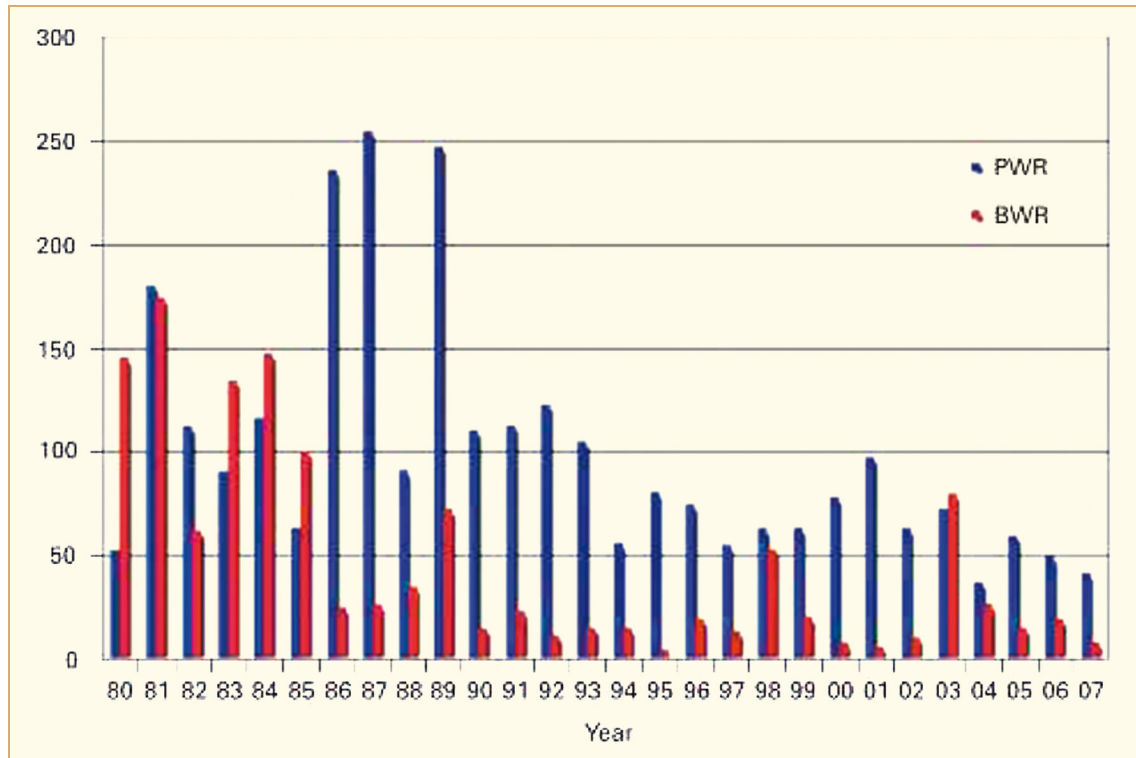


Figure 7-1: Fuel failures in US PWRs and BWRs, 1980-2007, Tompkins, 2008.

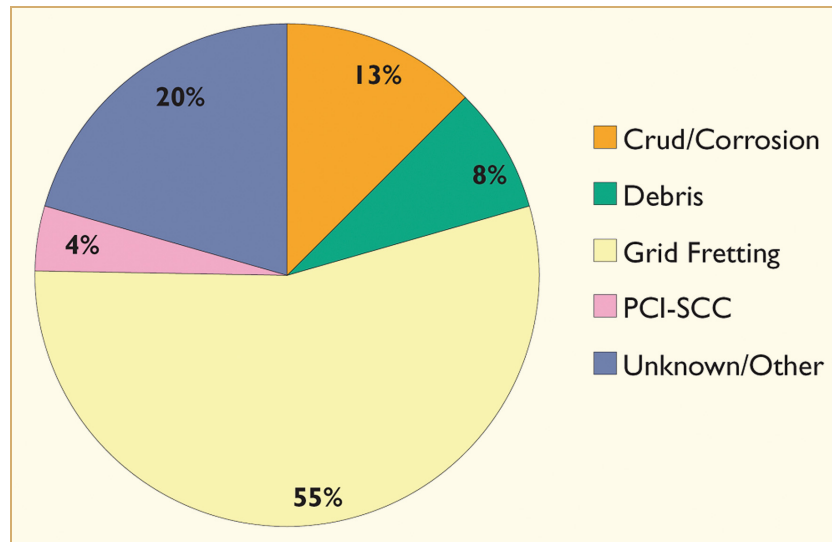


Figure 7-2: Percentage of fuel failures by mechanism for *PWRs* and *BWRs*, 2000-2007, Tompkins, 2008.

