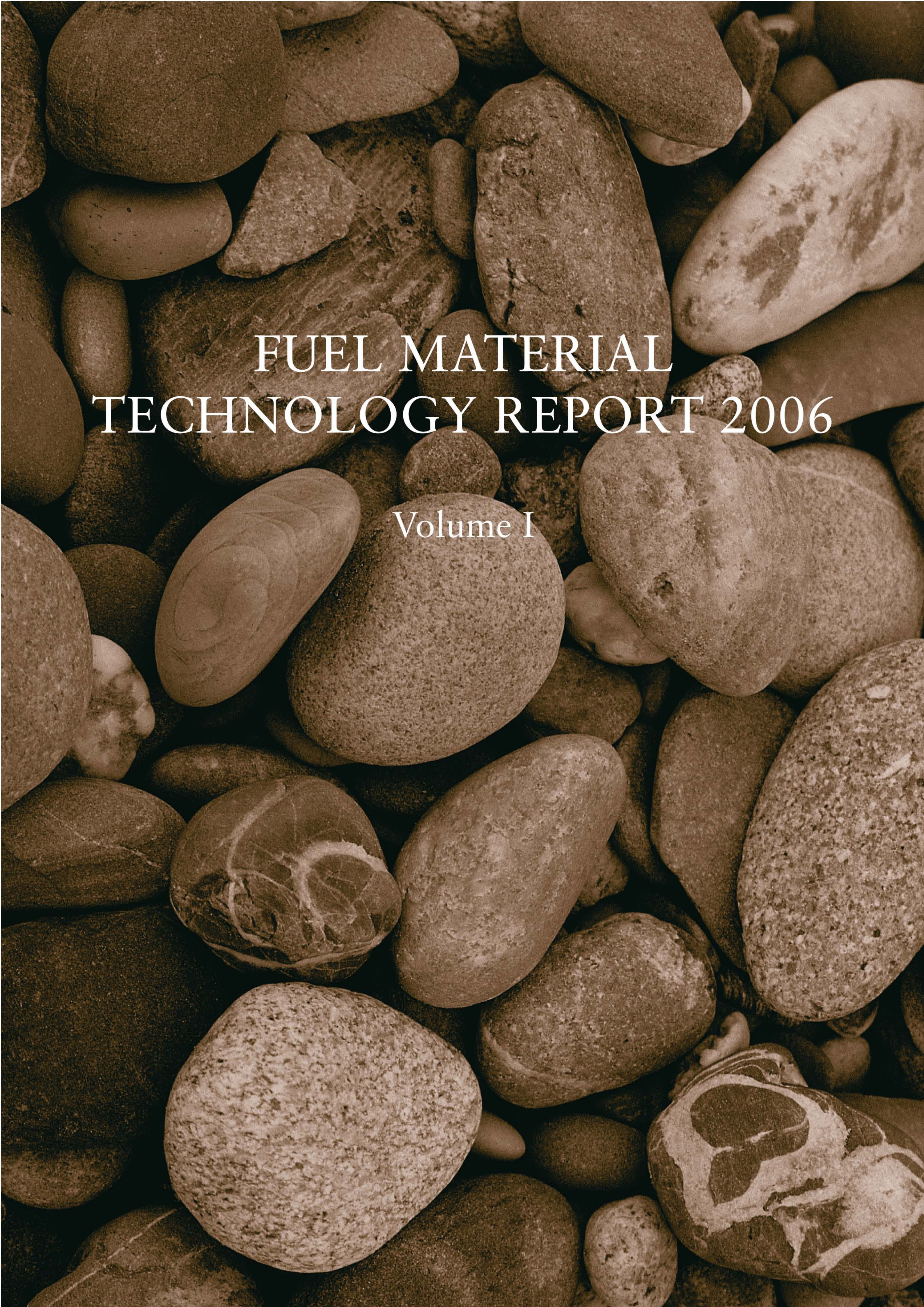

FUEL MATERIAL TECHNOLOGY REPORT 2006

Volume I

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ABBREVIATIONS

ABSS	AB Sandvik Steel
ABWR	Advanced BWR
ADU	Ammonium DiUranate
AECL	Atomic Energy of Canada Laboratories
ANT	Advanced Nulear Technology
AOA	Axial Offset Anomaly
AOD	Axial Offset Deviation
AP	Advanced *PWR*
ASTM	American Society for Testing and Materials
ASME	American Society of Mechanical Engineers
AUC	Ammonium Uranyl Carbonate
B&W	Babcock&Willcox
BCC	Body Centered Cubic
BOL	Beginning of Life
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium
CE	Combustion Engineering
CEZUS	Companie Europen Ziconium Ugine Sandvik
CF	Corrosion Fatigue
CGR	Crack Growth Rate
CILC	Crud Induced Localized Corrosion
CR	Control Rod
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CRUD	Chalk River Unidentified Deposits
CPR	Critial Power Ratio
DF	Debris Filter
DFBN	Debris Filter Bottom Nozzle
DHC	Delayed Hydride Cracking
DNB	Departure from Nucleate Boiling
DNBR	Departure of Nucleate Boiling Ratio
DO	Dissolved Oxygen
DX	Duplex
EAC	Environmentally Assisted Cracking
EB	Electron Beam
ECP	Electrochemical Corrosion Potential
ELS	Extra-Low Sn
EOL	End Of Life
ESSC	Enhanced Spacer Shadow Corrosion
FA	Fuel Assembly

FC	Fuel Channel
FDI	Fuel Duty Index
FGR	Fission Gas Release
GC	Guide Channels
GE	General Electric
GNF	Global Nuclear Fuel
GT	Guide Tube
HAZ	Heat Affected Zone
HCP	Hexagonal Close-Packed
HPA-4	High Performance Alloy
HPUF	Hydrogen Pick Up Fraction
HTP	High Thermal Performance
HWC	Hydrogen Water Chemistry
HWR	Heavy Water Reactor
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
I.D.	Inner Diameter
IFBA	Integral Fuel Burnable Absorber
IFM	Intermediate Flow Mixing
IGSCC	Intergranular Stress Corrosion Cracking
IPHT	In Process Heat Treated
IRI	Incomplete Rod Insertion
IT	Instrument Tube
IZNA	Information on Zirconium Alloys
KKM	KernKraftwerk Mühleberg
KNFC	Korea Nuclear Fuel Company
KWU	KratwerkUnion
L	Laser
LAS	Low Alloy Steels
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LPBN	Low Pressure drop Bottom Nozzle
LTP	Low-Temperature Process
LWR	Light Water Reactor
MA	Mill Annealed
MDA	Mitsubishi Developed Alloy
MF	Magnetic Force
MHI	Mitsubishi Heavy Industries
MOX	Mixed OXide
MRP	Material Reliability Program
NDA	New Developed Alloy
NFI	Nuclear Fuel Industries

NMC	Noble Metal Chemistry
NPD	Nuclear Power Demonstration reactor
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NG	Nuclear Grade
NWC	Normal Water Chemistry
O.D.	Outer Diameter
OPG	Ontario Power Generation company
PCI	Pellet Cladding Interaction
PCMI	Pellet Cladding Mechanical Interaction
PGP	Precipitate Growth Parameter
PWR	Pressurised Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
QC	Quality Control
R	Resistance
RBMK	Reaktor Bolshoi Mozhnosti Kanalov (in English Large Boiling Water Channel type reactor)
RCCA	Rod Cluster Control Assemblies
RCS	Reactor Coolant System
RFA	Robust Fuel Assembly,
RIA	Reactivity Initiated Accident
RPV	Reactor Pressure Vessel
RT	Room Temperature
RVH	Reactor Vessel Head
RXA	Recrystallised Annealed
SCC	Stress Corrosion Cracking
SCRI	Stress-Corrosion Susceptibility Index
SG	Steam Generator
SGHWR	Steam Generating Heavy Water Reactor
SICC	Strain Induced Corrosion Cracking
SLAR	Spacer Location and Repositioning
SOCAP	Second Order Cumulative Annealing Parameter
SS	Stainless Steel
SSM	Sandvik Special Metal
STR	Special Topic Report
SRA	Stress Relieved Annealed
SPP	Second Phase Particle
TEM	Transmission Electron Microscopy
TGSCC	Transgranular Stress Corrosion Cracking
TIG	Tungsten Inert Gas

TM	Transition Metal
TSHT	Tube Shell Heat Treated
TSS	Terminal Solid Solubility
TT	Thermal Treatment
UK	United Kingdom
US	Ultimate Strength
UT	Ultrasonic Testing
VVER	Voda Voda Energo Reactor
WABA	Wet Annular Absorber
WZ	Western Zirconium
ZIRAT	Zirconium Alloy Technology (Program)
ZIRLO	Zirconium Low Oxidation

UNIT CONVERSION

TEMPERATURE

°C + 273.15 = K

°C*1.8 +32 = °F

T(K)	T (°C)	T(°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**

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√m	ksi√inch
0.91	**1**
1	1.10

FOREWORD

The personal viewpoints and conclusions presented in the report that are beyond those quoted from references, are those of the individual authors and may not represent the collective view of all authors.

Throughout the report, references are given to previous Special Topical Reports furnished in the *ZIRAT*-5/*IZNA*-1 to *ZIRAT*-10/*IZNA* –5 Programs, that are available to *ZIRAT*-customers, and they are:

- "Impact of Irradiation on Material Performance" by Ron Adamson and Brian Cox, 2005.
- "Structural Behavior of Fuel and Fuel Channel Components" by Brian Cox, Friedrich Garzarolli, Peter Rudling, Alfred Strasser, 2005.
- "Corrosion of Zr-Nb Alloys" by Brian Cox, Friedrich Garzarolli and Peter Rudling, 2004.
- "Loss of Coolant Accidents, *LOCA* and Reactivity Initiated Accidents, *RIA*, in *BWR*s and *PWR*s" by Ron Adamson, Friedrich Garzarolli and Peter Rudling, 2004.
- "High Burnup Fuel Issues" by Ron Adamson, Brian Cox, Friedrich Garzarolli, Peter Rudling, Alfred Strasser, Gunnar Wikmark, 2003.
- "The Effects of Zn Injection (*PWR*s and *BWR*s) and Noble Metal Chemistry (*BWR*s) on Fuel Performance - an update" by Brian Cox, Friedrich Garzarolli, Peter Rudling, Alfred Strasser, 2003.
- "Corrosion of Zirconium Alloys" by Ron Adamson, Brian Cox, Friedrich Garzarolli, Peter Rudling, Alfred Strasser, Gunnar Wikmark, 2002.
- "Dimensional Stability of Zirconium Alloys", by Ron Adamson and Peter Rudling, 2002.
- "Water Chemistry and Crud Influence on Cladding Corrosion", by Gunnar Wikmark and Brian Cox, 2001.
- "Mechanical Properties of Zirconium Alloys", by Ron Adamson and Peter Rudling, 2001.
- "Manufacturing of Zirconium Alloy Materials", by Peter Rudling and Ron Adamson, 2000.
- "Hydriding Mechanisms and Impact on Fuel Performance", by Peter Rudling and Brian Cox, 2000.

Ron Adamson, Editor

CONTENTS

1 INTRODUCTION

Nuclear fuel is at the heart of nuclear power technology.

It is not only the largest part of the operating costs of a nuclear plan; it also contains the most hazardous part of the plant and faces some of the greatest technological challenges.

