FUEL DESIGN REVIEW HANDBOOK

Fuel Design Review Handbook

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1 INTRODUCTION (Alfred Strasser)

The reliable, safe and economical performance of nuclear fuel depends on its design, the processes used for its fabrication and the environment for its operation. Each of these three areas requires strict Quality Control (QC) and Quality Assurance (QA) to assure that those performance goals are achieved. The factors that control the quality are quite different in each of these areas, nevertheless they are interactive and the plant operator as well as the fuel vendor has to be aware of all of them. *This report provides a guide for reviewing and auditing the fuel design to assist in the assurance that it will perform its design functions adequately*. The trend to more demanding performance parameters and the competitive reload fuel market have resulted in a design evolution that could outpace experience. This has produced a renewed need for design reviews and associated audits to assure reliable fuel performance and maximize plant availability.

The guide to auditing fabrication processes is given in [Strasser & Rudling, 2004].

The overall objective of the design review is to assure that the Fuel assembly (FA) or fuel reload will perform reliably at or above the contracted conditions with adequate margins to design limits and licensing limits for its intended exposure in the reactor. Some detailed objectives include:

- To determine whether the design provides the best balance between meeting operating flexibility, reliability, licensing and economic goals.
- Evaluate whether the core can be operated for the intended cycle with adequate margins to provide sufficient flexibility and manoeuvrability.
- Determine compatibility with existing fuel in the core.
- Evaluate whether the design has eliminated past problems.
- Audit the vendor design QA system to assure that it is adequate and has been applied correctly, thereby also satisfying the requirements of 10 CFR[1] 50, Appendix B.
- Familiarize utility personnel with the vendor's design and its performance capabilities.

This is accomplished in several steps:

- Compare the design to the contractual and operating requirements.
- Review the nuclear, thermal-hydraulic, mechanical and materials designs and their interaction.
- Review the experience base and testing programs that provide the design bases.
- Review the drawings, materials and fabrication process specifications.
- Do independent analyses of specific areas for design verification.

A review of all aspects of the fuel design is not feasible or necessary within the time constraints of the utility and the vendor. This report intends to provide a guide to the items that have the greatest influence on fuel performance and prioritize the audits that are recommended. The objective is to do the most effective audit in the shortest time period.

[1] Code of Federal Regulations

The report provides the what? why? and how? for the audits by describing the design criteria, their influence on performance and the approach to reviewing the associated design features for the three distinct technical areas of nuclear, thermal-hydraulic, and mechanical/materials design, each written by experts in their field. A guide for design tool verification is included as well as a guide to auditing the vendor design QA system.

Additional sections include a description of the Loss-Of-Coolant Accident (LOCA), Reactivity Initiated Accident (RIA) and seismic design base accidents, their licensing criteria and their potential effects on the design. Applicable standards and guides for normal operation as well as for transients and accidents are described.

The report provides guidance for the important aspects of setting up and conducting an effective audit and for handling deviations from the design or design procedures when they occur.

2 THE FUEL SUPPLIERS AND THEIR DESIGN OFFICES (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 Westinghouse (W), GNF[2], AREVA NP[3], TVEL[4] and GE Canada in the western world. In Japan the fuel supplier as are dominated by Toshiba, Hitachi, MNF/MHI[5] and Nuclear Fuel Industries (NFI) in Korea by KNF and in India by Nuclear Fuel Complex (NFC), Table 2-1 and Table 2-2. The tables also indicate that there exists a large overcapacity to produce nuclear fuel today.

The ownerships and organisation structure of the fuel vendors have undergone large changes during the last couple of years and are described in the following subsections.

Table 2-1: Zr metal production

Product	Location	Company	Capacity per year (tons)	Process used
Zr Sponge from Zircon sands	Jarrie, France	AREVA NP[6]	2200	Chlorination – Kroll process
Zr sponge, Crystal bar from Zircon sands	Hyderabad, India	DAE	210	Not available
Zr sponge, Crystal bar from Zircon sand	Albany, Oregon, USA	ATI Wah Chang[7]	2000	Kroll process
Zr-metal from Zircon sand	Ogden, Utah, USA	Western Zirconium Div., W	3 000 000	Kroll process
Zr sponge, Crystal bar from Zircon sand	Glazov, Russia	TVEL	Not available	Electrolytic process[8]
Zr sponge, Crystal bar from Zircon sand	Ezeiza (ZMP), Argentina	National Aeronautics and Space Administration (NASA)	Not available	Not available

[2] Global Nuclear Fuel
[3] AREVA Nuclear Power
[4] Russian Fuel Vendor
[5] Mitsubishi Heavy Industries
[6] Previously CEZUS
[7] Previously Teledyne Wah Chang
[8] Changing to Kroll process

Table 2-2: Tubing, bars stock, flat product and fuel outer channel vendors.

Product	Location	Company	Capacity per year (tons)	Process used
		Tubing		
Tube Reduced Extrusion (TREX) from Zr sponge	Ugine, France	AREVA NP[9]	5000	Melting, forging, extrusion and swaging
Tubes from TREX	Montreuil Juigne, France	AREVA NP[10]	1200	Pilgering
Cladding and Guide Tubess (GTs) from TREX	Paimboeuf, France	AREVA NP [11]	5000 km/y	Pilgering
Tubes	Nagahama, Japan	AREVA NP [12]	300	Not available
Zr tubing from TREX	Duisburg, Germany	AREVA NP	2 200 km/y	Pilgering
Tubing from Zr sponge	Hyderabad, India	DAE	80	Pilgering
Zr tubing (from Zr metal)	Amagasaki, Japan	Zirco Products[13]	350	Not available
Zr tubing from Zr metal	Arnprior, Canada	GNF	1600 km	Not available
Zr tubing from Zr metal	Cobourg, Canada	ZPI	760	Not available
TREX from Zr-sponge	Albany, Oregon, USA	ATI Wah Chang[14]	Not available	Not available
Zr tubing from TREX	Wilmington, USA	GNF	Not available	Pilgering
Zr tubing from Zr alloy	Blairsville, Pennsylvania, USA	W	13 000 000 ft/y	Pilgering
Zr tubing from TREX	Kennewick, USA	SSM	2200	Pilgering
Zr tubing from Zr metal	Allens Park, USA	Nikko	500	Not available
Zr tubing from TREX	Chepetsky Plant, Glazov, Udmurtia	TVEL	2000 km/y (1992)[15]	Pilgering
Zr tubing Zr from TREX	Sandviken, Sweden	Sandvik Steel	1200	Pilgering
		Bar stock		
Ingots/barstock/TREX	Ugine, France	AREVA NP [16]	5000	Melting, forging, extrusion and swaging
Bar stock	Albany, Oregon, USA	ATI Wah Chang		Extrusion, rolling and swaging
Bar stock	Ogden, Utah, USA	Western Zirconium		Extrusion, rolling and swaging
		Flat products		
Strips/sheets from slabs	Rugle, France	AREVA NP [17]	450	Forging, hot and cold rolling
Strips/sheets from slabs	Albany, Oregon, USA	ATI Wah Chang[18]	Not available	Forging, hot and cold rolling
Strips/sheets from slabs	Ogden, Utah, USA	Western Zirconium	Not available	Forging, hot and cold rolling
		BWR[19] Fuel Outer Channels		
BWR Fuel Outer Channels	San Diego, California, USA	Carpenter	Not available	Sheet forming, machining, welding
BWR Fuel Outer Channels	Carpenter	AREVA NP[20]	Not available	Sheet forming, machining, welding
BWR Fuel Outer Channels	Wilmington, USA	GNF		Sheet forming, machining, welding
BWR Fuel Outer Channels	Kobe, Japan	Kobe Steel		Sheet forming, machining, welding

[9] Previously CEZUS
[10] Previously CEZUS
[11] Previously Zircotube
[12] Previously Zircotube
[13] Previously Sumitomo
[14] Previously Teledyne Wah Chang
[15] [IAEA No. 379, 1996]
[16] Previously CEZUS
[17] Previously CEZUS
[18] Previously Teledyne Wah Chang
[19] Boiling Water Reactor
[20] AREVA NP BWR Fuel Outer Channels are made by Carpenter

2.1 AREVA

AREVA NP headquarter is in Paris, France with main subsidiaries in the United States and Germany. AREVA has a 66 % and Siemens a 34 % of the shares in AREVA NP.

However, The CEO of Siemens informed the CEO of AREVA on January 27th, 2009 of his decision to exercise the option to sell the shares held by Siemens of AREVA NP's capital. In accordance with the shareholders' agreement of January 30, 2001, this notice, formulated before January 30, 2009, will take effect on January 30, 2012. This transaction ends all share capital ties. Once it has been completed, AREVA NP will be a wholly-owned subsidiary of AREVA.

AREVA established a relationship with Mitsubishi in 2007.