In the European BWRs debris fretting is one of the major failure causes. Debris filters in the BWR fuels do not eliminate the debris fretting failures while the debris filters seems to be efficient in the PWRs.

Table 7-2 and Table 7-3 provide key data for some of the most recent fuel-failure cases.

Table 7-2: Summary of previous *PWR* failure key events, see previous *ZIRATI/IZNA*-reports for details.

Nuclear unit	Type of primary failure	Comment
<i>TMI-1</i> , Cy 10, 1995	Nine high peaking fuel rods, Zr-4 Cladding, failed after 122 days of operation. <i>CRUD</i> /corrosion related failures	<ul style="list-style-type: none"> All failed and degraded pins reportedly had (<i>DCP</i>) Distinctive <i>CRUD</i> Pattern¹⁴ High peaking factors, thermal-hydraulic conditions. Calculations indicated that no boiling should have occurred on the pins with <i>DCP</i>, although the pins with <i>DCP</i> were calculated to have a slightly higher temperature. Water chemistry (low pH at <i>BOC</i>, pH < 6.9, max LiOH 2.2 ppm) Some, <i>AOA</i> effect was found reaching a maximum in the middle of cycle 10. The source of the <i>CRUD</i> could not be determined. The <i>CRUD</i> sampling showed that the nickel-to iron ratio was in the range 1.25 to 16.7, which was reportedly somewhat lower than in previous investigations.
Seabrook, Cy 5, 1997	Five one-cycle <i>ZIRLO</i> rods failed. <i>CRUD</i> /corrosion related failures	<ul style="list-style-type: none"> Longer cycle in transition to 24-month cycle Possibly <i>CRUD</i>-induced overheating resulting in substantial nucleate boiling.

¹⁴ This acronym implies that the fuel inspection revealed crud deposits on the fuel rod and that the deposits were uneven in the rod circumference

8 *LOCA/RIA*

There were limited number of results related to zirconium alloy manufacturing and alloy system published in the past twelve months. This topic will be covered in greater depth in the *ZIRAT₁₄/IZNA₉ AR*.

9 Fuel performance during intermediate storage (Alfred Strasser)

9.1 Introduction

Most countries that generate nuclear power are in the process of developing criteria, designs and sites for the permanent disposal of spent nuclear fuel, but they have yet to become licensed realities. Meanwhile the pools at the nuclear plant sites are filling up with spent fuel and the utilities are transferring the spent fuel from the pools to dry cask storage sites that are located, mostly at the plant sites. Exceptions are the central, large intermediate pool facilities that serve all the plants in Sweden (CLAB facility) and all the plants in Finland (KPA-STORE). The lack of a licensed permanent fuel repository in any country has placed total reliance on intermediate storage. As a result the dry storage technology has become a major activity and business component of today's back-end fuel strategies.

The US nuclear plants will run out of pool storage space in about 2013 and as a result all the plants have or are planning to have dry storage facilities, or Independent Spent Fuel Storage Installations (ISFSIs). As of the beginning of 2008 there were 50 ISFSIs in 30 States (Figure 9-1) with 6 to 8 new ISFIs planned for 2008-2009 and 9 to 11 in 2010-2011, Brach, 2008.