Operation of 4 meter long 9-10 mm diameter fuel pins for 4-8 years at boiling or near boiling conditions in an intense field of radiation requires a careful design, wise material selection and qualified supervision. Operating conditions create unique water chemistry environments, transformation and transmutation of the fuel materials at extreme temperatures, and fission products which are highly radioactive and some times gaseous.

The scientists and engineers responsible for this product and for its success in producing safe and environmentally friendly energy must be highly qualified and knowledgeable about materials, water- and radio-chemistry, corrosion, irradiation effects and integral behaviour of the nuclear fuel assemblies.

At a time when the nuclear industry faces a number of challenges, two will put special demands on the transfer and presentation of knowledge in this specialized field. First, circumstances are now creating a new start for the nuclear industry globally. As a consequence, recruitment of new engineers and scientists to cover the needs of this complex and demanding industry is of high priority.

Second, the people who created this industry and gained the experience of the first 20-30 years of research and development are now retiring or have already retired. This threatens to create a large gap in the knowledge and experience which are essential for the new generation to be able to fulfil its part in the revival of nuclear power.

ANT International has for several years worked intensely to bridge this gap in the field of nuclear fuel technology. We have felt the need of a comprehensive document which would cover all aspects of nuclear fuel (and to some extent core components) its materials, design and operation experience, in a manner which would be useful both to the experienced engineer or scientist and the coming generation of specialists.

In order to be able to produce this document in four volumes at the quality level and with an educational value to many audiences *ANT* International has joined forces with some of the most experienced and outstanding experts in the field of nuclear fuel worldwide, as follows:

Dr. Ron Adamson, Zircology Plus

Prof. Brian Cox, University of Toronto

Mr. Friedrich Garzarolli, Erlangen

Dr. Rolf Riess, Erlangen

Mr. Peter Rudling, *ANT* International

Mr. Stig Sandklef, *ANT* International

Mr. Alfred Strasser, Aquarius Services

This first volume covers short general outlines of the designs of *PWR*, *BWR*, *CANDU*, *VVER* and *RBMK* reactors a with focus on the corresponding fuel assembly and supporting structure designs together with the rational for the selection of the materials used in the different applications. This volume also gives an overview of in-reactor fuel performance, of fuel performance codes and the manufacturing process of the fuel assembly.

The effect of radiation on the fuel materials is highly complex and specific for the nuclear industry as is the interaction between materials and cooling water chemistry in the radioactive environment. These effects are treated in a systematic and detailed manner for all types of water cooled nuclear fuel and associated materials in volume two (planned to be issued in 2007).

The importance of nuclear fuel examination before and after irradiation, for either proven or new designs and materials, cannot be overestimated. It is a source of information needed to control the operation, verify the safety limits and to facilitate further development. The technology employed is unique and requires special knowledge both to use and to understand the results. The fourth volume of the report (planned to be issued in 2008) is therefore focused on fuel examination technology to ensure that new specialists can benefit from todays experience and build on it for the future.

The one component in the core that interacts most with nuclear fuel and where the respective designs are most interdependent is the power and reactivity control element. This interaction has great importance for the safe operation of the plants and the fuel and we have dealt with control elements in volume three (planned to be issued in 2008) to get the same systematic and detailed descriptions as for the fuel itself.

Cooperation with the experts named above has extended for several years and together we have produced a number of reports and given numerous seminars to the nuclear fuel industry within the ZIRconium Alloy Technology, *ZIRAT* and Information on ZircoNium Alloys, *IZNA*. The interested reader can get additional information from the following *ZIRAT/IZNA* reports published by *ANT* International and referenced throughout the Fuel Material Technology Report:

ZIRAT 10/*IZNA* 5 Special Topic Report on *Impact of Irradiation on Material Performance* by Ron Adamson and Brian Cox 2005.

ZIRAT 10/*IZNA* 5 Special Topic Report on *Structural Behaviour of Fuel and Fuel Channel Components* by Friedrich Garzarolli, Brian Cox, Alfred Strasser and Peter Rudling 2005.

ZIRAT 9/*IZNA* 4 Special Topic Report on *Corrosion of Zr-Nb Alloys* by Brian Cox, Friedrich Garzarolli and Peter Rudling 2004.

ZIRAT 9/*IZNA* 4 Special Topic Report on *Loss of Coolant Accidents, LOCA, and Reactivity Initiated Accidents, RIA, in BWRs and PWRs* by Peter Rudling, Friedrich Garzarolli and Ron Adamson 2004.

ZIRAT 8/*IZNA* 3 Special Topic Report on *High Burnup Fuel Issues* by Ron Adamson, Friedrich Garzarolli, Peter Rudling, Gunnar Wikmark and Alfred Strasser 2003.

ZIRAT 8/*IZNA* 3 Special Topic Report on *The Effects of Zn Injection (PWRs and BWRs) and Noble Metal Chemistry (BWRs) on Fuel Performance - An Update* by Brian Cox, Friedrich Garzarolli, Peter Rudling and Alfred Strasser 2003.

ZIRAT 7/*IZNA* 2 Special Topic Report on *Corrosion of Zirconium Alloys* by Ron Adamson, Brian Cox, Friedrich Garzarolli, Peter Rudling, Alfred Strasser and Gunnar Wikmark 2002.

ZIRAT 7/*IZNA* 2 Special Topic Report on *Dimensional Stability of Zirconium Alloys*, by Ron Adamson and Peter Rudling 2002.

ZIRAT 6/*IZNA* 1 Special Topic Report on *Water Chemistry and CRUD Influence on Cladding Corrosion*, by Gunnar Wikmark and Brian Cox 2001.

ZIRAT 6/*IZNA* 1 Special Topic Report on *Mechanical Properties of Zirconium Alloys*, by Ron Adamson and Peter Rudling 2001.

ZIRAT 5/*IZNA* 1 Special Topic Report on *Manufacturing of Zirconium Alloy Materials*, by Peter Rudling and Ron Adamson 2000.

ZIRAT 5/*IZNA* 1 Special Topic Report on *Hydriding Mechanisms and Impact on Fuel performance*, by Peter Rudling and Brian Cox 2000.

The feedback given from our customers and their input to this project is highly appreciated. We could not have done it without them. Special thanks also to Anette Medin for her meticulous work in editing the contributions from our experts.

We are proud to present this document to the nuclear industry and we truly believe it will assist in making the nuclear industry a continued success in competence, quality and safety.

2 GENERAL REACTOR CHARACTERISTICS

2.1 INTRODUCTION

There are essentially, 5 different types of commercial water cooled reactors, Table 2-1, whose main characteristics are provided in Table 2-2. The reactor types and their characteristics are described more in the following subsections. The water chemistry characteristics of the different reactors will be described more in detail in Vol. 2 of the Fuel Material Technology Report.

Table 2-1: Current status of commercial water cooled reactors of different designs, World Nuclear Industry Handbook, 2006.

	BWR	*PWR*	*CANDU*	*VVER*	*RBMK*
Number of operating reactor units	94	203	37	54	12
Number of reactor units under constructions	5	7	4	7	0

Table 2-2: Design parameters in water cooled reactors, Adamson et al., 2002.

Parameter		**Western type** ***PWR***	***VVER*** **(440/1000) MW**	***CANDU***	***BWR***	***RBMK***
1.	Coolant	Pressurized H_2O	Pressurized H_2O	Pressurized D_2O	Boiling H_2O	Boiling H_2O
2.	Fuel assembly materials (pressure tube materials)	Zr-4, *ZIRLO*, DUPLEX, M5, *MDA*, *NDA*, Inconel, *SS*	E110, E635	Zry-4 (Zr2.5Nb)	Zry-2, Zry-4, Inconel, *SS*	E110, E635 (Zr2.5Nb)
3.	Average power rating, (kW/l)	80-125	83/108	9-19	40-57	5
4.	Fast neutron flux, average, $n/cm^2.s$	6-9E13	5E13/7E13	1.5-2E12	4-7E13	1-2E13
5.	Temperature, °C					
	Average coolant inlet	279-294	267/290	249-257	272-278	270
	Average coolant outlet	313-329	298/320	293-305	280-300	284
	Max cladding *O.D.*	320-350	335/352	310	285-305	290
	Steam mass content, %				7-14	14
6.	System pressure, bar	155-158	125/165	96	70	67
7.	Coolant flow, m/s	3-6**	3.5/6	3-5	2-5**	3.7
8.	Coolant chemistry					
	Oxygen, ppb	<0.05	<0.1		200-400	<20
	Hydrogen (D_2),					-
	ppm	2-4			see section 2.3	
	cc/kg	25-50	30-60	(3-10)		
	Boron (as boric acid), ppm	0-2200	0-1400	-***	-	-
	Li (as LiOH), ppm	0.5-3.5	0.05-0.6	0.6	-	-
	K (as KOH), ppm	-	5-20		-	-
	NH_3, ppm		6-30			
	NaOH, ppm		0.03-0.35			

** Variation from lower to upper part of the core and from plant to plant

*** Not in coolant but in moderator

2.2 *PWR (ALFRED STRASSER)*

Overview of the Primary Circuit

Components

The typical *PWR* reactor primary system shown on Figure 2-1 consists of the reactor vessel with its internals that support the fuel core, piping for the coolant recirculation system, a pressurizer, steam generators, pumps and valves. Auxiliary systems control the boron (B) and lithium (Li) additions and removals as well as a demineralizer, cleanup system for removal of impurities and a system to provide make-up water.

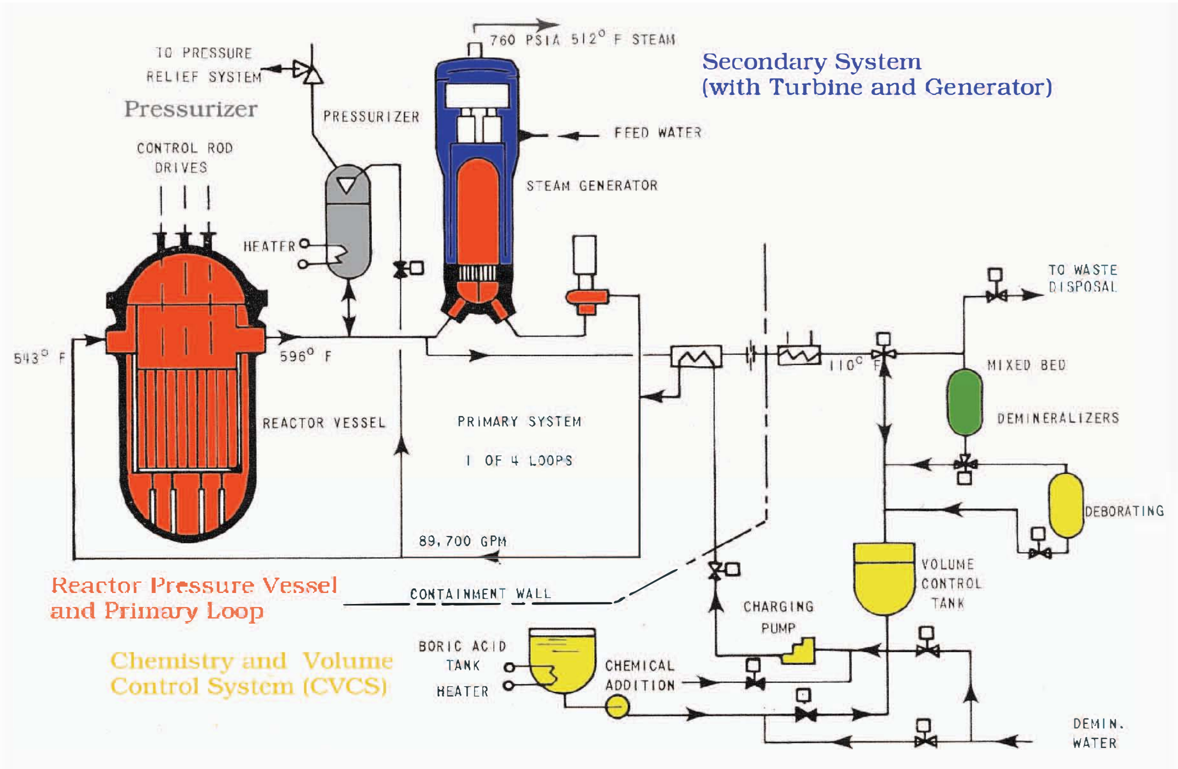

Figure 2-1: Primary Coolant Systems for a Large *PWR*, modified figure according to Cohen, 1985.