The current organisation of AREVA NP is however shown in Figure 2-1 and described in details in the following.

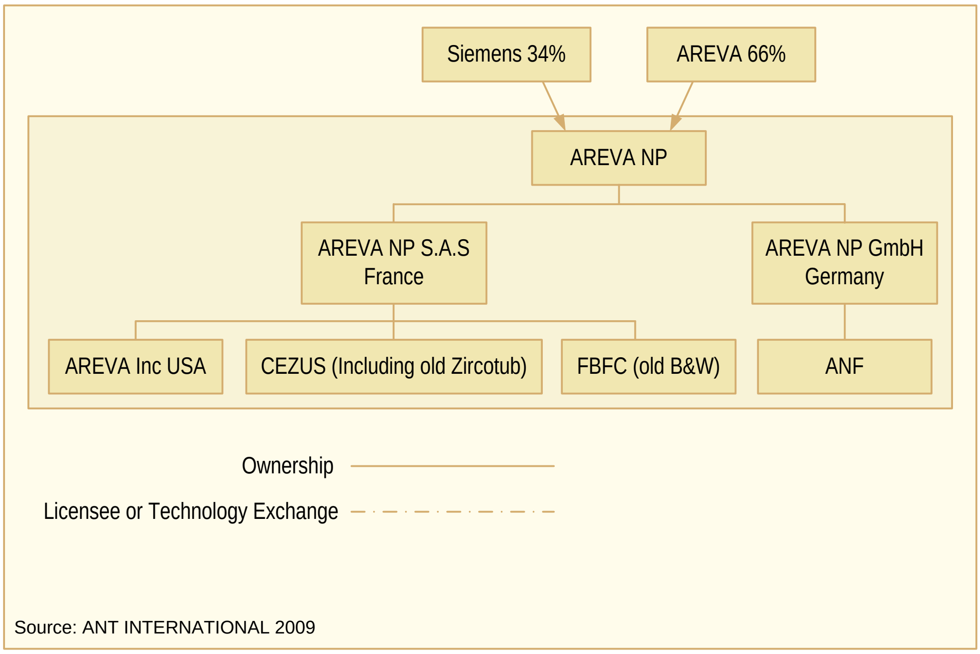

Figure 2-1: The organisation structure of AREVA NP. AREVA NP is owned by Siemens 34 % and AREVA 66 %. NFI in Japan used to be a Siemens licensee for BWR fuel and Babcock & Wilcox for PWR Fuel in 1970s. However, all of the current NFI fuel designs are developed and owned by NFI.

The Fuel Sector of AREVA NP has facilities in France, Germany and in the USA as follows:

- ANF, which manufactures zirconium products and fuel assembles in Germany. ANF, a wholly-owned subsidiary of AREVA NP GmbH, is based in Lingen in Germany, with facilities in Duisburg and Karlstein, as follows:
  - The Duisburg facility produces cladding and structural tubes for Pressurised Water Reactor (PWR) and BWR markets.
  - The Karlstein facility produces spacers, upper and lower tie plates for fuel assemblies and BWR water channels. The components manufactured are used for fuel fabrication at AREVA NP's fuel manufacturing plants.
  - The Lingen facility produces UO_2 powder, pellets, fuel rods and fuel assemblies for PWRs and BWRs.

- Companie Europen Ziconium Ugine Sandvik (CEZUS) Manufacture zirconium and hafnium products. CEZUS, a wholly-owned AREVA NP subsidiary, works on all aspects of zirconium metallurgy, from conversion of zircon, the ore of zirconium, to production of a variety of zirconium products such as zirconium alloy tubes, bars and sheet material for all types of reactor, including PWRs, BWRs and Canadian Deuterium Uranium (CANDU) reactors. CEZUS has five production sites in France, as follows:
  - Jarrie, produces zirconium sponge per year.
  - Ugine specializes in vacuum arc melting and hot forging of zirconium, using the sponge from Jarrie and recycled scraps. It then supplies these semi-finished products to other facilities to be processed into tubing, slab and strip.
  - Montreuil-Juigné, produces TREXs by pilgering once the extruded billets, received from Ugine. TREXs produced are delivered to fuel rod cladding and GT manufacturers and to Paimboeuf.
  - Paimboeuf produces cladding and GTs.
  - Rugles manufactures flat products used e.g. for spacer straps and channels (BWRs).
- FBFC, a wholly-owned AREVA NP subsidiary, is the world's first producer of PWR fuel assemblies (old Babcock &Wilcox), based on enriched uranium (UO_2) and of Enriched Reprocessed Uranium (ERU), as follows:
  - The Pierrelatte facility produces spacers for the PWR assemblies supplied to the different FA manufacturing plants of the group, notably the Romans and Dessel sites. It also produces spacers for MELOX's Mixed Oxide (MOX) fuel assemblies.
  - The Romans facility produces UO_2 powder and pellets, nozzles, fuel rods and fuel assemblies for PWRs.
  - The Dessel facility located in Belgium produces assemblies for PWRs based on UO_2 and MOX fuels. The facility also manufactures plugs and springs for fuel assemblies.
- AREVA NP Inc. is a wholly-owned subsidiary of AREVA NP SAS[21]. Its headquarter is in Lynchburg, Virginia.

2.2 GNF

GNF designs, develops and manufactures BWR nuclear fuel. GNF has manufacturing and fuel service facilities in US and Japan as well as in Spain through a join venture with ENUSA[22].

Toshiba and Hitachi each own 24.5% of GNF Japan Co. while General Electric (GE) holds 51%, Figure 2-2.

However, GE and Hitachi merged their nuclear business in July 2007 (GE-Hitachi Nuclear Energy), a year after Toshiba bought W. A new office has been set up by GE-Hitachi in San Jose, CA, which will market BWRs just after Toshiba established an office in US to gain market shares in the US with W AP 1000s (PWR) and Advanced Boiling Water Reactors (ABWR).

[21] AREVA Nuclear Power, French part
[22] Enusa Industrias Avanzadas S.A.

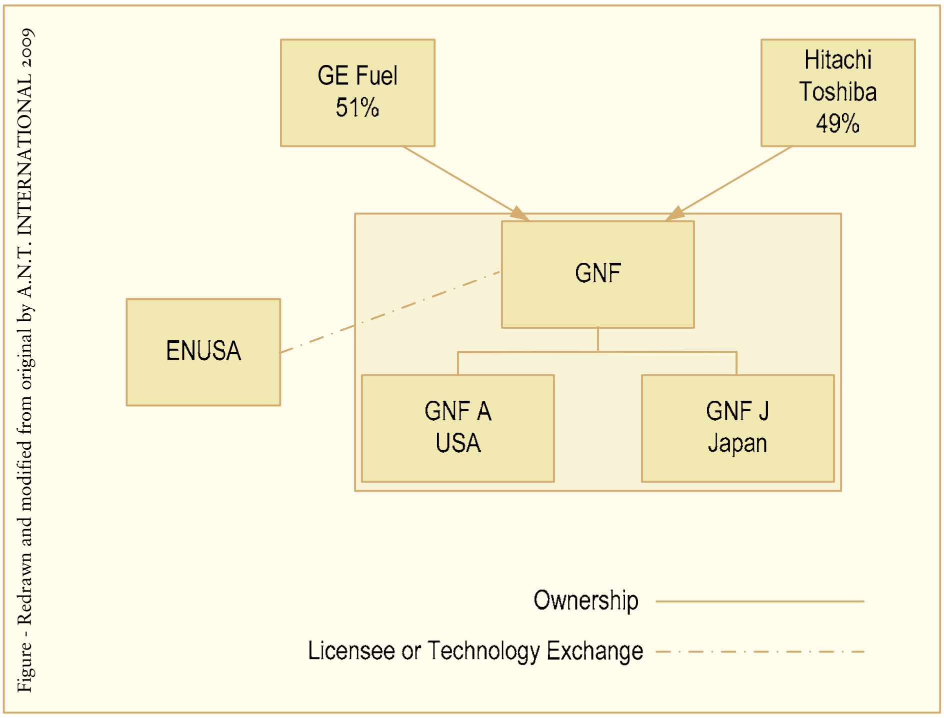

Figure 2-2: The organisation structure of GNF. GNF is owned by GE Fuel 51 % and Hitachi/Toshiba 49%. ENUSA in Spain is a licensee.

GNF has fuel design and fuel component and FA manufacturing facilities in Japan and USA (Wilmington). GNF is also supported by four global research centers, as follows:

- Global Research Center Headquarters, Niskauyana, New York, USA
- John F. Welch Technology Center, Bangalore, India
- Global-Research –Europe, Munich, Germany
- China Technology Center, Shanghai, China

2.3 W

The British Nuclear Fuels Ltd (BNFL) acquisition and integration of the ABB Atom and Combustion Engineering (CE) fuel organizations into W in 2000 brought together three businesses. In 2006, BNFL sold Westinghouse Electric Co. to Toshiba Corp., which owns 67% of the stocks in Westinghouse Electric Company. Westinghouse Electric Company provides fuel, services, technology, plant design, and equipment for the commercial nuclear electric power industry.