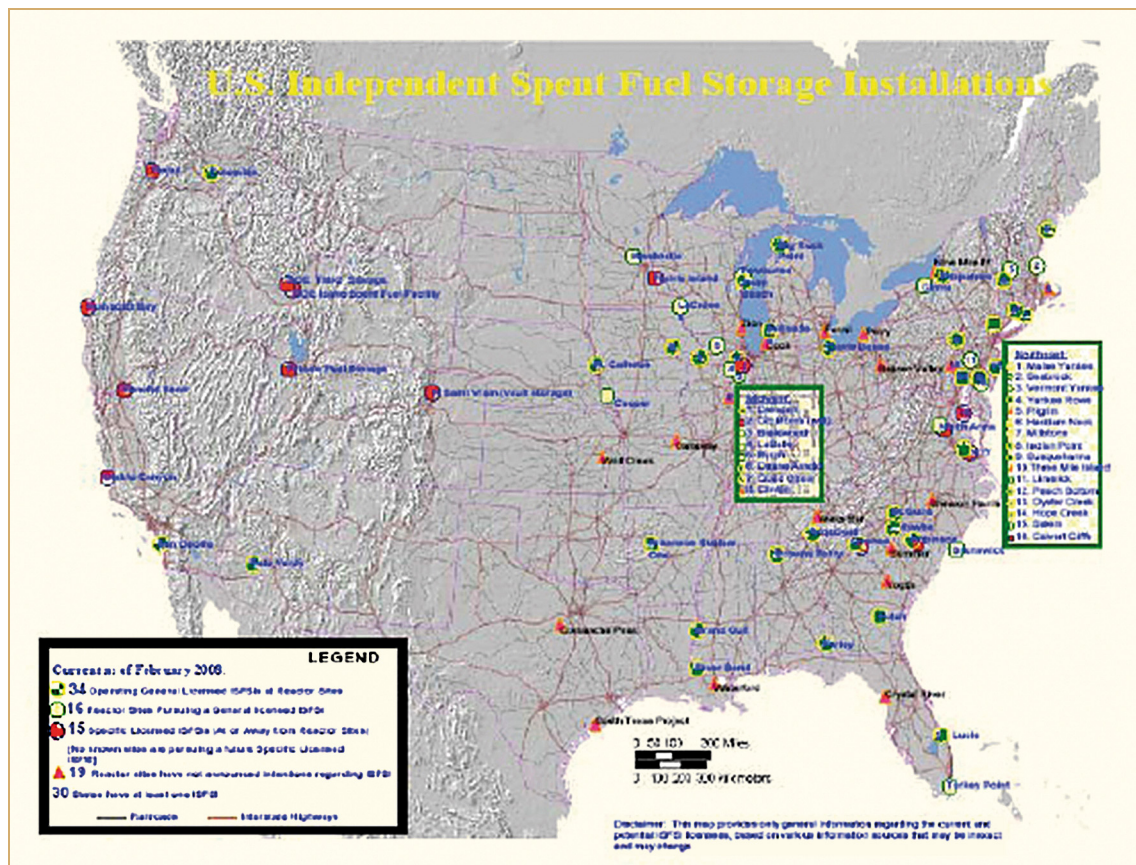


Figure 9-1: ISFI locations – 2008, Kellar, 2008.

As of the beginning of 2008 the total amount of spent fuel in the US was about 58000 tons of heavy metal, of which 10500 tons were stored in close to 1000 dry storage casks. The projection for 2017 is a total of over 22000 tons (MTU) of fuel stored in 2000 casks at 68 sites for 108 plants, Kraft, 2008.

The permanent repository is planned to be in *Yucca Mountain* about 90 miles (145 km) from Las Vegas, Nevada. Its planned capacity is about 63000 tons of fuel, so that the spent fuel from wet and dry storage would fill it up by 2009. Fuel from dry storage alone would fill it by about 2038 assuming extended operation of current reactors, so that planning for a second site is being considered.

A cross-section of the repository is shown in Figure 9-2 and indicates that the fuel will be stored about 1500 feet (450 m.) underground. The fuel will be contained in horizontal tunnels in a canister that is in a cask or waste package as shown in the lower right hand part of Figure 9-2. An enlarged view of the manmade section of the repository is shown in Figure 9-3. The drip shield to protect the waste package from even an unlikely water source, is made of titanium. The canister will hopefully be a universal one that will be loaded with fuel at the spent fuel pool of the reactor site and transported to the permanent repository without having to reload the fuel into another canister. This canister, currently under development, is called *TAD* for Transportation, Aging and Disposal, described in Section 9.9.3.