The primary system operates at a pressure of 2,250 psi (15.5 MPa) and the *pressure boundary* consists of the reactor vessel, the recirculation piping, pressurizer, steam generator tubing and pumps as shown in Figure 2-2. There are *heat transfer surfaces* within the system that consist of the fuel cladding transferring heat to the primary coolant and the steam generator tubing transferring heat to the secondary coolant. The gamma heating of all the components within the reactor vessel is relatively minor, but is also removed by the coolant.

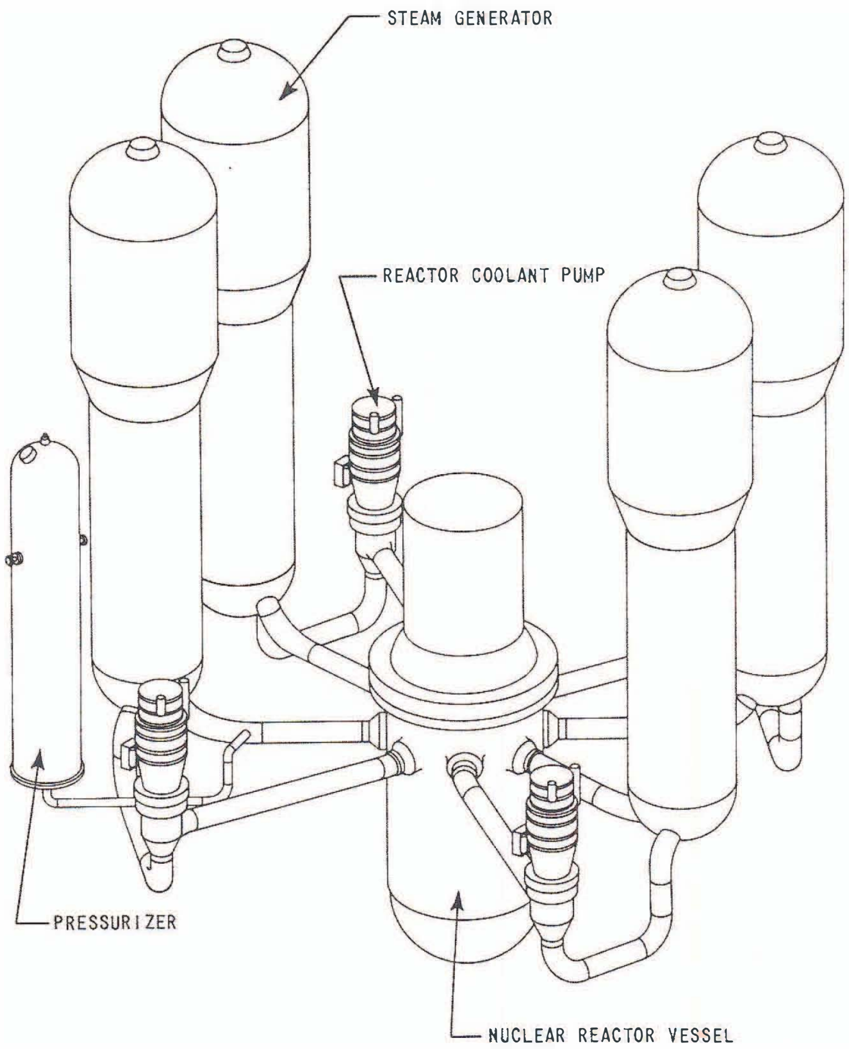

Figure 2-2: *PWR* Primary System – Pressure Boundary, Westinghouse, 2005.

The *structural components* within the reactor include the upper and lower support plates, the core barrel, the core support components and others shown on Figure 2-3. Structural components are used in the pressurizer and steam generator as well, but not all are in contact with the primary coolant.

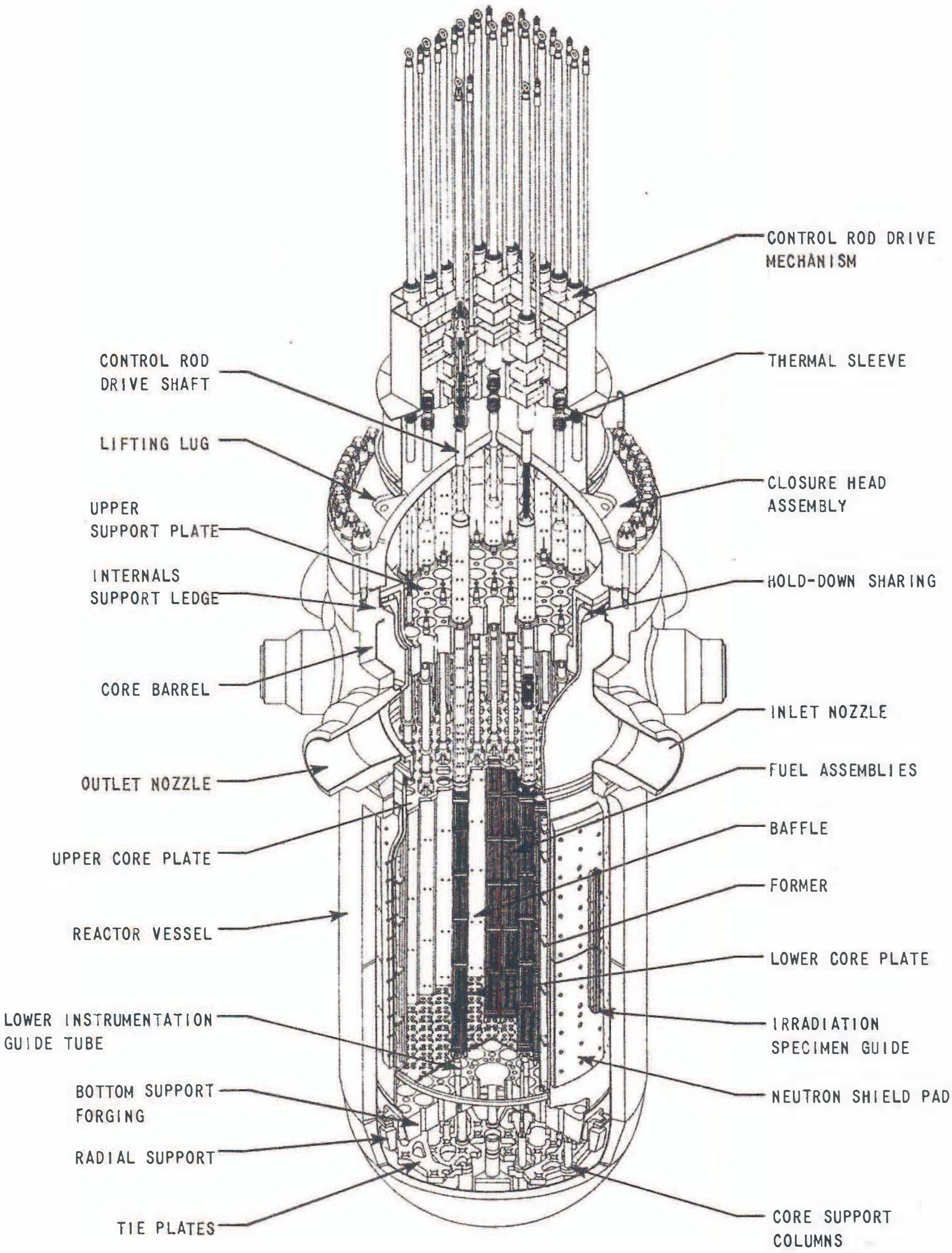

Figure 2-3: *PWR* Pressure Vessel Internals, Westinghouse, 2005.

The *mechanical components* consist of the pump and valves, some of which are quite complex.

The materials selected for the components in contact with the primary coolant are austenitic alloys, that is 18-8 type stainless steels (*SS*) and nickel (Ni) base alloys, because they have excellent uniform corrosion resistance to the normal *PWR* water chemistry. The reactor pressure vessel is made of a high strength low alloy steel, but with *SS* cladding on the inside, in contact with the coolant. Pumps and valves employ some additional high strength, wear and corrosion resistant alloys.

Water Chemistry Environment

The *quality of the water* that is in contact with the materials is specified in several sets of specifications and recommendations:

- The *NSSS* vendor's recommendations and perhaps warranty conditions for the plant materials,
- The fuel vendor's specifications for maintaining the fuel warranties,
- The EPRI Guidelines, --- a good review was made by Fruzzetti et al., 2004.
- The utility's internal guidelines that may or may not be identical to some of the above.

The details of the water chemistry are discussed in Vol. 2 and only the general operating range is given here.

A reducing atmosphere is maintained in the coolant by the addition of about 3 ppm of hydrogen (H). Oxygen limits have been set at <2– 5 ppb.

The pH is controlled by the balance of the B in the form of boric acid and Li in the form of lithium hydroxide in the range of 6.9 to 7.4. The quantity of soluble B shim in the coolant is adjusted to control the reactivity and has been as high as 1800-1900 ppm at the beginning of the cycle and decreased approaching zero at the end of the cycle. The Li content varies to control the desired pH program throughout the cycle and the current maximum practice of about 4 ppm is limited by the potential effect of zirconium alloy cladding corrosion with increased Li levels.

Additions of Zn are made to control activity transport to low levels by its incorporation in the oxide films of the structural materials and piping, a mechanism that reduced cobalt (Co) and Ni transport from the stainless steels. Levels of 4-5 ppb Zn are used for this purpose. Higher levels of 10-20 ppb have been used for reducing stress corrosion cracking of steam generator tubing, but inconsistent results have reduced this practice.

Other impurities are present for reasons other than the purposeful additions, such as:

- Fe, Ni, Cr, Co and potentially other metallic materials in solid or solution form, the product of uniform corrosion of the primary circuit materials,

- Al, Ca, Mg, Si from contamination during maintenance work, impurities in the boric acid or Li additions or other sources such as Boroflex spent fuel racks,

- Cl, F, SO_4, as well as oxidizing agents in ionic forms as the result of leakage from other systems, such as the demineralizers, pump seals, instrument lines, etc.

Many of these impurities were not known and taken into account at the time the plant materials were selected and perhaps even if known, the same materials may have been chosen. As a result of experience with these impurities new recommended limits have evolved, some of which are noted below.