Westinghouse Nuclear Fuel, a wholly-owned subsidiary of Westinghouse Electric Company, has relationships with major nuclear fuel groups, such as MHI in Japan, Korea Nuclear Fuel Company (KNFC) in Korea, and ENUSA in Spain.

Westinghouse Nuclear Fuel manufactures PWR, BWR and Voda Voda Energo Reactor (Russian type PWR) (VVER) fuel in its facilities in the United States, Sweden, and the United Kingdom.

Westinghouse Nuclear Fuel has 7 manufacturing locations, as follows:

- Columbia, South Carolina, USA – The Columbia site is the Headquarter for manufacturing operations in the USA, including major PWR and VVER chemical and mechanical fabrication facilities, product engineering, and testing laboratories.
- Monroeville, Pennsylvania, USA - Westinghouse Core Engineering Services and Technology personnel are located at the Westinghouse Energy Center in Monroeville.
- The Specialty Metals Plant in Blairsville, Pennsylvania, USA produces and finishes seamless zirconium alloy tubing used in commercial nuclear fuel.
- The Western Zirconium Plant in Ogden, Utah, USA is processing zirconium and zirconium alloy materials.
- The Windsor Fuel Components Facility in Windsor, Connecticut, USA) produces components for use in combustion engineering (CE) and ATOM reload fuel. At this site also CE product marketing, contract administration, and product engineering is Located.
- Westinghouse Electric Sweden (WES) is located in Västerås, Sweden) is responsible for engineering, design, and manufacturing for European PWR and international BWR fuel customers.
- Springfields Fuels Limited in Preston, Lancashire, U.K.) is the headquarter of U.K. Fuel Business. Nuclear fuel chemical and mechanical fabrication for AGR and Magnox fuel customers. Conversion services for the production of natural uranium hexafluoride and uranium dioxide products are also done at this facility.

2.4 NFI

In 2009, Toshiba's Westinghouse Electric Co. purchased 52 % of NFI while Sumitomo Electric Industries Ltd. owns 24% and Furukawa Electric Co. also owns 24% of the shares.

NFI develops, designs and produces BWR and PWR nuclear fuel.

NFI has the overseas relationship related to Light Water Reactor (LWR) nuclear fuel with the Westinghouse Group and information exchange on PWR fuel with KNFC.

2.5 MHI/MNF

In April 2009, Mitsubishi Nuclear Fuel Co. was restructured. The shareholders Mitsubishi Material Corporation and MHI of Mitsubishi Nuclear Fuel Co. entered into a deal with AREVA and Mitsubishi Corporation to form a new company, which will integrate development, design manufacturing and sales. The new ownership structure is: MHI 35%, MMC 30%, AREVA 30% and MC 5%.

The new company, Mitsubishi Nuclear Fuel Co., will supply Japanese customers with UO_2 and MOX fuels for PWRs and BWRs.

AREVA and MHI are also planning to invest 50% each in a nuclear fuel fabrication facility in US.

2.6 TVEL

Rosatom Nuclear Energy State Corporation (Rosatom) is a state Corporation in Russia, the regulatory body of the Russian nuclear complex. It is comparable in function to the Nuclear Regulatory Commission(NRC) in US. Its headquarter is in Moscow. Rosatom controls nuclear power holding Atomenergoprom, nuclear weapons companies, research institutes and nuclear and radiation safety agencies.

TVEL with its headquarter in Moscow was founded in 1996 and is a wholly-owned subsidiary of Rosatom. TVEL has the following facilities:

- Mashinostroitelny Zavod ((MSZ) - machine engineering plant in Elektrostal)
- Novosibirsk Chemical Concentrates Plant in Novosibirsk
- Chepetsky Mechanical Plant in Glazov, Udmurt Republic
- Moscow Composite Metal Plant

TVEL designs and manufactures fuel assemblies for VVER, PWR and Reaktor Bolshoi Mozhnosti Kanalov (in English Large Boiling Water Channel type reactor) (RBMK) reactors. TVEL is an exclusive supplier of nuclear fuel to Russian, Bulgarian, Hungarian, Ukrainian and Slovakian reactors. It also supplies fuel assemblies to all European countries with VVER reactors. TVEL also exports nuclear fuel components, e.g. fuel pellets.

Siemens, which is negotiating an exit from the join venture with AREVA, see Section 2.1, have announced a proposal to join Rosatom Corp. to design, construct and operate nuclear plants. Rosatom will have 51% and Siemens 49% of the joint venture. The Siemens – Rosatom venture will focus on further development of Russias LWR VVER reactor technology. It will also deal with marketing and sales plus construction of new plants and upgrades of existing ones.

2.7 KNF

Korea Nuclear Fuel, KNF, is one of the subsidiaries of Korea Electric Power Co. (KEPCO) (others are Korea Power Engineering Co. Ltd (KOPEC); Korea Plant Service & Engineering Co., Ltd (KPS); and Korea Hydro & Nuclear Power Co. Ltd. (KHNP)).

KNF designs and manufactures CANDU and PWR nuclear fuel for the Korean market.

KNF has constructed a zirconium alloy tubing Techno Special Alloy (TSA) plant for PWR fuel tube supply and commercial operation started in November 2008 (TSA plant –). KNF is now looking into providing zirconium alloy tubes outside Korea. The current products are:

- Fuel cladding tubes
- Guide thimble tubes
- Instrument Tubes (ITs)
- Others

The production capacity of dry conversion and pelletizing process facilities was recently increased. KNF has also installed a Gd rod production line for PWR fuel as well as a Gd pellet production line.

- KNF has technical co-operation with:
  - Westinghouse USA
  - AREVA NP, France
  - GE-Hitachi Nuclear Energy, Canada (for CANDU fuel)
  - NFI, Japan
  - ENUSA, Spain
  - TVEL, Russia

2.8 NFC (India)

The NFC was established in 1971 as a major industrial unit of Department of Atomic Energy, for the supply of nuclear fuel bundles and reactor core components. The NFC in Hyderabad is unique in many respects. It is the only Complex of its kind where Uranium concentrates on the one hand and zirconium mineral on the other are processed at the same location all the way to produce finished fuel assemblies and also zirconium alloy tubular components, for supplies to the Nuclear Power Industry.

NFC produces fuel for India's Pressurised Heavy Water Reactors (PHWRs) as well as for their BWRs.

The complex has different types of production facilities which include:

- The Zirconium Oxide Plant for processing of Zircon to pure zirconium oxide.
- The Zirconium Sponge Plant for conversion of zirconium oxide to pure sponge metal.
- The Zircaloy Fabrication Plant for producing various zirconium alloy tubings and also sheet, rod and bar products.
- The Uranium Oxide Plant for processing crude uranium concentrate to pure uranium di-oxide powder.
- The Ceramic Fuel Fabrication Plant for producing sintered Uranium oxide pellets and assembling of the fuel bundles for the PHWRs.
- The Enriched Uranium Oxide Plant for processing of imported enriched uranium hexafluoride to enriched uranium oxide powder.
- The Enriched Uranium Fuel Fabrication Plant for producing enriched UO_2 pellets and the fuel assemblies for the BWR reactors.

2.9 CONUAR S.A. (Argentine)

Combustibles Nucleares Argentinos S.A. (CONUAR) in Buenos Aires was established in 1981. CONUAR is manufacturing and supplying the following products:

- Zirconium
- Fuel cladding
- Fuel engineering
- Fuel assemblies for conventional PWRs and BWRs.
- Fuel elements for Pressure Heavy Water Reactors (PHWR):
  - Atucha I (Siemens - KraftwerkUnion (KWU) Design)
  - Embalse (CANDU Type)

3 STRUCTURE AND COMPONENTS OF THE FA (Peter Rudling)

3.1 Introduction

There is a wide range of environments in which LWR fuel is operated. Table 3-1 shows typical ranges of various parameters.

Table 3-1: Design parameters in water cooled reactors.