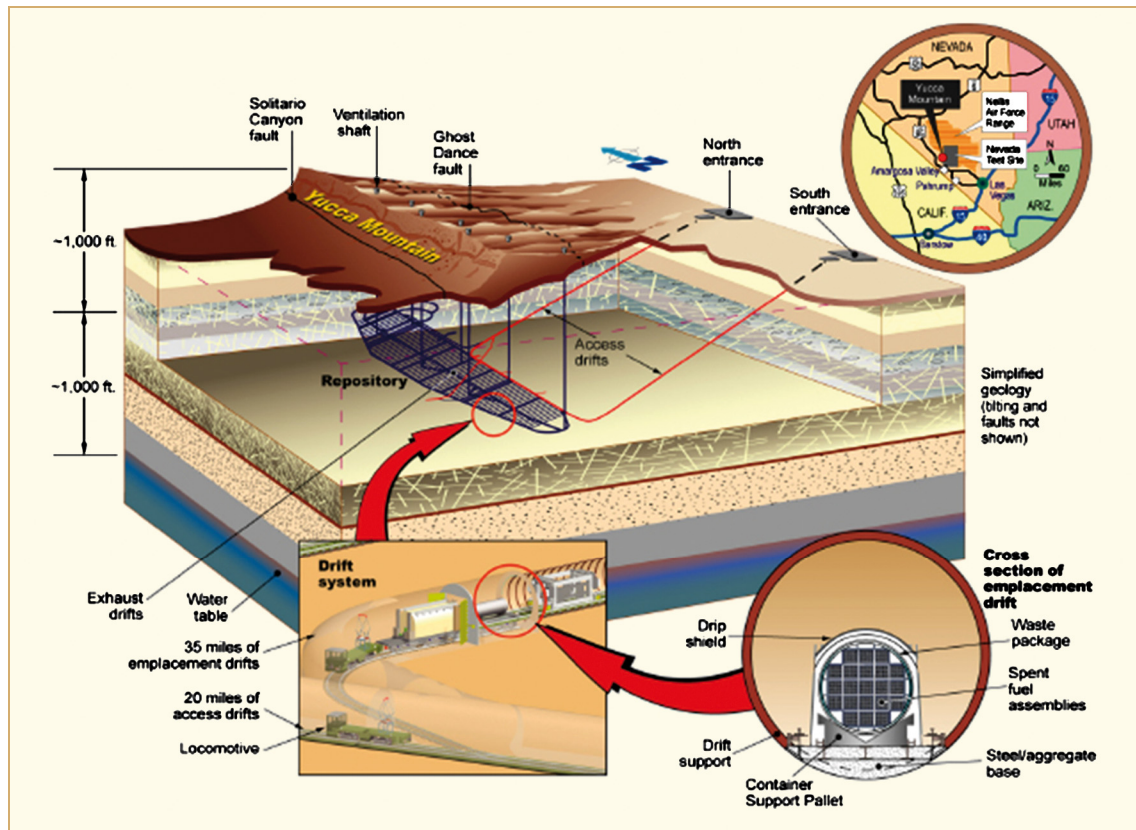


Figure 9-2: YUCCA Mountain permanent storage concept, Sproat, 2007.