The impurities with the most significant effect on plant materials corrosion, specifically *SCC*, are the halides (Cl, F) and SO_4. The usual limit on halides is <1 ppb. A limit for SO_4 has not been set yet; in fact the analyses for this ion have been limited. Its appearance seems to occur primarily during shutdown.

Limits for the zeolites (Al, Ca, Mg) have been suggested as 5 ppb to prevent their deposit on the fuel and their potential densification of the crud that would increase the thermal resistance of the crud.

Limits on the metallic impurities have not been set; however the transport of Ni and Co is being controlled by Zn injection to limit activity transport, as noted above. The remaining elements will have an effect on fuel crud formation and cladding performance rather than have an effect on plant materials.

The *temperature of the water* is at its highest in the reactor core with core average inlet temperatures of 542-559°F (284-293°C; 557-566K) and outlet temperatures of 602-619°F (318-327°C; 591-600). Hot assembly exit temperatures can go up to 636°F (336°C; 609K). The temperatures in the steam generator are lower and produce steam in the range of 518-535°F (270-280°C; 543-553K).

***SCC* and *IASCC*:** The major life limiting factor of primary circuit materials has been the stress corrosion cracking (*SCC*) and the irradiation assisted *SCC* (*IASCC*) of austenitic alloys that include the 18-8 type stainless steels and the Ni base alloys. The *SCC* affects the components outside the reactor vessel, such as the piping and steam generator materials. The *IASCC* obviously occurs within the reactor, including the vessel itself, where the components are exposed to radiation. The combined presence of four parameters at certain minimum levels, are responsible for *SCC*:

- Oxidizing species in the water,
- A stress and/or strain in the material,
- An *SCC* sensitive microstructure,
- Extended reactor service time.

These and other factors that can enhance or mitigate susceptibility to *SCC* are discussed subsequently. They apply to both *PWR*s and *BWR*s even though in the *BWR*s the oxidizing species and water chemistry are different (pH 5.6-7.5 instead of 6.9-7.4), the coolant temperature is lower by about an average of 75°F (24°C; 24K) and there is extensive bulk boiling in the core instead of limited nucleate boiling.

Reactor Pressure Vessel (*RPV*) and Low Alloy Steels (*LAS*): The *RPV* consists of a welded steel shell and bottom enclosure with a bolted closure head assembly, Figure 2-3. The *RPV* has to contain the reactor pressurization of 2250 psia (15.5 MPa) and resist corrosion by the coolant; as the primary containment for the reactor core, it has to meet the highest reliability standards. The design criteria are met by a duplex material that consists of a low alloy steel (*LAS*) with high strength but insufficient uniform corrosion resistance, clad on the inside with stainless steel (*SS*) with somewhat lower strength, but excellent general corrosion resistance. The *SS* cladding can be prone to *SCC* and *IASCC*, but the *LAS* is much more resistant to *SCC*.

The *LAS* vessel is fabricated by forming and welding plates and the head is generally a forging usually made of the A533 alloy or equivalent. The compositions of *LAS* are given in Table 2-3. The stainless steel cladding, usually Type 308 *SS* is applied by weld overlay and approximately 5 mm (0.1969 in) thick. The compositions of *SS* are given in Table 2-4.

Table 2-3: Low Alloy Steels (*LAS*).

AISI type	Fe	Cr	Ni	Mn	Mo	Si	V	C max.	S max.	Other constituents
	Composition (wt.%)									
Carbon (ferritic) steels										
A106	Balance	-	-	0.3-1.1	-	0.10 min.	-	0.30	0.025	0.025 max (P)
A501	Balance	4-6	-	1.0	0.40-0.65	1.0	-	0.10	0.030	0.040 max (P)
A508/2	Balance	0.35	0.7	0.7	0.6	-	0.05	0.27	-	
A508/3	Balance	-	0.6	1.3	0.52	-	0.05	0.20	-	
A508/4	Balance	1.7	3.3	-	0.5	-	0.03	0.23	-	
A508/5	Balance	1.7	3.3	-	0.5	0.15-0.30	0.1	0.23	-	
A533	Balance	-	-	1.15-1.50	0.45-0.60		0.05	0.25	-	
Low-alloy (bainitic) steels										
ICr-IMo-0.25V	96.07	1.0	-	0.85	1.25	0.25	0.25	0.33	-	
2;Cr-IMo (Grade 22)	95.7	2.42	-	0.49	0.98	0.28	-	-0.026	0.009	0.012 (P) and 0.05 (Cu) Nb in stabilized version
Ni-Cr-MoV (A469 Class 8)	Balance	1.25-2.00	3.25-4.00	0.60	0.30-0.60	0.15-0.30	0.05-0.15	0.28	0.018	0.015 (P)
Ni-Cr-MoV (A470 Class 8)	Balance	0.90-1.50	0.75	1.00	1.00-1 50	0.15-0.35	0.20-0.30	0.25-0.35	0.018	0.015 (P)
NCCr-MoV (A471 Class 8)	Balance	0.75-2.00	2 00-4 00	0.70	0 20-0.70	0.15-0.35	0.05	0.28	0.015	0.015 (P)

Table 2-4: Composition of Stainless Steels.

Designation	Ni	Cr	Fe	Mn max	C max	S max	Si max	P max	Mo	Nb	Ta max
	Composition Percent										
304	8-11	18-20	bal	2.00	0.08	0.03	1.00	0.04			
304L	8-11	18-20	bal	2.00	0.03	0.03	1.00	0.04			
308	10-12	19-21	bal	2.00	0.08	0.03	1.00	0.04			
316	11-14	16-18	bal	2.00	0.08	0.03	1.00	0.03	2.0-3.0		
316L	11-14	16-18	bal	2.00	0.03	0.03	1.00	0.03			
321	17-19	9-12	bal	0.50	0.08	0.03	1.00	0.04	Ti = 5xC min.		
347	9-13	17-20	bal	2.00	0.08	0.03	1.00	0.03		10 x C	
348	9-13	17-20	bal	2.00	0.08	0.03	1.00	0.04		10 x C	0.10
410	0.5 max.	17-20	bal	1.00	0.15	0.03	1.00	0.04			
17-4 PH	3-5	15-18	bal	1.00	0.07	0.03	1.00	0.04	Cu 3-5	Nb + Ta 0.15-0.45	

The materials specified for the Westinghouse Advanced *PWR* (*AP*1000) are listed in Table 2-5. The materials have not changed except that the low C version of Type 308 (0.03% C max.) is proposed instead of Type 308, Westinghouse, 2005.

The major *RPV* issue is with the effect of radiation on the mechanical properties of the steel, such as the ductility, impact resistance and fracture toughness and is independent of water chemistry. The only water chemistry related questions arise in the event of the cracking or failure of the *SS* liner that would allow ingress of the water to the *LAS*. Laboratory tests for *SCC* of the duplex structure have shown that cracks propagating through the *SS* are arrested at the interface with the *LAS*. *SCC* crack growth has only been observed in laboratory tests in oxygenated *BWR* water with very high Cl and SO_4 levels, Seifert et al., 2003.

Table 2-5: Reactor Coolant Pressure Boundary Materials Specifications for the Westinghouse *AP* 1000.

Component	Material	Class, Grade, or Type
Reactor Vessel Components		
Head plates (other than core region)	SA-533 or SA-508	GR B, CL 1 or CL 3
Shell courses	SA-508	CL 3
Shell, flange, and nozzle forgings	SA-508	CL 3
Nozzle safe ends	SA-182	F316LN
Appurtenances to the control rod drive mechanism (CRDM)	SB-167 or SA-182	TP690 or F304LN, F316LN
Instrumentation tube appurtenances, upper head	SB-167 or SA-182, SA312, SA376	TP690 or F304LN, F316LN
Closure studs	SA-540	GR B23 or GR B24, CL 3
Monitor tubes and vent pipe	SA-312 or SA-376 or SB-166, SB-167	TP304LN, TP316LN or TP690
Cladding, buttering, and welds	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, or ERNiCrFe-7
Pressure boundary welds	Low alloy steel	SFA 5.5, 5.23, 5.28
Steam Generator Components		
Pressure plates	SA-533	GR B, CL 1
Pressure forgings (including nozzles and tube sheet)	SA-508	CL 3a
Nozzle safe ends	SA-182	F316LN
Channel heads	SA-508	CL 3a
Tubes	S B- 163	TP690TT
Cladding, buttering, and welds	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, or ERNiCrFe-7
Pressure boundary welds	Low alloy steel	SFA 5.5, 5.23, 5.28
Manway studs/nuts	SA-193, SA-194	GR B7
Pressurizer Components		
Pressure plates	SA-533	GR B, CL 1
Pressure forgings	SA-508	CL 3
Nozzle safe ends	SA-182	F316LN
Cladding, buttering, and welds	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, or ERNiCrFe-7
Pressure boundary welds	Low alloy steel	SFA 5.5, 5.23, 5.28
Manway studs/nuts	SA-193, SA-194	GR B7
Reactor Coolant Pump		
Pressure forgings	SA-182 or SA-336	F304LN, F316LN
Pressure casting	SA-351 or SA-352	CF3A
Tube and pipe	SA-213; SA-376 or SA-312	TP304LN, TP316LN
Pressure plates	SA-240	304LN, 316LN
Closure bolting	SA-193 or SA-540	GR B7 or GR B24, CL 4
Pressure boundary welds	Low alloy steel	SFA 5.5, 5.23, 5.28

Table 2-5: Reactor Coolant Pressure Boundary Materials Specifications for the Westinghouse *AP* 1000, Cont'd.

Component	Material	Class, Grade, or Type
Reactor Coolant Piping		
Reactor coolant pipe	SA-376	TP304LN, TP316LN
Reactor coolant fittings, branch nozzles	SA-376, SA-182	TP304LN, TP316LN
Surge line	SA-376	TP304LN, TP316LN
RCP piping other than loop and surge line	SA-312 and SA-376	TP304LN, TP316LN
Pressure boundary welds	Low alloy steel	SFA 5.5, 5.23, 5.28
CRDM		
Latch housing	SA-336	F304LN, F316LN
Rod travel housing	SA-336	F304LN, F316LN
Welding materials	SFA 5.4 or 5.9	308L, 309L
Valves		
Bodies	SA-182 or SA-351	F304LN, F316LN or CF3A
Bonnets	SA-182, SA-240 or SA-351	F304LN, F316LN, 304LN, 316LN or CF3A
Discs	SA-182, SA-564 or SA-351	F304LN, F316LN or GR 630 or CF3A
Stems	SA-479 or SA-564	F316, F316LN or GR 630
Pressure retaining bolting	SA-453 or SA-564	GR 660 or GR 630
Pressure retaining nuts	SA-453 or SA-194	GR 6 or TP410

The major *RPV* related problem in *PWR*s has been the cracking of the control rod drive mechanism (*CRDM*) nozzles that penetrate the reactor vessel head (*RVH*) shown in Figure 2-4. The tube, or nozzle containing the control rod drive is Alloy 600, welded with Alloy 182 to the Type 308 *SS* clad A533 *RPV* steel. The *RVH* operates between 547 and 605°F (286 – 318°C; 559-591K).