	Parameter	Western type PWR	VVER (440/1000) MW	BWR
1.	Coolant	Pressurised H_2O	Pressurised H_2O	Boiling H_2O
2.	FA materials (pressure tube materials)	Zr-4, ZIRLO, DUPLEX, M5, MDA, NDA, Inconel, SS	E110, E635	Zry-2, Zry-4, Inconel, SS
3.	Average power rating, (kW/l)	80-125	83/108	40-57
4.	Fast neutron flux, average, $n/cm^2.s$	6-9E13	5E13/7E13	4-7E13
5.	Temperature, °C			
	Average coolant inlet	279-294	267/290	272-278
	Average coolant outlet	313-329	298/320	280-300
	Max cladding Outer Diameter (OD)	320-350	335/352	285-305
	Steam mass content, %			7-14
6.	System pressure, bar	155-158	125/165	*70*
7.	Coolant flow, m/s	3-6*	3.5/6	*2-5**
8.	Coolant chemistry			
	Oxygen, ppb	<0.05	<0.1	*200-400*
	Hydrogen (D_2),			
	ppm	2-4		*<1.8***
	cc/kg	25-50	30-60	
	Boron (as boric acid), ppm	0-2200	0-1400	-
	Li (as LiOH), ppm	0.5-6.0	0.05-0.6	-
	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
** Dependant on whether hydrogen is being added to the feedwater or not.

There is a wide variety of different types of fuel assemblies for LWRs. The fuel rod array for BWRs was initially 7x7 but there has been a trend over the years to increase the number of 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 PCI margins. However, to increase utility competitiveness, the LHGRs of 9x9 and 10x10 FA has successively been increased, and peak LHGRs are today almost comparable to that of the 7x7 and 8x8 older designs.

Also for PWRs there has been a trend to greater subdivision of fuel rods, e.g. from W 15x15 to 17x17 design. However, since PWRs do not have the same flexibility with core internals and control rods as BWRs, to accomplish this requires modified reactor internals. There is however, one exception namely DC Cook 1 which is switching to 17x17 by changing the reactor internals. Figure 3-1 shows the current PWR fuel rod array designs.

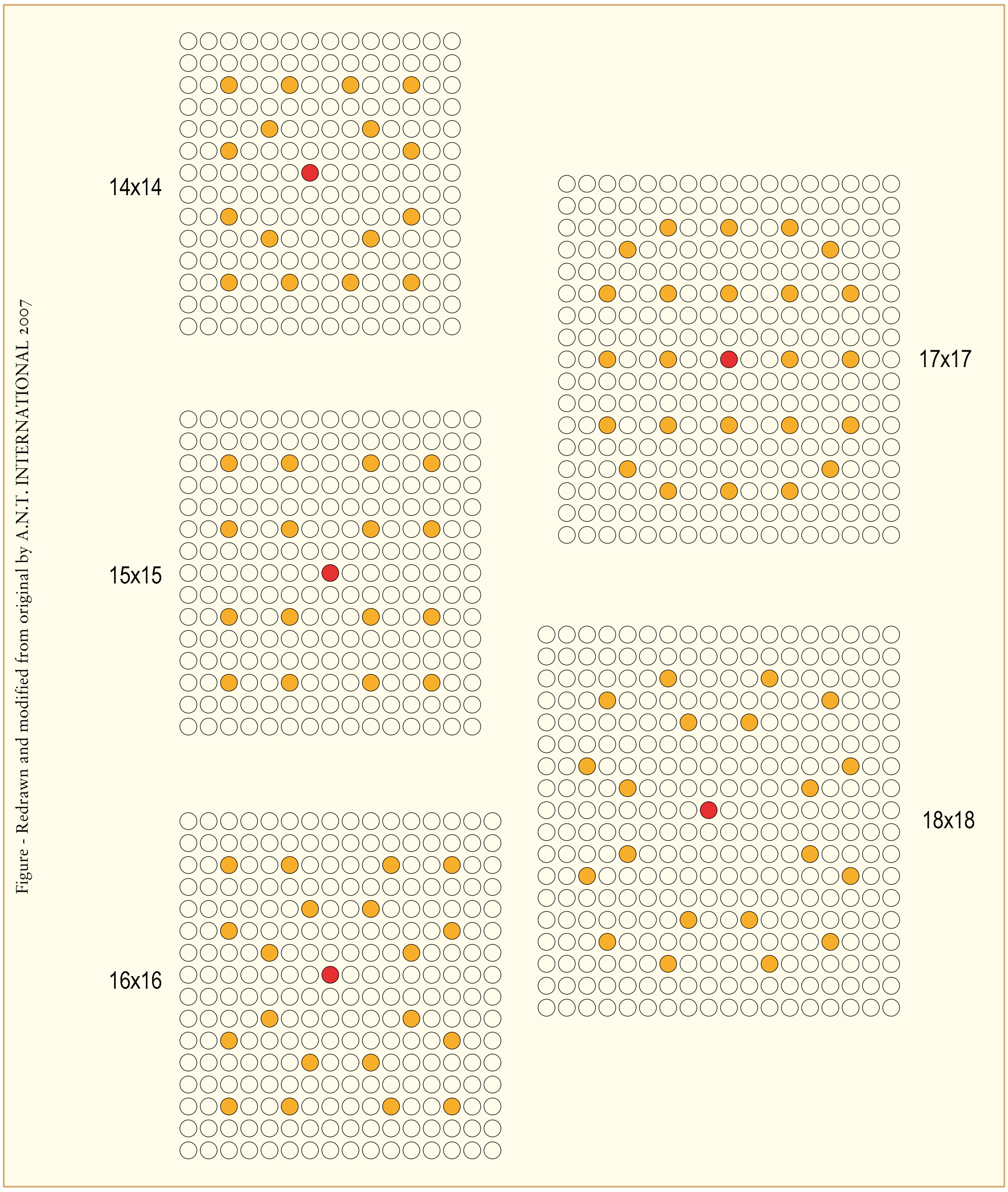

Figure 3-1: Layouts of different PWR FA design. Rods marked with yellow colour are GTs into which the control rod cluster is inserted. The position marked by a red filled circle is the IT position.

In most PWRs, 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, RCCAs, the poison part of which moves into guide thimbles (or GTs). These guide thimbles are an integral part of the assembly structure.

In all BWRs, 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 (DBAs) (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 FA components are shown and the material selections for these components are provided. The selection of the different structural materials are based on their nuclear and mechanical properties as well as their cost, in order to ensure acceptable performance during normal operation and accidents.

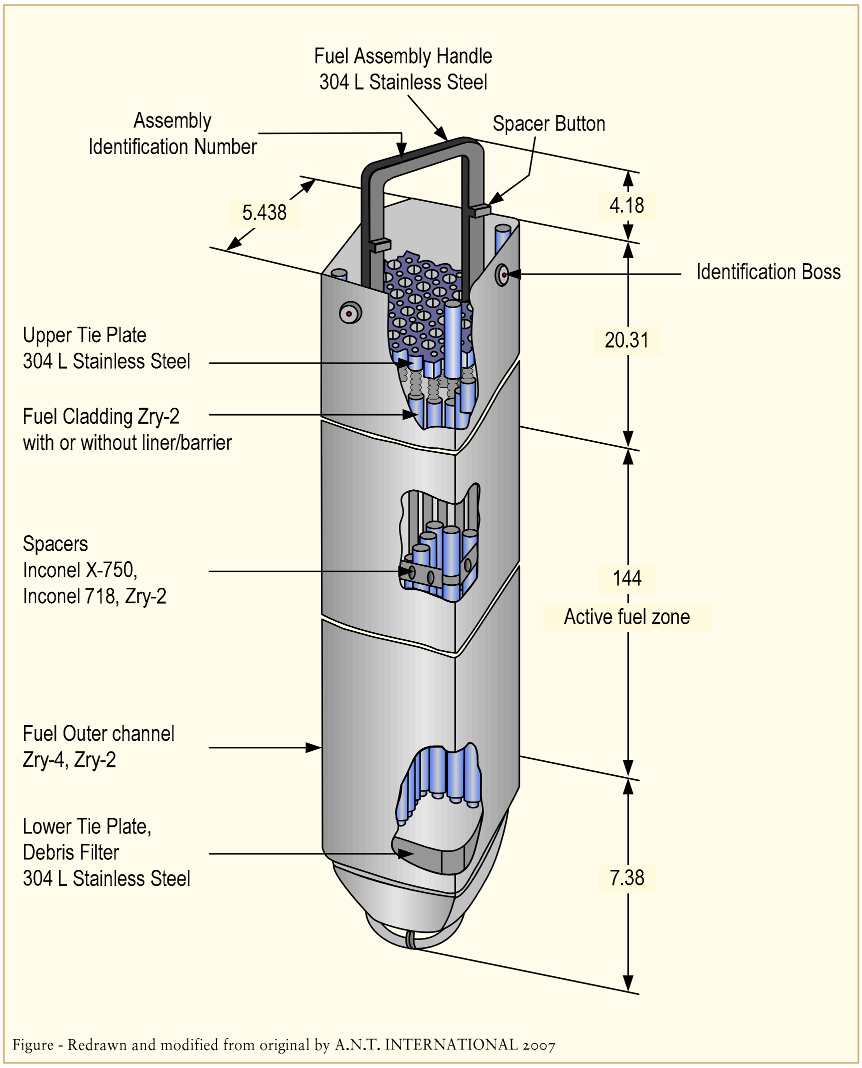

Figure 3-2: Typical BWR FA in inches.