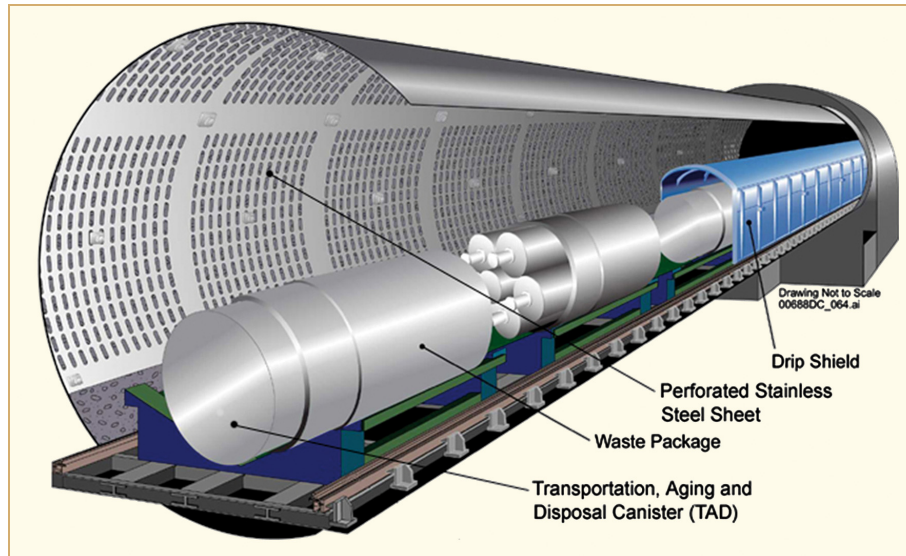


Figure 9-3: Man-made features of the Yucca Mountain repository, Office of Civilian Radioactive Waste Management (OCRWM, 2008).

The licensing process for the Yucca Mountain Repository has been initiated by the Department Of Energy's (DOE) issuance of a License Application (LA) in June 2008 which was reviewed and then docketed by the NRC in September, 2008. "Docketing" means that the NRC has accepted the application for review. A simplified flow chart for the process up to the potential NRC approval for construction is shown in Figure 9-4. The NRC review is followed by hearings by the Atomic Safety and Licensing Board (ASLB), other government agencies and the public. Final approval by the NRC must be preceded by the approval of the ASLB. To get this far Congress must approve sufficient funds for this project – a highly political move. One should note that the State of Nevada is very much against this project in their state and has spent many dollars lobbying against it. The schedule for this process is at least 3, but more likely 4 years, or completion by 2013. If construction can start at that time, unobstructed by appeals, the optimistic estimates for initial fuel receipt are 2018, but NRC's estimate is 2020. When the federal government took over this project in 1983, they estimated fuel receipt in 1998.

The delay in a permanent storage site combined with spent fuel pools filled to capacity have resulted in the development of dry storage as a significant, well regulated industry as noted by the statistics above. In addition to the at-reactor dry storage sites, central intermediate storage sites have been considered. In the US an intermediate Away-From-Reactor (AFR) dry storage site, in the Skull Valley Indian Reservation, has actually been approved by the ASLB and the NRC subsequently issued a construction license in 2006. The Department of Interior approval that is also needed was not given, however, as they denied the rail access to the site and they also denied the lease between the Skull Valley Band of Goshute Indians and the consortium of utilities that formed the Private Fuel Storage Group (PFS). The case is now being appealed in the US District Court in Salt Lake City, Martin S, 2008.

10 Potential burnup limitations (Peter Rudling)

10.1 Introduction

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

- Corrosion limits of Zry-4 in high power *PWRs*, are extended by the alternate use of improved cladding alloys. 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. However, this modified operation will in most cases result in higher corrosion duty of the zirconium materials. Thus, it is believed that the corrosion may always be limiting for plant operation even with the new type of alloys. Furthermore, the influence of *CRUD* on corrosion may increase with increasing duty.
- Bowing of *PWR* fuel assemblies contributed in part by irradiation growth, creep and hydriding of Zry-4, has been reduced by improved guide tube materials (i.e., lower irradiation growth and hydriding rates), reduced assembly hold-down forces, and other mechanical/thermomechanical design changes, but not yet finally eliminated.
- 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.
- *RIA* and *LOCA* related burnup licensing limits are in the process of being assessed by additional experimental data and analyses. It would appear that the current *LOCA* limits are sufficiently conservative for fuel burnups up to 75 MWd/kgU. The *RIA* limits (threshold enthalpies) may continue to decrease as a function of burnup due to the increase in clad corrosion and hydrogen uptake.
- 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.