The cracking was identified in the Alloy 600 just above the J-groove weld subsequent to a leak of primary water into the annulus between the nozzle and the upper head. Some of the cracking has extended circumferentially to the root of the J- groove weld where the coolant would depressurize, boil, flash to steam and could concentrate its impurities. The water chemistry potentially present in the annulus and crack were evaluated, White et al., 2003 and some of the conclusions reached were:

- The corrosion potential of Alloy 600 would be the same as in *NWC*, since a reducing atmosphere would be maintained in this location,
- A corrosion potential close to the Ni/NiO equilibrium would be present, the key chemical parameter that determines *SCC* susceptibility,
- The most likely environment would be normal primary water or hydrogenated steam, both without O penetration.

The reactor heads have been and are being replaced to correct the problem and the replacements are substituting Alloy 690 for 600 and the weld Alloy 182 may be replaced by Alloy 152 for improved *SCC* resistance.

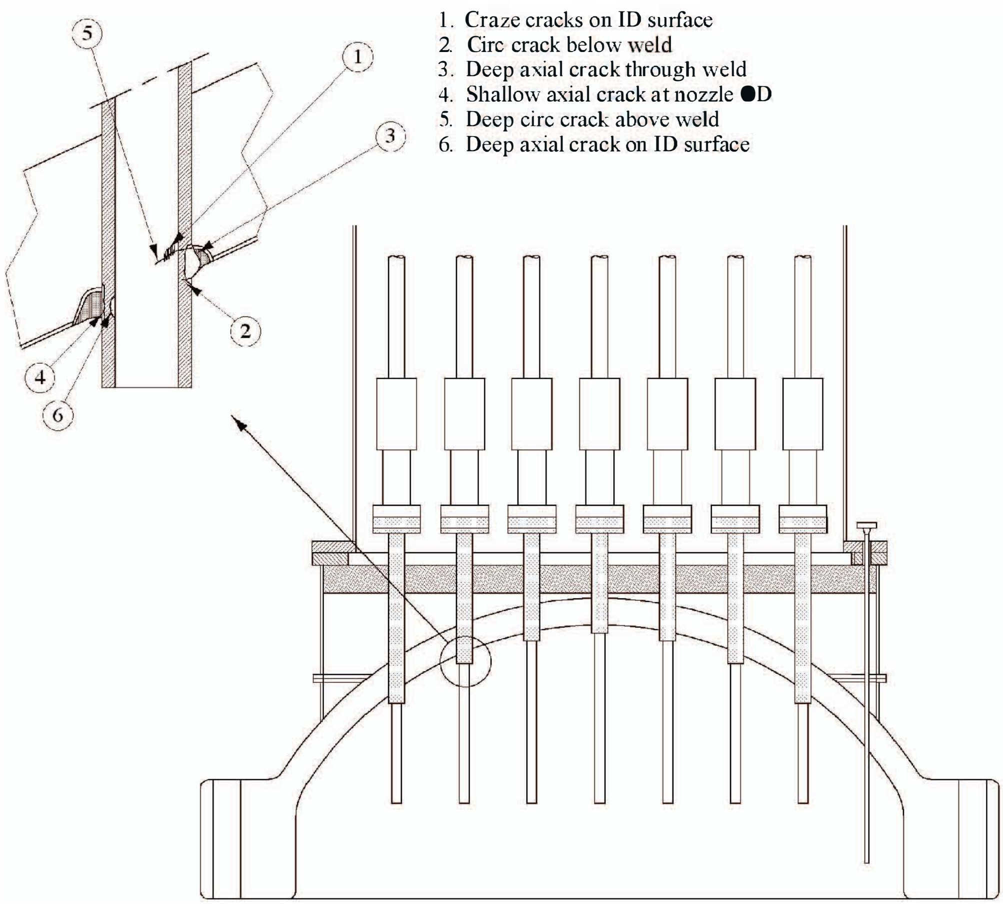

Figure 2-4: Location Of Typical *PWR* Reactor Vessel Closure Head Nozzle *PWSCC* (Alloy 600 *CRDM* Nozzle Shown with Alloy 182 J-groove Weld and Buttering, Low Alloy Steel Shell, and Stainless Steel Cladding), White et al., 2003.

An extensive international program involved both analytical and experimental evaluation of the crack growth rates as well as the evaluation of plant experience in the *US* and Europe. The objective was to develop a model for *CGR* and life expectancies of cracked Alloy 600 nozzles and other Alloy 600 components in *PWR*s with the exception of the thin walled steam generator tubes, White et al., 2003.

The high sensitivity of crack growth rates to temperature was the reason for standardizing and normalizing all data at 327°C (617°F; 600K), the most likely temperature in this region. The large heat to heat scatter and the limited data per heat make the interpretation of the dependence of the *CGR* on the stress intensity factor difficult and for those manipulations I leave the reader to the reference. The curve for *CGR* vs. the stress intensity factor, K, developed by the EPRI Material Reliability Program (*MRP*) with much of the data base is shown in Figure 2-5.

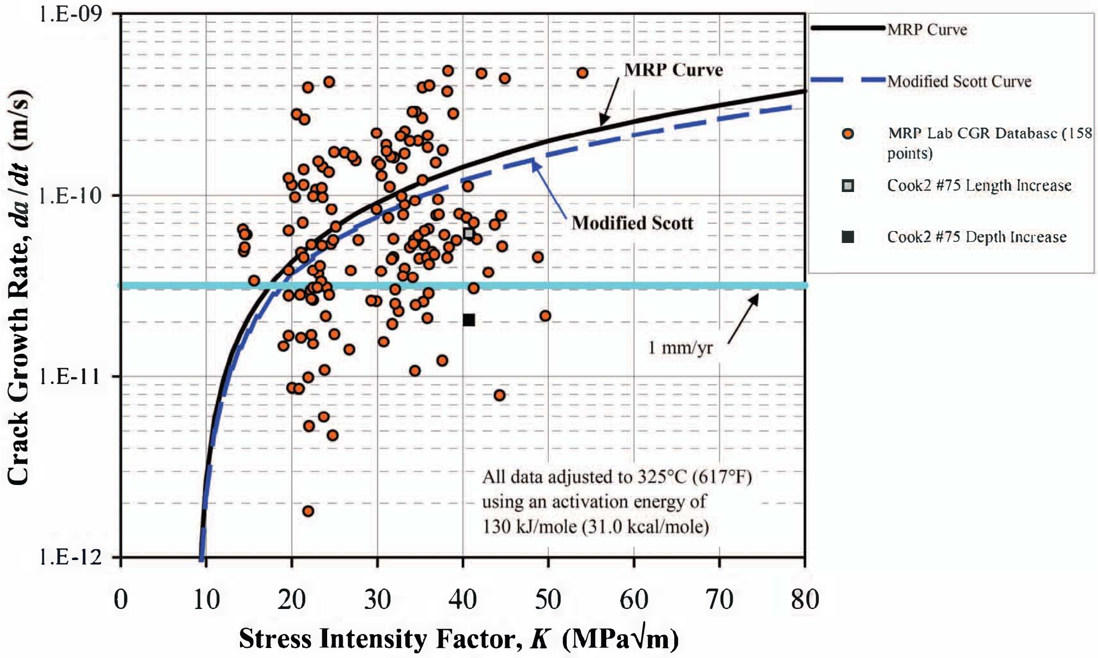

Figure 2-5: Screened Laboratory Data For Alloy 600 With The *MRP* Crack Growth Curve, The Modified Scott Curve, and *CGR* Data For Cook 2 Nozzle #75, White et al., 2003.

Reactor Internals and Recirculation Piping: The major components of the reactor internals, shown in Figure 2-3, include the core barrel, the upper and lower support plates and their associated assemblies. The functions of the components are both hydraulic, to contain and direct the coolant flow and structural, to support the core at the appropriate spacing of the fuel elements and control elements. All of the components are made of austenitic Type 300 *SS* with compositions shown on Table 2-4.

3 FUEL ASSEMBLY DESIGN

3.1 *GENERAL OUTLINE OF FUEL ASSEMBLY DESIGNS AND FUNCTIONS*

3.1.1 *PWR* and *BWR* (Alfred Strasser and Peter Rudling)

There is a wide variety of different types of fuel assemblies for Light Water Reactors, *LWR*s. The fuel rod array for *BWR*s was initially 7x7 but there has been a trend over the years to increase the number of Fuel Assembly, *FA*, rods and today most *FA* designs are either of 9x9 or 10X10 square configuration design. The driving force for this trend was to reduce the Linear Heat Generation Rate, *LHGR*, which resulted in a number of fuel performance benefits such as lower Fission Gas Release, *FGR*, and increased Pellet Cladding Interaction, *PCI*, margins. However, to increase utility competitiveness, the *LHGR*s of 9x9 and 10x10 *FA* has successively been increased, and peak *LHGR*s are today almost comparable to that of the 7x7 and 8x8 older designs.

Also for *PWR*s there has been a trend to greater subdivision of fuel rods, e.g. from Westinghouse 15x15 to 17x17 design. However to accomplish this one had to go to a new reactor design this since the *PWR*s do not have the same flexibility with core internals and control rods as do *BWR*s. Figure 3-1 shows the current *PWR* fuel rod array designs.