4 AUDIT PROCEDURES (Alfred Strasser)

4.1 Overall Objectives

The broad objectives of nuclear fuel design review and its related audits are to:

- Verify the fuel design performance levels and projected fuel cycle costs to achieve the reliable and safe utilization of the fuel.
- Verify that the fuel design protects the health and safety of the public.

The achievement of the fuel performance goals, including fuel reliability and low fuel cycle costs, is dependent on good technical design characteristics and fabricated quality of the fuel, both of which are the responsibility of the fuel vendor. Achievement of the goals is equally dependent on good reactor operating practices - a responsibility of the utility. The protection of the public is a joint responsibility of the regulator, the vendor and the utility. The protection is based in part on satisfying the licensing limits established by the regulators. However, the requirements for reliable fuel performance during normal operation (failure free) are more stringent than those for safe fuel performance. This is because most fuel failures do not have a significant influence on the operation of the reactor and therefore do not affect the general safety of the plant. While the government provides good safety limits, the vendors and the utilities must implement more exacting QA procedures to meet their economic goals. Fuel design reviews and audits support these objectives by providing an independent review of the design and design related activities.

The cost savings accrued from improved fuel performance due to the increased confidence provided by the QA system (including audits) must be balanced against the cost of the QA system. Determination of the cost/benefit of increased QA levels is one of the most difficult aspects of establishing a detailed QA system since at some point the increased levels and increased cost of the QA system will bring diminishing economic returns in terms of performance.

In order to ensure that the audit itself is cost effective, the audit time must be used to concentrate on areas that are most closely related to potential performance problems. The main purpose of this Handbook is to identify and prioritize the QA of the design features that are most important for reliable fuel performance; however, one should be aware that details of some of these may vary from design to design and from one environment to another. Since the ability to identify the relationship between the design and performance of the fuel are critical, the scope and schedule of the audits should be established by, and the audits made by personnel familiar with, these areas of technology.

This Section will summarize the mechanics, or methods, of making the audits and will focus on areas that have the most sensitive relationship to fuel reliability. Good communications and relationship with the vendor are necessary to accomplish this goal and this topic is discussed as well. Adherence to these methods is essential in order to accomplish the most effective audit in the minimum amount of time.

Included in the section are:

- Description of various types of audits.
- The planning, scheduling and structure of an audit.
- The audit mechanics, or what to do during the audit.
- Handling of deviations.
- Post-audit follow-up.
- Audit of subcontractors.

4.2 Audit Types

4.2.1 Summary

The design review process can begin at the fuel procurement stage, especially since the ever increasing number of design features offered can leave the utility with the choice of less than fully proven designs. Most of the evaluation at this stage is concentrated in the nuclear design area in an attempt to verify reload enrichments and power distributions. However, mechanical and thermal-hydraulic issues deserve to receive attention as well to compare thermal margins and to verify solutions have been implemented to recent problems.

Once a vendor is selected or if the current vendor is offering a new or modified design, the mechanism for a detailed design review should be put in place. The major focus of the design review is on the three technical areas of the design:

- Nuclear design
- Thermal-hydraulic design
- Mechanical and materials design

In addition to the review of the respective designs, include audits of the:

- Topical reports
- Reload design reports
- Applicable licensing documentation
- Vendor quality system

Additional audits include the design bases accident analyses for:

- LOCA
- Reactivity insertion accident (RIA)
- Seismic event

The scope and emphasis of the audits will change according to the subject of the audit, which can vary as follows:

- New performance goals, such as cycle length or burnup, with standard assemblies.
- New fuel management method with standard or new assemblies.
- Change of one or more component in an otherwise standard assembly.
- A completely new assembly and reload from the same or a new vendor.
- One or more LTA.
- Uprate of reactor power with standard or new assemblies.
- Change in water chemistry with standard or new assemblies.
- Bid evaluation to compare several different designs.

4.2.2 Mechanical, Thermal-Hydraulic and Nuclear Audits

All three of these audits start with the review of the procurement documentation that defines the functional specification for the fuel design, the drawings, and the applicable design reports. For each topic the review will include but not be limited to the design basis, input data and criteria, the models and assumptions that were used, the margins and uncertainties involved, and the experience with similar fuel in similar reactors.

The *mechanical design review* should emphasize the relationship between the mechanical design bases and the application of the design to the plant. A change in design features or operating parameters should initiate the evaluation of the adequacy of margins for the four principle areas that can limit fuel performance, corrosion, hydriding, dimensional changes, PCI and fuel rod internal pressure. Mechanical compatibility within the core is another important parameter. The detailed guide is given in Section 8.

The *thermal-hydraulic design review* is generally most important for a reload from a new vendor or a modified design from the current vendor. Hydraulic compatibility, CHF margin, hydraulic stability and transient and accident analyses are some of the most important items to be reviewed for steady state operation. The detailed guide is given in Section 9.

The *nuclear design review* will evaluate the accuracy of the vendor's predictions of key parameters to assure that the nuclear performance objectives are met. Some of these parameters are power peaking and distribution, fuel reactivity as it affects cycle length and burnup, shutdown margin, reactivity margin (related to hot-cold swing in a BWR), peak discharge burnup, and compatibility with co-resident fuel. The detailed guide is given in Section 10.

The *licensing analyses* for steady state, transient, and accident analyses have to meet the regulators' criteria. The utilities' role is to review the input and compliance with regulator approved methodology, to assure that the analyses are performed correctly and to evaluate the extent of available margin for operating flexibility. A small margin may justify a reload and plant specific analysis. The details for steady state and transient analyses are discussed in each appropriate section above. The LOCA and RIA analyses are discussed in detail in Section 11 and the seismic analyses are discussed in Section 8.4.

4.2.3 Design Quality System Audit

A Quality System Audit is a detailed evaluation of an existing QA program for its conformance to company policies, standards, regulatory requirements and contract commitments. Nuclear quality system audits are based on criteria originally established in the US Code of Federal Regulations, 10 CFR 50, Appendix B and the many standards developed subsequently based essentially on this document. The Design Control and the Test Control requirements are some of the most important items in Appendix B that relate to *design QA*. The standards are discussed in Sections 6 and 7.

These important audits are used to determine whether the vendor is meeting its QA obligations according to the applicable standards and whether the vendor management controls are sufficient to ensure that the product will meet the contractual and performance requirements of the fuel. Independent design calculations by the utility should be subject to similar design QA audits. The recommended method for implementing these audits is discussed in Section 7.

5 DESIGN METHOD QUALIFICATION (Alfred Strasser)

5.1 Introduction

The design of fuel assemblies and their operation is based on databases, models and calculation methodologies embedded in most cases in computer codes. The ability of the codes to provide accurate and reliable predictions of the fuel performance is vital to the safe and economic performance of the nuclear plant. This requires that the codes be qualified by validation and verification. Code qualification is basic to the reliability of the design process in the same way that fabrication process qualification is basic to the reliability of the fuel fabrication process.

The credibility or reliability of a model or software product must be established through systematic testing of the model's accuracy and evaluation of the model's performance characteristics. The performance characteristics need to be determined for the full range of parameters that the code is designed to simulate. It is also important to test the code to determine the consequences if the code is used beyond its original design criteria, or beyond the range of applications for which it has been qualified. Confidence in the applicability of the code will increase with extended and systematic testing and, as a result, the codes will be modified with varying frequency. The latest approved version of the code should be used for the design.

Code qualification can consist of several types of activities. *Code validation* can be defined as the process of determining how well the code's theoretical foundation and computer implementation describe actual performance in terms of the degree of correlation between calculated and measured performance. *Code verification* can be defined as the process of demonstrating the consistency, completeness, and accuracy of the code with respect to their design specifications. *Code verification* is aimed at detecting programming errors, testing embedded algorithms and evaluating operational characteristics of the code by carefully selected examples of cases, test problems and test data sets. Code verification is limited to establishing the computer code with respect to design specifications. *Code validation* is more inclusive than code verification as it represents the final step in determining the validity of the quantitative relationships derived for the "real world system" that the model is designed to simulate.

The vendor's task for establishing a program for the verification and validation activities is quite broad and is part of the software life cycle. This process define the period during which a software system is active, beginning with the initial conceptual development and ending with its removal from active use. The life cycle activities of the overall software project are related to the verification and validation activities in Table 5-1 taken from the NRC's Software QA Program and Guidelines [Douglas, 1993]. This is a useful, overall guide to software QA along with a previous NRC publication prepared by the Pacific Northwest Laboratories, "Handbook of Software QA Techniques Applicable to the Nuclear Industry", [Bryant & Wilburn, 1987].