10.2 Corrosion and mechanical properties related to oxide thickness and H pickup

- *BWRs*: increased uniform and shadow corrosion, oxide thickness spalling, increased hydrogen pickup, and formation of radial hydrides due to longer residence time, higher power, modern power histories, and water chemistry changes. Current crucial issues are: shadow corrosion mechanisms its effect on channel bow, late increased corrosion and hydrogen pickup of Zry-2 at high burnups, and formation of radial hydrides, *CRUD*-chemistry-corrosion interaction, effect of water chemistry impurities, as well as specific effects of *NMCA* with or without Zn-injection.
- *PWRs*: increased uniform corrosion, oxide thickness, spalling, and new Zr alloys due to longer residence time and higher Li, higher power, more boiling. The development of new Zr-alloys is still ongoing. The data base for several of the new alloys is still limited. Zr-Nb alloys are occasionally affected by accelerated corrosion due to surface contaminations and/or boiling. Welding of the new alloys may need improved processes (Zr-Nb alloys) and chemical compositions between dissimilar metals such as e.g. *ZIRLO* and Zry-4 may result in inferior corrosion resistance. Luckily, the corrosion temperatures at these elevations in the core are significantly lower than the peak temperatures.
- Decreased ductility and fracture toughness as consequence of the increased hydrogen pickup during any situation (e.g., *RIA*, *PCMI*, *LOCA* and post-*LOCA* events, seismic event, transport container drop-accident conditions).
- Increased growth due to higher hydride volume and thick oxide layers.

- Increased corrosion due to impact of hydrides at the cladding outer surface.
- Impact of corrosion and hydrogen pickup on creep behaviour of fuel claddings during class 1-IV events and during intermediate storage.
- Increased effects of irradiation and hydrides on the fracture toughness of thin-walled zirconium alloy components.

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*, *MDA*, *NDA*, and *M5/Zr1Nb*. *DX* is another alternative that may be necessary to achieve satisfactory mechanical properties.
- Zr-alloys such as Optimised *ZIRLO*, Modified *MDA*, *S2*, Modified *E635* with reduced Sn content in comparison to the original composition of *ZIRLO*, *MDA*, *NDA* and *E635* are being explored.
- 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 at high burnups.
- Modification of manufacturing processes (to get optimum sized, more stable *SPPs*).
- *Zry-4* fuel channels for controlled positions
- Improved alloys under development.

Improved water chemistry and *CRUD* control.

10.3 Dimensional stability

- Increased dimensional changes of components and differential dimensional changes 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:
 - Decreased thermal margins (*LOCA* and dry-out).
 - Control rod insertion difficulties (safety issue).

Most sensitive component

Potentially all zirconium alloy components, but currently Zircaloy *PWR* guide tubes and *BWR* channels. 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, E635 (Anikuloy).
- Modified mechanical design to provide lower hold-down forces, stiffer assemblies, etc.
- Refinement of the understanding of the effects of corrosion and hydriding at moderate and high burnups.
- 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 might be improved by an appropriate final heat-treatment in the alpha-phase. Also applies to BWR materials control of texture (close to ideal isotropy) is critical for good performance at high burnups).

Increase margin for *BWR*

- Uniform microstructure and texture throughout the flow channel.
- Use of lower growth material, such as beta-quenched material in as-fabricated step, NSF or other Nb-modified zirconium alloys.
- Channel management programs, including assessment of degree of control over specific reactor periods.
- More corrosion and shadow-corrosion resistant material in channels and spacers.
- Increased understanding of basic phenomena driving the channel bow process, including flux and hydrogen-driven processes.
- Lowered hydride pickup and increased uniformity of hydride distribution in channels and spacers through heat treatment and alloy choice.

10.4 *PCI in BWRs and PWRs*

Increased sensitivity for *PCI/PCMI* due to:

- More fission products (I, Cs and Cd) produced.
- Increased Transient Fission Gas Release, *TFGR*.
- Formation of a rim zone at the pellet periphery at fuel pellet average burnup of about 50 MWd/kgU.
- Fuel-clad bonding at fuel pellet average burnup of about 50 MWd/kgU.
- Increased swelling.
- Increased transient fission gas swelling.
- Microstructural changes in fuel clad material and in the liner (barrier) in the case of liner (barrier) BWR fuel.
- Development of a hydride rim at the clad outer surface (that may promote *PCMI* failures).
- Fuel design changes resulting in up-rated duty on the clad.

Most sensitive component

Fuel cladding

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