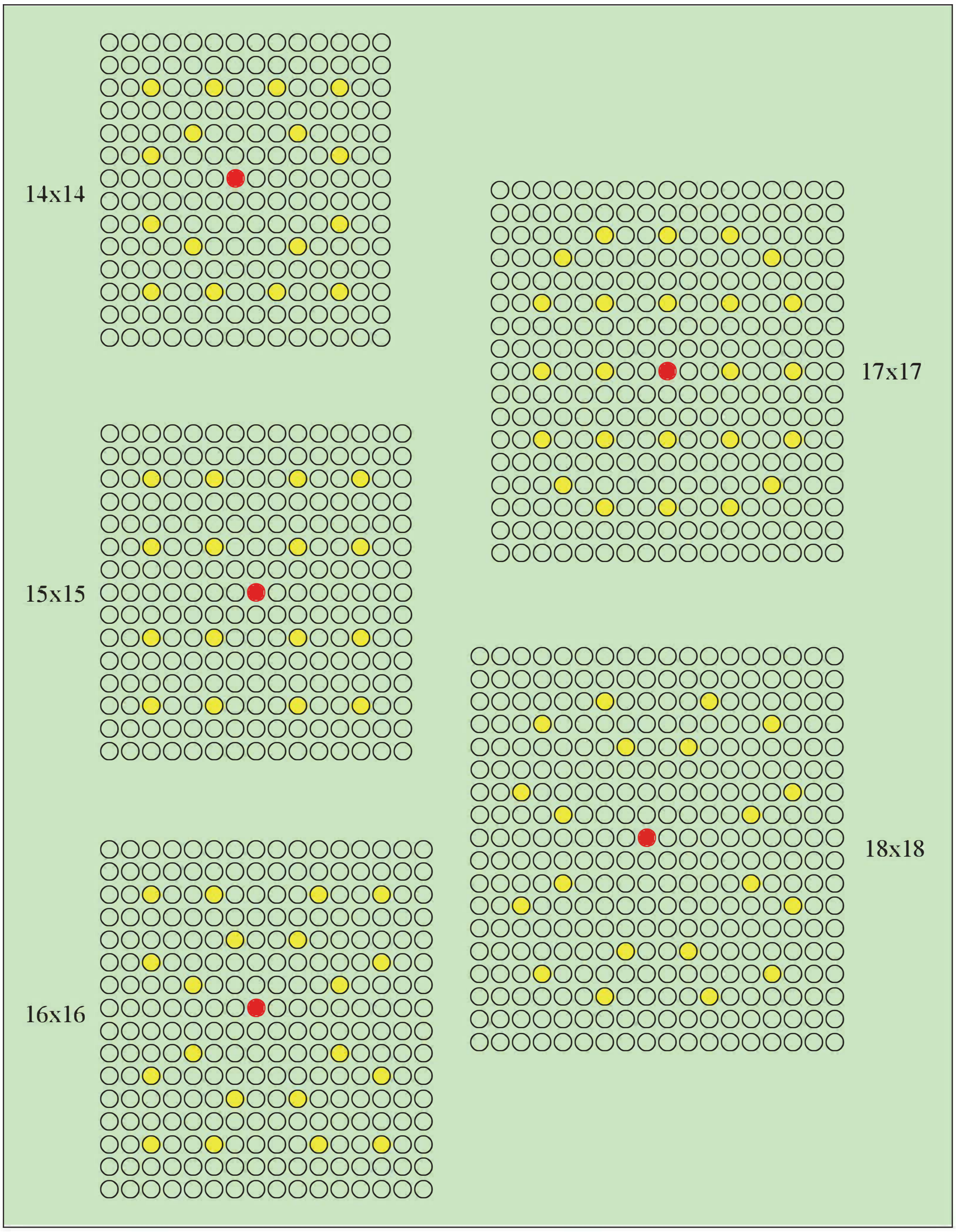

Figure 3-1: Layouts of different *PWR* fuel assembly design, Rods marked with yellow colour are guide tubes into which the control rod cluster is inserted. The position marked by a red filled circle is the instrument tube position.

In most *PWR*s, the assemblies are positioned in the core by bottom and top fittings, and the lateral clearances are restricted by the assembly-to-assembly contacts at the spacer-grid levels. Furthermore, the control rods consist of rod cluster control assemblies, *RCCA*s, the poison part of which moves into guide thimbles (or guide tubes). These guide thimbles are an integral part of the assembly structure.

In all *BWR*s the assemblies are enclosed in "fuel channels" surrounding the assemblies and between which the blades of the control rods moves.

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

- maintain proper positioning of the fuel rods under normal operating conditions and in design basis accidents (e.g. seismic effects, *LOCA*, *RIA*)
- permit handling capability before and after irradiation.

Figure 3-2 and Figure 3-3 show a typical *BWR* and *PWR FA*, respectively. Also, the different fuel assembly components are shown and the material selections for these components are provided. The reason for the difference in structural material selection is that in general the most inexpensive material is chosen for a specific component that yields the lowest cost to produce the component while ensuring adequate performance during normal operation and accidents.

4 FUEL ROD, ASSEMBLY AND PRESSURE TUBE PERFORMANCE

4.1 *FUEL ROD AND ASSEMBLY (AL STRASSER AND PETER RUDLING)*

The performance of the critical fuel components is the result of a complex interaction of a large number of variables that challenge the evaluation of the mechanisms in progress and the prediction of their behavior at extended and more severe conditions. The technologies involved include just about every aspect of materials science imaginable: properties of materials, metallurgy, structural mechanics, coolant chemistry, physical chemistry, and their basic mechanisms just to mention a few examples. In addition, exposure to radiation changes all of the physical properties and processes: the properties of the structural materials and of the coolant change, transformations in structure and composition occur in all the materials (true alchemy!), and these processes occur in a non-homogeneous and non-equilibrium manner throughout the core.

A study of the materials' performance is difficult even outside the reactor's radiation field and provides limited data. Test reactors offer a good tool for evaluating a limited number of variables and mechanisms and have provided some valuable data; however, the operation and use of these reactors is expensive. The final performance evaluation is in the power reactor itself since it provides all the variables of importance; however, the lack of instrumentation, the inability to control testing time, as well as the difficulty of separating variables makes interpretation of ongoing processes difficult. The final evaluation of new materials and fuels for high burnups progresses necessarily through the stages mentioned: ex-reactor testing, test reactor evaluation of samples, power reactor evaluation of samples or full fuel assemblies.

The degree of success achieved in fuel performance to date has been remarkable considering the lengthy evaluation process required and the tough conditions the fuel assembly is exposed to in service.

During normal operation the following material/component properties have a major impact on fuel performance and may limit the discharged burnup[21]:

- *Corrosion* of zirconium alloy cladding and the water chemistry parameters that enhance corrosion,
- *Dimensional changes* of zirconium alloy components,
- Stresses that challenge zirconium alloy *ductility* and the effect of hydrogen (H) pickup and redistribution as it affects ductility,

[21] The product of irradiation time (in days) and power generated by the fuel (in MW)

- *Fuel rod internal pressure,*
- Pellet-cladding interactions *(PCI)* and pellet-cladding mechanical interactions *(PCMI),*
- *Grid-to-rod fretting*
- *Fuel rod collapse*

The list has not changed significantly in over a decade, Goldstein et al., 1990. The only items above that have posed limits to extending burnups have been *corrosion* and *dimensional changes* in both *BWR*s and *PWR*s and *PCI* in *BWR*s, more details are provided in the *ZIRAT-8/IZNA-3* Special Topics Report on *High Burnup Fuel Issues,* Adamson et al., 2003(b) and Adamson et al., 2004. Improved materials and operating procedures have been able to exceed all of these limits and have not reached new limits within current operating strategies.

Poor fuel performance may result in fuel failures, Figure 4-1 to Figure 4-4 and Table 4-1, Table 4-2 and Table 4-3

Other material performance problems have occurred in reactors such as:

- *Manufacturing defects*- An example is the internal hydriding of fuel rods due to moisture in the pellets.
- Debris and baffle jetting fretting defects
- *Degradation of failed fuel*

The subsections to follow summarize the major parameters that influence the potential burnup limitations and the current and potential fixes that can extend the limits. The detailed discussion of their technology and current experience base are discussed in Vol. 2.

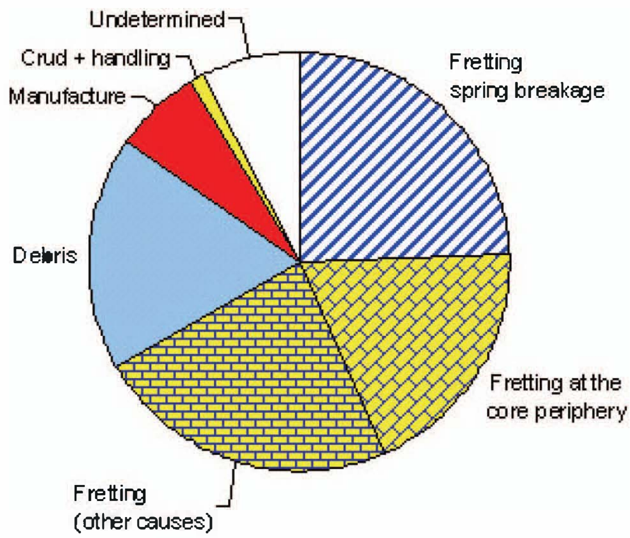

Figure 4-1: Siemens (now AREVA NP) *PWR* Fuel Rod Failure Causes 1992-2001, Klinger et al., 2002.

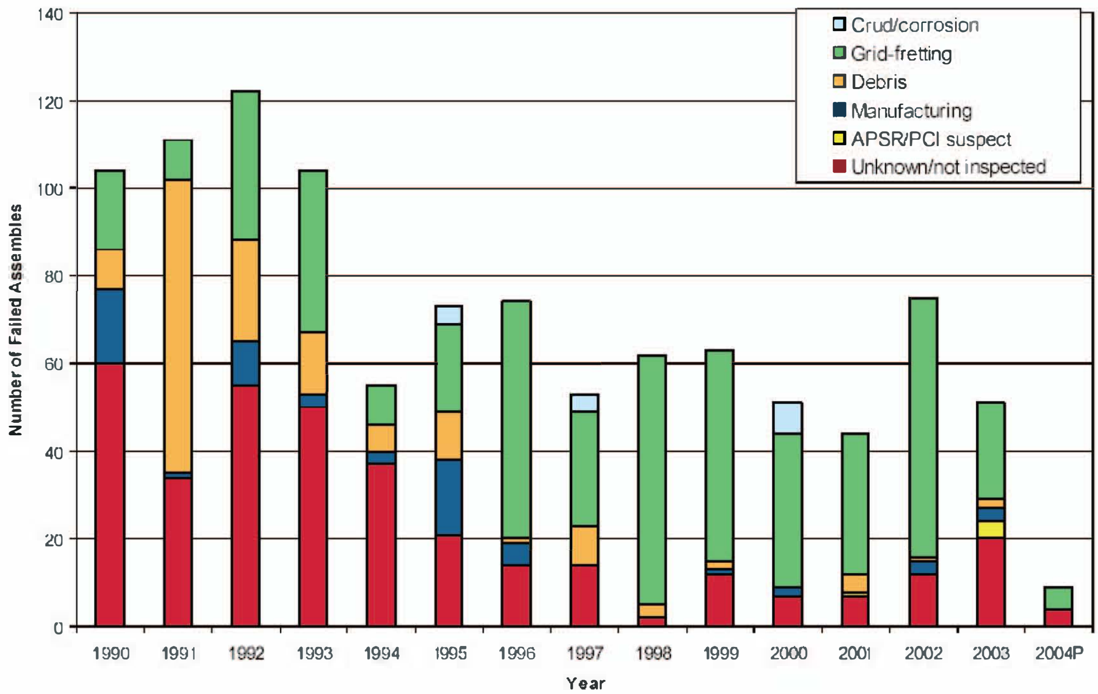

Figure 4-2: Trend in *US PWR* failure root causes (2004 results are incomplete), Yang et al., 2004.

5 FUEL PERFORMANCE CODES (PETER RUDLING)

5.1 INTRODUCTION

Analytical methods (computer codes) are used in safety analysis, either as standalone codes or in a coupled manner. These codes must include relevant models that accurately describes the performance of the fuel during normal operation and during design basis accident such as a *LOCA* and *RIA* at all burnup levels. The computer codes are of different types as described below.

Steady state fuel rod codes: The steady-state single-rod codes, like FRAPCON, COMETHE, TRANSURANUS, METEOR and TOUTATIS calculate thermal quantities such as the fuel rod radial temperature profile and fission gas release to the fuel rod gap, and mechanical quantities such as creep deformation and irradiation growth.

Results are used for many purposes like axial clearance between rods and end fittings, fuel rod internal gas pressure to compare with system pressure, cladding oxide thickness to compare with established limits or to initiate transient calculations (*LOCA* and *RIA*), stored fuel energy for *LOCA* analysis and fission gas repartition between grains, grain boundaries and porosities for *RIA* fuel failure mechanisms studies.

These codes consist of numerous models and correlations to describe gap conductance, material properties such as thermal conductivity and specific heat, radial power profiles, stress-strain equations, mechanical properties, creep properties, fuel-swelling, fuel-densification, waterside corrosion, and hydrogen absorption.

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

- The steady-state codes do not include transient heat-transfer terms in their solution equations; and
- The transient codes do not include long-term phenomena like creep.