Table 5-1: Verification and validation activities by major life cycle activity.

Major life cycle activity	Verification and validation activities
Requirements definition	• Inspect requirements • Develop overall verification and validation plan • Conduct the software requirements review
Design	• Inspect design • Develop qualification test plan • Develop acceptance test plan • Conduct the preliminary design review • Conduct the critical design review
Implementation	• Develop unit test plans • Inspect unit designs, unit code, and unit test plans • Perform unit testing • Inspect unit test results • Develop integration test plans • Inspect integration test plans • Perform integration testing • Inspect integration test results • Develop qualification test procedures
Qualification testing	• Perform qualification testing • Write qualification test report • Develop acceptance test procedures
Installation and acceptance	• Perform acceptance testing • Write acceptance test report
Sustaining engineering and operations	• Perform, as appropriate, the verification and validation activities, defined above for requirements definition, design, implementation, qualification testing, installation and acceptance. • Performance regression testing as well as new tests for all levels of testing, as appropriate.

It should be noted that the various categories of the software life cycle and software QA are identified and organized somewhat differently in each guidebook and standard, although they cover essentially the same territory. The various categories can all be related to 10 CFR, Appendix B, even though they do not explicitly reference this regulation.

The utility auditor is unlikely to get involved in the broad, total QA of the software life cycle or even specific areas such as code testing or model testing, unless it is a new code that is introduced into the design process or one with identified problems. The majority of the audit activity will be in connection with code validation and verification.

The tools used for fuel designs can be divided into codes with and without the need for license approvals by the regulators. An example of a code that does not need NRC approval would be stress analyses of holddown springs in PWR nozzles. Examples of codes that require NRC approval are those that calculate DNBR in PWRs and CPR in BWRs, fuel rod internal pressures and pellet melt temperatures, as well as those that calculate transients and accidents including the DBAs of LOCA, RIA and seismic events. These are discussed in the individual sections on mechanical, thermal-hydraulic and nuclear design (Sections 8, 9, 10, 11).

Time constraints will not permit the review and audit of every design method. The need for selecting the methods for review can be based on the following criteria:

- Small margin to design criteria.
- Poor correlation of code predictions to experience.
- Limited data base and benchmarking or test data available.
- Changes to a code.
- New code or new application of a known code.
- Qualification of the code for predicting new performance parameters.

Details of the audit process are discussed in the next section.

5.2 Areas for Review and Audit

5.2.1 Documentation and Approvals

A review of the code manual and discussion with the vendor will familiarize the auditor with the objectives of the code selected for review, the models and methodology built into the code, the capability and limitations of the code.

The first step in an audit of that code would be to check the manual whether the latest version of the code was used in developing the design, whether that version has been approved by all the reviewers named in the Design QA manual and whether any unapproved changes have been made.

A second, informative step would be to review the vendor's corrective action program and QA audit reports of the software system in question.

5.2.2 Qualification Tests – All Codes

Verification and validation of the technical capabilities of the code can be made by reviewing the adequacy of code calculation in comparison with results from:

- Test cases with known solutions.
- Analyses without computer assistance.
- Other validated computer programs.
- Recent experimental and test data.
- Confirmed published data and correlations.

6 U.S. STANDARDS (Jerald Holm)

6.1 Introduction

The United States regulations most relevant to fuel designs are 10 CFR 50 Appendix A (General Design Criteria), 10 CFR 50 Appendix B (Quality), 10 CFR 50.46 (Loss of Coolant Accident), 10 CFR 50.59 (Changes), 10 CFR 50.67 (Radiological), and 10 CFR Part 100 (Radiological). The NRC guidance, which is most important to fuel design in the U.S.A., is the Standard Review Plan (SRP), particularly Sections 4 and 15. The important regulations are discussed in more detail in Section 6.2 and the NRC guidance is discussed in Section 6.3. Section 6.4 lists some guidance documents that the NRC has stated are acceptable for reference or which, while not endorsed by the NRC, provide useful guidance.

The regulations provide the requirements for the fuel design and the NRC guidance documents define acceptable approaches to satisfy the requirements. The fuel design must meet these requirements and do so in a manner consistent with the guidance in order for the fuel design to be approved by the NRC.

Each of the fuel vendors has an NRC approved topical report that defines the generic fuel design criteria that its designs will meet (examples are [ANF-89-98PA, 1995], [EMF-92-116PA, 1999], [BAW-10179PA, 2004], [NEDO-24011PA, 2005] and [WCAP-9272PA, 1985]). These generic design criteria address the mechanical, nuclear and thermal-hydraulic requirements for the fuel design. The generic design requirements satisfy the regulations and either comply with the NRC guidance or provide an alternative that the NRC accepts. The fuel vendor topical reports on generic fuel design criteria were established so that minor changes to the vendor fuel designs could be made without NRC review and approval. The fuel vendor can make changes to the fuel design as long as the NRC approved generic design criteria are satisfied and no other NRC approved topical report requires revision.

6.2 Regulations

10 CFR 50 Appendix A General Design Criteria for Nuclear Plants

The regulation 10 CFR 50 Appendix A specifies the general design criteria (GDC) for nuclear reactors. The general design criteria are often expressed in terms of FA performance and thus have a significant influence on the fuel designs.

The GDC that are of primary importance to the area of fuel design are GDC 10, 11, 12, 27, 35, and 62.

Criterion 10 – Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of Anticipated Operational Occurrences (AOOs).

GDC 10 establishes the requirement for SAFDLs that must be satisfied during normal operation and AOOs. This is the most significant GDC in that the majority of the fuel design activities, in particular design analyses, are directed toward demonstrating compliance with the SAFDLs. The SAFDLs are defined in Section 4.2 of NUREG-0800, which is discussed below in Section 6.3.

Criterion 11 – Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 11 establishes the requirement that, in the power operating range, the net effect of the prompt inherent nuclear feedback should be to compensate for reactivity increases. This has a direct effect on the nuclear design of the FA, which then influences the mechanical (or geometric) design and the thermal-hydraulic design.

Criterion 12 – Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

GDC 12 establishes the requirement that reactor oscillations that could challenge the SAFDLs should either not be possible or readily detected and suppressed. This GDC has a significant effect on the BWR design activities but little on PWR design.

Criterion 27 – Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System (ECCS), of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

GDC 27 establishes the requirement that there should be sufficient shutdown capability that the core is amenable to cooling. In practice this establishes shutdown margin requirements such that the core can be placed in a subcritical condition following an accident.

Criterion 35 – Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC 35 establishes requirements related to the capability of ECCS to minimize damage to the fuel and cladding during a loss of coolant accident. The regulation 10 CFR 50.46 augments these requirements and provides information related to demonstrating compliance with the GDC.

Criterion 62 – Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

GDC 62 establishes the requirement that criticality should be prevented in the fuel storage and handling facilities. This requirement impacts the nuclear design with respect to the initial reactivity of an assembly.

The GDC that are of secondary importance to the fuel design through their influence on the boundary conditions for analyses are 13, 20, 25, 26, and 28.

Criterion 13 – Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 20 – Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 25 – Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26 – Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 28 – Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Since the GDC are part of the regulations they must be complied with literally. Because the GDC are written in terms of objectives, the approach taken to comply by each fuel vendor may be significantly different with respect to details.

The details of each fuel vendor's approach to compliance with the GDC are important to the audit activities. These details can be understood by reviewing the generic fuel design criteria document for each fuel vendor (examples in [ANF-89-98PA, 1995], [EMF-92-116PA, 1999], [BAW-10179PA, 2004], [NEDO-24011PA, 2005] and [WCAP-9272PA, 1985]), which has been approved by the NRC. It is also important to review the supporting methodology documents related to each criterion, which have also been approved by the NRC. The methodology documents that are important to be familiar with are those that are related to the specifics of the fuel design review or audit plan.

The major NRC approved fuel vendor documents, which are important to the fuel design, can be found in the Technical Specifications Section 5.6.5 related to the establishment of the Core Operating Limits Report.

10 CFR 50 Appendix B QA Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

This regulation states the requirements for the QA program for nuclear power plants. Each fuel vendor has a QA program that complies with 10 CFR 50 Appendix B. There are eighteen items specified in Appendix B and, while all are important to the fuel design activity, one is more directly applicable then the others. Item III is entitled "Design Control" and reads as follows "Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests." These areas are normally a primary focus of the utility audit.

This is discussed further in Section 7.2.2 item 3 and Section 7.3 for audit methods.

7 DESIGN QA SYSTEM AUDIT (Alfred Strasser)

7.1 Introduction

The importance of an effective fuel design QA system cannot be overemphasized. Deviations from qualified design procedures will have a potentially more extensive negative effect on fuel performance than deviations from qualified fabrication processes. This is because a design deviation will affect all the fuel produced while a fabrication deviation is usually limited to an identifiable fuel component production run that can be scrapped or reworked. An actual example of a failure to adhere to the design QA system (Document Control) is the delivery of an entire reload batch of fuel to a plant with the wrong BA content. This was due to the fuel being fabricated to out-of-date drawings that were incorrectly delivered to the fabrication plant.

Each vendor and qualified subcontractor must have a QA Manual that describes the company's QA system. The manual may be in two parts. The first part could be very general, restricted to principles of operation and intended to satisfy regulators and reviewers interested in assuring that basic requirements are met. The second part is likely to be very detailed, describing how the QA activities are implemented, and providing guidance to company personnel associated with QA activities. The first part is generally available on a non-proprietary basis; however, the second part will probably describe proprietary operational procedures and have a restricted distribution. Each utility auditor should be familiar with both QA system descriptions since understanding the vendor's QA system is necessary for an effective audit. Obtaining both QA manual types from the vendor and maintaining them at the utility's offices is recommended to facilitate the auditors' familiarization with the systems prior to their audits as well as provide a convenient reference during the design process.

7.2 Basic System Requirements

7.2.1 History

The "parent" of all basic QA requirements in the nuclear field was established by the US Government's CFR Title 10, Section 1. These documents set forth the rules and regulations that govern the actions of the NRC and the applicable licensees. The CFR requirements are general and the detailed regulations and standards are evolved from these. The most relevant regulations to the fuel design processes described in this Handbook are contained in Appendix B to 10 CFR 50, "QA Criteria for Nuclear Power Plants and Fuel Processing Plants", issued in 1970. These criteria, in turn, were originally developed, prior to this time, by the US naval propulsion program under Admiral Rickover.

Appendix B has become the basic "constitution" of QA, as its 18 criteria cover all of the QA requirements and *the NRC requires that all licensed codes meet the applicable criteria of Appendix B*. Like other types of "constitutions", this one outlines the general principles and actions required of a QA system and are open to very broad interpretation. More detailed guidance was needed and this task was assumed by various standards organizations. The 18 criteria have been adopted by most countries, in one form or another, by their regulatory bodies, as well as their standards organizations. The 18 criteria are discussed in Section 7.2.2, and points that have a particularly important bearing on the design process are recommended for audit in Section 7.3.

In the US, the task of providing detailed guidance was assumed by the ANSI and the American Society of Mechanical Engineers (ASME) and for some issues by the NRC's RGs. The initial ANSI Standard N45.2 was the equivalent of Appendix B and its series of detailed standards were numbered N45.2 with additional numerical digits. The NRC staff reviewed each ANSI standard and subsequent to its issuance issued a parallel RG, which either endorsed or modified the ANSI standard. Subsequently a joint standard was issued with the ASME, NQA-1, in 1979, which had essentially the same structure as its predecessor. The latest version of this standard was issued in the year 2000.

The international community developed its series of standards with the International Organization for Standardization (ISO) starting in the 1980-ies and these are also based on the criteria outlined in 10 CFR 50, Appendix B. The ISO technical committees have memberships from the countries interested in the special topics covered by each standard. Both governmental and non-governmental organizations take part in developing the standards. Publication of a standard requires approval by a minimum of 75% of the members. The ISO 9000 series of standards is the set applicable to fuel fabrication. The recent issues also combined the standards with ANSI and the American Society for Quality (ASQ) and their latest publications are from the year 2000. These standards are discussed in Section 7.2.2.

7.2.2 Appendix B, the 18 Point Constitution

1) Organization

The responsibility for establishing and implementing a QA Program must be established and described. Of particular importance is the independence of the personnel that attain and verify quality. Auditors should confirm that the QA organization reports directly and independently to the manager of the design organization and has direct access to the president of the company.

2) QA program

Each vendor is required to have a documented QA Program Plan that provides (a) control over identified activities affecting quality and safety, including organization, (b) recognition of the need for special skills, (c) training of personnel that perform those activities, (d) management review of the status and adequacy of the QA Program. As noted in Section 7.1, the utility should have copies of the QA Program Plan in its offices in order to facilitate the auditors' familiarization with each vendor's program.

3) Design control

One of the most important items in the design process and the one most important to audit is the control of the process. Items that are included in this topic are:

- *Design input,* which includes design bases, performance and regulatory requirements, qualified design tools, their selection and approval.
- *Design process,* which describes and documents the verifiable process that will be used, design analyses that are controlled and documented using qualified computer codes, and changes that are justified, documented and approved.
- *Design verification,* will use design reviews, alternate calculations or perform qualification tests to show that the design tool produces the correct solutions for each design parameter; details are discussed in Section 5.
- *Change control,* relates to changes in design, operating conditions, non-conformances and must be documented, reviewed and either rejected or approved by authorized staff. Significant changes due to an incorrect design will trigger reviews of the applicable design process and verification.

- *Interface control,* will identify and control design interfaces between organizations; design information transmitted across interfaces will be documented and controlled.
- *Documentation and records,* will be stored and maintained according to written procedures. The documents will include those that show that the design has been verified according to standards and according to the sources of input that support the design. All changes and revisions are included.

4) Procurement document and control

Technical and applicable quality requirements, including a documented QA Program that meets applicable standards, must be included or referenced in the Procurement Documents issued to subcontractors. Other items include the utility's rights of access to the vendor's data and documents, the document requirements and the non-conformance reporting requirements.

5) Instructions, procedures and drawings

Documents must be prepared for all activities that affect quality and preparation of these documents must include appropriate quantitative and qualitative criteria for attaining the desired quality and for satisfactory accomplishment of the activities described.

6) Document control

A control system must assure that all quality related documents are issued only after review for adequacy and approval by authorized personnel. Changes to the documents require the same degree of control as the original documents and must be distributed to the locations where the related work is being performed. Auditors should review the documentation of critical steps in the design to assure that the drawings and specifications and QC plan are all the latest, compatible versions with the work being performed.

7) Control of purchased material, equipment and services

This point applies primarily to the fabrication of the fuel, but could have application to the design process for purchases of software, computers or services. Measures must be established to assure that purchased materials, equipment and/or services conform to the requirements of the purchase documents, including all quality requirements including the QA Program plan. It requires the vendor (the purchaser) to evaluate the subcontractors and suppliers and acquire documentary evidence that the procurement requirements are being met. Utility auditors should assure that the vendors have documented evidence of such evaluations and audits and in selected cases the utility auditors should evaluate the subcontractors independently.

8) Identification and control of material parts and components

This applies to the fabrication process.

9) Control of special processes

This applies to the fabrication process.

10) Inspection

Inspection of activities affecting quality must be performed by individuals independent of those that performed the activities, to verify conformance to documented requirements. This also applies primarily to the fabrication process, but can be interpreted to relate to the QC of the design process.

8 MECHANICAL DESIGN REVIEW (Alfred Strasser and Peter Rudling)

8.1 Introduction

The review of the mechanical design of a FA includes 3 parts: (1) review of the overall design specifications, including planned operation and functional requirements and mechanical design objectives, including reliability goals and licensing requirements; (2) review of drawings and specifications, if applicable; and (3) review of overall mechanical design bases and limits for the assembly and for each component. Each of these parts is discussed below.

8.1.1 Review of Overall Design Specifications

The reliable mechanical performance of the FA is strongly dependent on the planned operation of the assembly during its operating lifetime. This planned operation includes:

- Cycle length and overall residence time,
- Average and peak burnups to be achieved,
- Fuel management scheme,
- Start-up and shutdown procedures,
- Applicable plant technical specifications,

and defines:

- Core power history and the power histories for the assembly and for individual fuel rods,
- Hydraulic environment, including coolant temperature and pressure, and coolant chemistry.

Functional requirements, such as these, should be defined in the contractual documents and provide some of the bases for the nuclear and thermal-hydraulic design as well. Applicable limits should be included in the vendor’s design documentation.

The general mechanical design objectives should be defined by the contract and the vendor’s documentation and should include:

- A clear statement of the design mechanical performance requirements.
- Reliability and mechanical warranty, if applicable.
- Applicable design procedures, including application of design margins.
- Applicable vendor QA program(s) and its conformance to regulatory and national standards.
- The requirement to meet current regulatory criteria relative to mechanical performance under normal and accident conditions.
- Clear statement of any assumptions used.

The specific mechanical design objectives are discussed in detail for each component in their respective sections.

The review of overall design specifications should address whether the functional requirements and design objectives are complete and consistent with contractual and licensing requirements.

8.1.2 Drawing Review

The drawings, and specifications where applicable, for the FA, the subassemblies such as the fuel rod and spacers, and single components such as the cladding tube and fuel pellets should be given a general overview. Detailed review of the drawings and specifications should be made at the time of the audit of the individual components or assemblies.