However, the transient codes need to incorporate models, correlations, and properties for cladding plastic stress-strain behaviour at elevated temperatures, effects of annealing, behaviour of oxides and hydrides during temperature ramps, phase changes, and large cladding deformations such as ballooning.

The transient codes are used for analysing fuel rod response to transients and accidents like *RIA* and *LOCA* and may include failure models.

The RELAP5/Mod3.3 was developed as a best estimate transient code of light water reactor coolant systems during postulated accidents. The code models the coupled behaviour of the reactor coolant system and the core for operational transients such as anticipated transient without scram.

The consolidated code TRAC-M, is the *NRC* result to joint in a single code the capabilities and models of TRAC-P, TRAC-BF1 and RELAP, with the goal to reduce the maintenance costs of the previous structure based in three different codes and to achieve a better performance of the resulting code.

Reactor static codes – To generate the neutronic cross sections used in the multi-dimensional kinetic codes, instantaneous thermal-hydraulic parameters such as fuel and moderator temperatures reactor static codes are applied.

Reactor kinetics codes – Reactor kinetics codes, e.g. RAMONA, PARCS, SIMULATE-K, CORETRAN and SAPHYR calculates fuel assembly averaged neutron flux and power during reactor core power transients. A point-kinetics model may be used to analyse a whole core event such as coolant temperature changes in a *PWR*. However, localised events such as rod ejection in a *PWR* or control rod drop in a *BWR* require multidimensional neutron kinetic analysis.

Thermal-hydraulic, T-H, codes – Codes like TRAC, RELAP, CATHARE, ATHLET and RETRAN are T-H- codes that calculates flow, temperature and pressure during normal operation and transients. In most cases these codes contain point kinetics models to model the reactor power in *PWR* and 1-D kinetics models for *BWR*s. These codes may also be coupled with a 3-D neutronic kinetics code. The T-H codes have typically at least one coolant channel that models a few individual fuel rods. The fuel rod models in the T-H codes will normally include heat transfer correlations between the coolant and the fuel rod cladding, the pellet-clad gap for gap conductance, an average value for thermal conductivity and heat capacity. For *LOCA* analysis, these codes also contain normally ballooning, burst and oxidation models.

Subchannel codes: These codes are used to analyse the coolant flow distribution within the fuel assembly. These codes normally include a 3-D model for two-phase flow, 1-D models of the different fuel rods and detailed models for the heat transfer between the cladding and the coolant. These codes are used to calculate the critical heat flux to demonstrate that the Departure of Nuclear Boiling Ratio/Critical Power Ratio, *DNBR*/*CPR*, requirements are met.

5.2 *FUEL ROD STATIC AND TRANSIENT CODES*

To ensure that fuel does not fail during normal operation, anticipated operational occurrences and that coolability is maintained during postulated accidents (such as *LOCA* and *RIA*), fuel design criteria (such as maximum rod internal pressure, peak fuel temperature, boiling crises not allowed, peak cladding temperature during *LOCA*, etc.) are specified by the Regulators, see Vol. 2 for more details. In most countries the Regulators are applying the same criteria as USNRC.

The fuel vendors are using fuel performance codes to determine the thermal limits on their fuel design that will ensure that the fuel design criteria are met, Figure 5-1. For each fuel design criteria there will be a thermal limit varying with burnup and the most limiting thermal limit will establish the operating regime of the fuel design, see Figure 5-2.

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

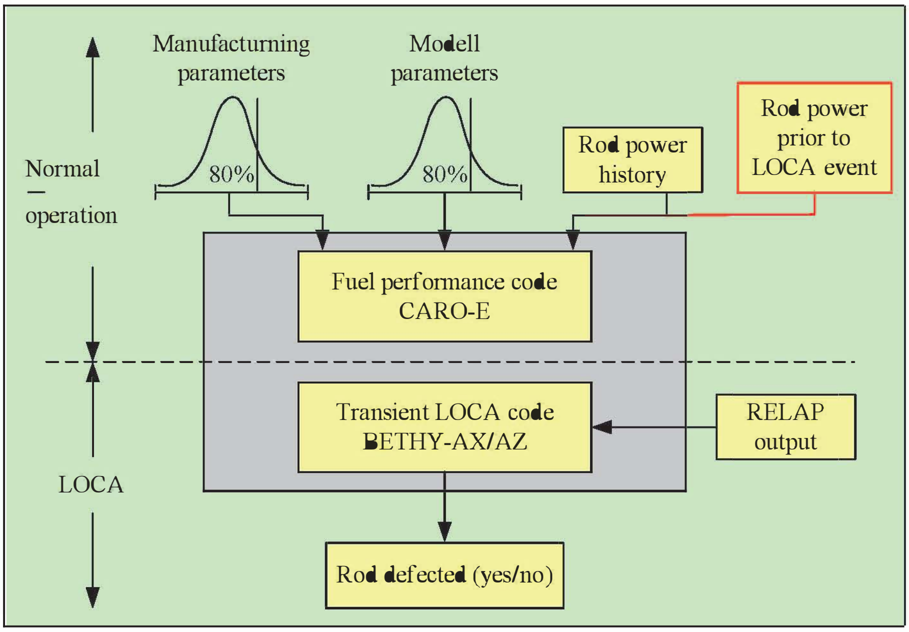

Figure 5-1: Schematics showing the process of establishment of thermal limits, modified figure according to Heins, 2004.

6 MANUFACTURING OF PRESSURES TUBES IN CANDU AND RBMK REACTORS (BRIAN COX)

Although initial studies focussed on ($\alpha+\beta$) quenching (~880°C) or β-quenching (~960°C) of the already extruded tubes, the use of "heat-treated" tubes was ultimately not pursued either in Russia or in Canada. However, several reactors were built with heat-treated tubes. These were the Japanese prototype reactor "Fugen"; the *CANDU*-BLW prototype in Gentilly-1 and the 180 MWe Karachi Nuclear Power Plant (KANUPP). Of these only the last is still operating. In addition a few heat-treated Zr-2.5Nb tubes were installed in other prototype reactors – 4 in *SGHWR*; (1 heat-treated + 1 cold-worked tube) in the 25 MWe *CANDU* prototype *NPD*. Three in-reactor loop tubes were also exposed in test reactors NRX and NRU. The tubes in Gentilly-1 saw very little irradiation exposure, but comparisons of the properties of the others have been published, Chow et al., 1996, Koike & Asada, 1988 and Koike et al., 1994. For present purposes only the comparison of the corrosion behaviour is appropriate, Figure 6-1. Although the corrosion of heat-treated tubes appears to be better than that of cold-worked tubes (base on this limited evidence), the hydrogen uptakes all appear to fall in the same scatter band, Figure 6-2. The low deuterium contents of the *NPD* tubes may reflect the comparatively low temperatures and fluences in this reactor.

The Russian and Canadian fabrication routes for cold-worked Zr-2.5Nb pressure tubes are very similar. Both comprise an initial β-quench of the billet prior to extrusion in the ($\alpha+\beta$) phase field (780-850°C), Figure 6-3, Cheadle et al., 1974, followed by a similar cold-working to final size (typically 25-28% recrystallised, Shishov et al., 1996. Only small differences in billet shapes and sizes and extrusion ratios distinguish the two products. The Canadian ingots are now quadruple melted compared to double melting for the Russians. The quadruple melting reduces impurities such as H, C and Cl in the initial ingot. This has resulted in a major improvement in fracture toughness, Figure 6-4, Coleman et al., 1996, and has reduced the average initial hydrogen content

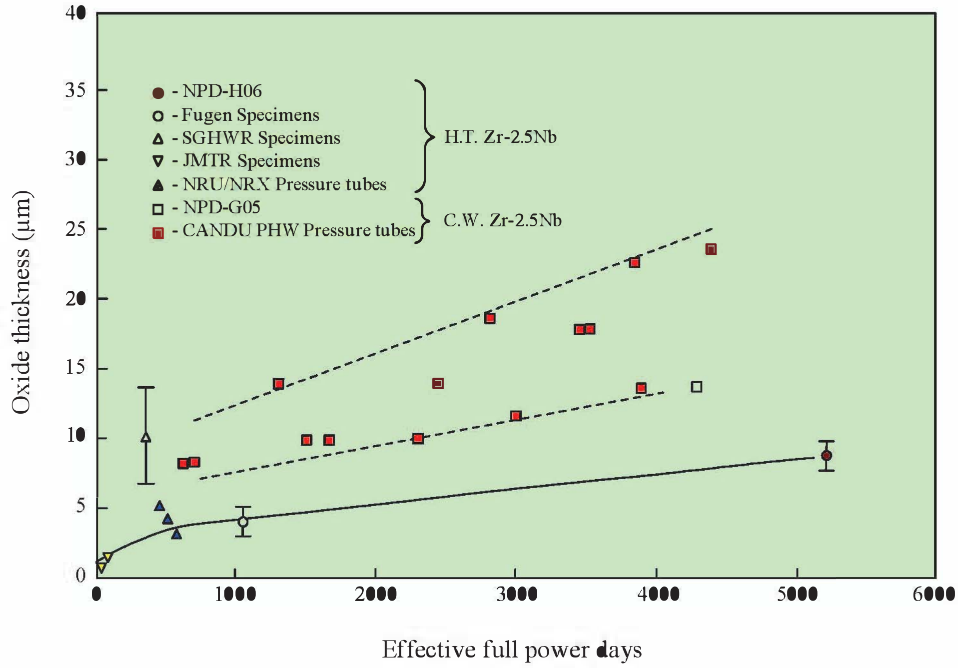

Figure 6-1: The maximum oxide thickness measured on the inside surface of Tube 589 is compared with other Zr-2.5Nb tubes and specimens as a function of time, Coleman et al., 1996.

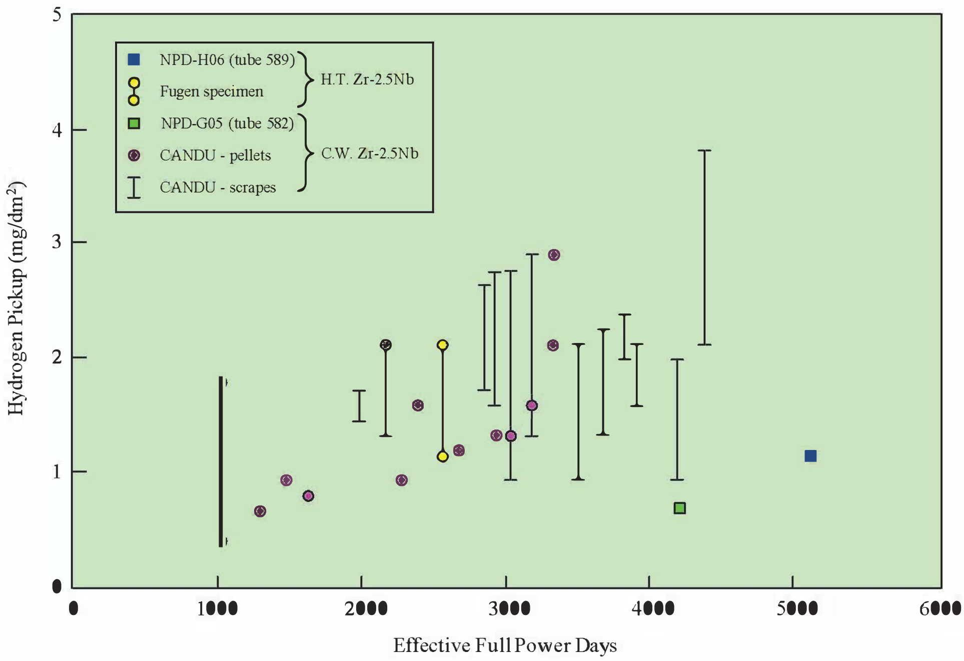

Figure 6-2: The maximum deuterium pickup measured in Tube 589 is compared, with terms of equivalent hydrogen, with other Zr-2.5Nb tubes and specimens as a function of time.

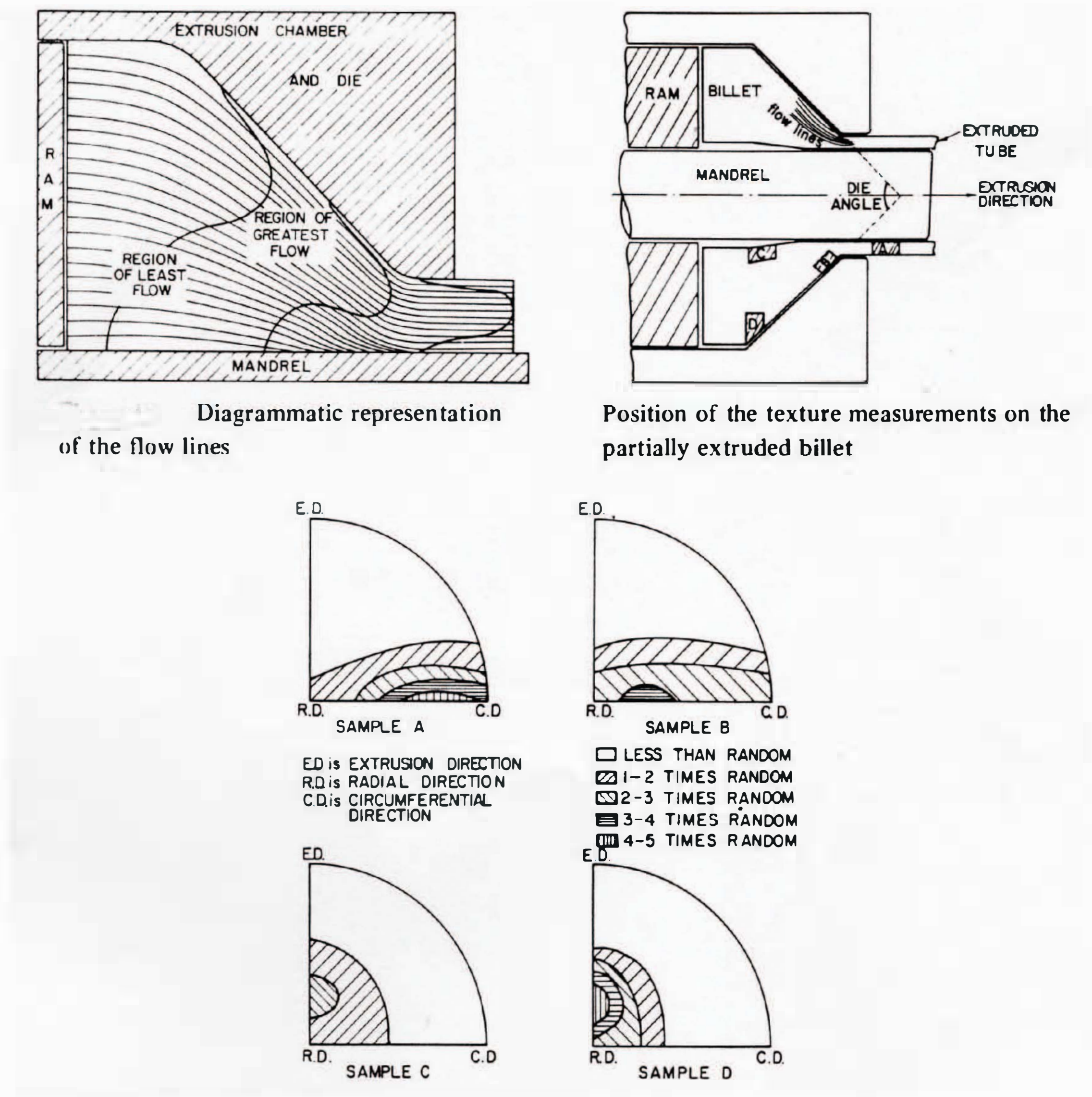

Figure 6-3: The (0002) pole figures of samples A, B, C and D from the partially extruded billet.

7 MANUFACTURING OF *BWR* AND *PWR* FUEL MATERIALS

7.1 CURRENT INDUSTRY SETUP (AL STRASSER AND PETER RUDLING)

The original number of fuel suppliers has been reduced by a series of corporate mergers and is currently dominated by a few large fuel manufacturers such as British Nuclear Fuel/Westinghouse), *GNF*, Framatome ANP and TVEL, Table 7-1. The table also indicates that there exists a large overcapacity to produce nuclear fuel today.

The ownerships and organisation structure has undergone large changes for *GNF*, Framatome ANP and Westinghouse, see Figure 7-1 to Figure 7-3. In addition, during the period when this document was being written, the Westinghouse organization has been purchased by Toshiba.

Table 7-1: Fuel suppliers and their fuel fabrication plants (worldwide list, capacities, etc.), World Nuclear Industry Handbook, 2003.

Facility	Capacity	Annual production	Operator	Product
		Europe		
Dessel, Belgium	40 tHM[34]/y 400 tHM/y	38 tHM NA*	Belgonucleaire FANP (old FBFC)	LWR/MOX PWR
Karlstein, Germany	400tU/y	NA	FANP GmbH	Components only, no pellet capacity
Lingen, Germany	550 tHM/y	450	FANP GmbH	BWR; PWR
Novosibirsk	1000 tU/y	NA	TVEL	VVER assemblies only
Ust-Kamenogorsk	2650 tU/y	NA	TVEL	VVER
Elektrostal, Russia	570 t/y 700 t/y	570 t/y 230 t/y	Mashinostroitelny Mashinostroitelny	RBMK fuel assemblies VVER fuel assemblies
Cadarache, France	15 tHM/y	NA	Cogema	LWR MOX
Melox, France	120 tHM/y	NA	Melox	LWR MOX
Pierrelate, France	400 tHM/y	NA	FANP	PWR, LWR MOX
Romans, France	750 tHM/y	NA	FANP	PWR
Juzbado, Spain	300 tHM/y	234 tHM	ENUSA	BWR; PWR
Västerås, Sweden	600 tUO_2/y	317 tUO_2	Westinghouse Sweden	BWR; PWR fuel components
NOFC Springfields, UK	330 tU/y	37 tU	BNFL	PWR
MDR Sellafield, UK	8 tU/y	NA	BNFL	MOX
		North and South America		
Ezeiza, Argentina	300tU/y	160 tU	NASA	PHWR
Resende, Brazil	100 tHM/y	16 tHM	Industrias Nucleares do Brasil	PWR
Columbia, USA	1200 tHM/y	NA	Westinghouse	PWR
Lynchburg, USA	400 tHM/y	232 t	FBFC	PWR
Richland, USA	700 tHM/y	NA	FANP GmbH (old ANF)	BWR; PWR
Wilmington, USA	1100 tHM/y	NA	GNF	BWR
Windsor, USA	300 tHM/y	NA	Westinghouse CE	PWR
Peterborough, Canada	1000 tHM/y	NA	GE Canada	PHWR
Port Hope, Canada	1200 tHM/y	NA	ZPI	PHWR
Toronto, Canada	1050 tHM/y	NA	GE Canada	PHWR
		Asia		
Hyderabad, India	300tHM/y 25tHM/y	NA NA	DAE DAE	PHWR BWR
Kumatori, Japan	284 tHM/y	130	NFI	BWR; PWR
Tokai, Japan Tokai, Japan	440 tHM/y	NA	Mitsubishi	PWR
	200 tHM/y	100	NFI	BWR; PWR
Yokosuka City, Japan	750 tU/y		Japan Nuclear Fuels	
Taejeon, Korea	400 tU/y 200 tU/y 200 tHM/y	NA NA NA	KNFC KNFC KNFC	PHWR PWR PWR

*Not available

[34] ton Heavy Metal

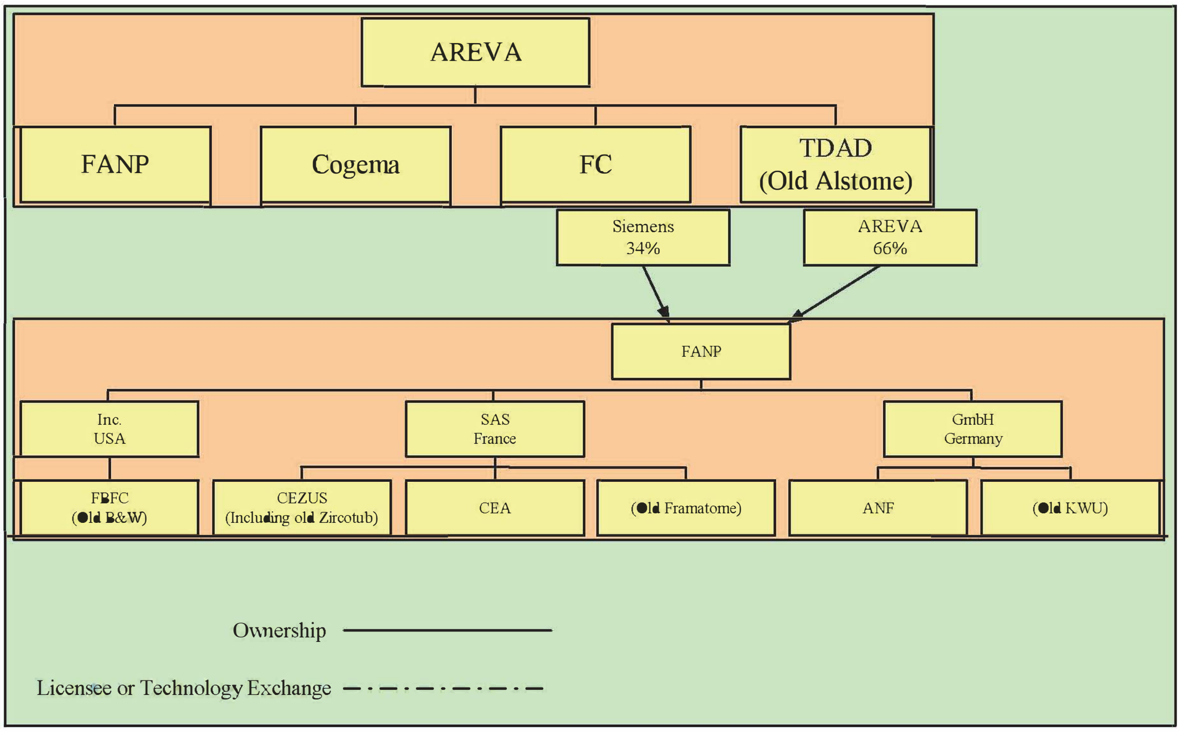

Figure 7-1: The organisation structure of Framatome ANP, FANP[35]. FANP is owned by Siemens 34 % and AREVA 66 %.

[35] Framatome NP has recently changed name to AREVA NP

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