The overview should answer questions such as:

- Is the drawing set complete?
- Does the drawing set and individual drawings meet contractual conditions? Do they represent the FA model that was purchased?
- Do they have the appropriate QA and procedure approvals?
- Are they the latest revisions?
- If changed from a previously supplied fuel batch, what is the reason for the changes?

8.1.3 Overview of Mechanical Design Bases and Limits for Components

The mechanical performance and reliability of the FA and its components is affected by the applied stresses and strains, dimensional changes, hydriding and the effects of corrosion resulting from the operation of the assembly over its design lifetime in conjunction with the effects of irradiation on the material properties and on the coolant condition. Sources for applied stresses and strains, dimensional changes, hydriding and corrosion are summarized as follows.

- Stresses, strains, and cyclic loading due to:
  - Mechanical interactions between the assembly and support structure and control rods and between assembly components and the weight of components.
  - Fuel rod cladding differential pressure, including FGR, and PCI and PCMI.
  - Differential growth between fuel rods and assembly structural components and between the assembly and the support structure and control rods.
  - Hydraulic forces, including pressure differentials.
  - Hold-down forces.
  - Vibration forces.
- Dimensional changes and their effect on stresses due to:
  - Creep
  - Thermal expansion, including in-plane and axial components.
  - Irradiation growth.
  - Fuel rod elongation due to PCMI.
  - Hydrogen pick-up induced growth.
  - Oxidation induced growth.

- Power history, particularly the overall power history for individual rods and its effect on cladding stresses due to:
  - Temperature
  - Number, magnitude and rate of power increases.
  - Power cycling.
  - Low power operation.
- Environmental exposure and corrosion effects on mechanical performance, particularly of fuel rod cladding, due to:
  - Temperature
  - Coolant condition (pressurized, boiling, subcooled boiling).
  - Coolant hydrolysis due to irradiation.
  - Water chemistry additives (boron, lithium, zinc, hydrogen, noble metals).
  - Coolant pH, as impacted by water chemistry additions and impurities and intrusions.
  - Debris in coolant.

These parameters need to be considered in the mechanical design of the assembly and its components and are listed here to provide an overview. Additional factors may need to be considered for specific components. The stress and stress distribution for each component is the result of a unique and complex interaction of the various factors listed above and is different for each component. Hence, the parameters that need to be considered are discussed in detail for each component in the Sections that follow.

In general, one can say that the most complex interaction of thermal, mechanical, environmental and operating parameters is represented by the thermal-mechanical design of the fuel rod. The zirconium alloy cladding tube is exposed to both internal and external forces while irradiation and corrosion and hydriding are changing its properties during operation. The internal forces result from fuel swelling and thermal expansion and gas generation and release, and the external forces result from the coolant pressure and restraint by the spacers and structural members such as GTs and water rods. The internal and external pressures result in a constantly changing pressure differential on the cladding. Oxidation of the cladding occurs internally due to the fuel and externally due to corrosion by the (water) coolant. Internally generated fission products can result in cladding embrittlement that in turn can result in Irradiation Assisted Stress Corrosion Cracking (IASCC) of the cladding. External oxidation results in hydrogen pick-up by the cladding that in turn changes its properties. Irradiation and the accumulated fluence with time affect the cladding properties as well. Corrosion by the coolant reduces the wall thickness of the cladding and thus reduces the strength of the cladding. Also, the deposition of corrosion products, including oxide, on the cladding impedes heat transfer and results in an increase in cladding temperature that may accelerate corrosion and wall thinning and decrease the margin to rupture. Details are discussed in Section 8.2.1.2.1 and shown in Figure 8-2.

The design principles of the fuel rod are generic to BWRs and PWRs and will vary from design to design in dimensions, materials used, power history and environmental conditions.

9 THERMAL-HYDRAULIC DESIGN REVIEW PWRs AND BWRs (Kenny Epperson and Jerald Holm)

9.1 Introduction

9.1.1 General Thermal-Hydraulic Design Considerations

The reliable thermal-hydraulic performance of the FA is strongly dependent on the fuel cycle planned for the reload batch and the specific assembly design in the reactor. This includes the:

- Cycle energy or length and burnup.
- Core loading pattern.
- Core power generation limits, peaking factors, and maximum FA power levels.
- Hydraulic environment, including coolant flow rate, system temperature and pressure.
- Applicable plant Technical Specifications to the fuel and core design.

Functional requirements, such as these, should be defined in the contractual documents and provide bases for the nuclear and mechanical design as well. Applicable limits for these parameters should be included in the vendor's design documentation.

The general thermal-hydraulic design objectives defined by the contract and the vendor's documentation should include:

- A clear statement of the thermal-hydraulic performance requirements of the design.
- Fuel reliability and thermal-hydraulic warranty as applicable.
- Applicable design policies including establishment and tracking of design and operating margins.
- Application of the vendor's QA Program and its conformance to a regulatory and national standard.
- All assumptions should be clearly stated.

The reliable thermal-hydraulic performance of the FA and its components is affected directly by the heat transfer characteristics from the cladding surface. Any level of inadequate heat removal, from small degradations due to thin deposits on the cladding surface all the way to catastrophic loss of cooling due to vapour blanketing, will directly affect the thermal performance of the fuel. The following items impact the cladding heat transfer characteristics:

- Design and as-built condition of the components:
  - Fuel cladding surface.
  - Grid straps including strip intersection geometry and weld conditions.
  - If applicable, mixing feature geometry including thickness, angles, and blockage ratio.
  - Top and bottom FA nozzles or tie plates.
  - FA spring hold down forces.

- Plant operation effects including:
  - Core power level.
  - FA inlet and outlet temperatures.
  - System pressure.
  - Core/assembly flow rates.
  - Flow between assemblies due to local hydraulic mismatches (PWRs).
- Environmental exposure effects including:
  - Water chemistry additives: boron, lithium, zinc.
  - Water chemistry impurities and intrusions, including local deposition of corrosion products (CRUD).
  - Debris in coolant blocking flow channels.

These parameters need to be considered in the thermal-hydraulic design of the assembly and its components and are listed here to provide an overview.

In general, the most complex interaction is the grid geometry, especially if mixing features are present on the grid design. The grid in a FA not only provides structural support for the fuel rods, the grids are the primary means of increasing heat transfer on the fuel rod surface. This is accomplished by the fluid turbulence the grids provide downstream. This increased turbulence both increases the surface heat transfer coefficient on the fuel rods in forced convection and also promotes mixing of the fluid, decreasing local temperatures.

The other FA components such as end grids, nozzles, and inserts, will affect the hydraulic performance of the assembly. This in turn can affect the local flow rate at the cladding surface. The design for these components includes hydraulic testing to determine the resistance to flow, usually represented by form loss coefficients. The highest hydraulic resistance components are typically the nozzles, particularly debris resistant bottom nozzle designs. While these have little affect on CHF in full cores, they can affect the response in mixed cores of different fuel types or nozzle designs. These components do directly affect the FA liftoff calculations that are part of the design criteria.

The design review should focus on the items that are most likely to affect fuel performance. A good initial approach is to ask the vendor to:

- Identify the components or assemblies with the smallest thermal margins to design limits, whether those be CHF, FA lift, cross flow, or other.
- Identify the areas that the vendor believes have the greatest uncertainties.

9.1.2 Design Criteria

The thermal-hydraulic design analyses are performed to demonstrate compliance to specific criterion (in response to the General Design Criteria or GDC discussed in Section 6). The demonstration of compliance is integral in nature. In this context the phrase “integral in nature” means that the evaluation of the thermal-hydraulic criteria can only be performed as part of the evaluation of the plant performance and requires input from both the nuclear and thermal-hydraulic characteristics of the fuel design. Neither the entire FA, nor its components, can be evaluated independent of the plant behaviour during anticipated operation occurrences or accidents.

It is primarily Criterion 10 that defines the thermal-hydraulic design criteria that must be satisfied by the fuel. This GDC states the fuel design limits are not exceeded during normal operation or AOOs. The primary thermal-hydraulic criterion, or SAFDL, is related to the overheating of the cladding. Proper thermal-hydraulic design of the reactor core and associated systems is necessary to assure that sufficient margin exists with regard to maintaining adequate heat transfer from the fuel to the Reactor Coolant System (RCS). Compliance with GDC 10 provides assurance that the integrity of the fuel and cladding will be maintained, thus preventing the potential for release of fission products during normal operation or AOOs. In PWRs this SAFDL is expressed in terms of DNBR and in BWRs in terms of the CPR.

PWR coolant conditions are primarily subcooled during normal operation and in the low quality region during AOOs. DNB occurs when the fuel rod surface heat flux becomes high enough that significant boiling occurs and a large number of steam bubbles accumulate near the surface of the fuel. This results in a rapid reduction in heat transfer capability and a very rapid temperature rise at the cladding surface.

BWR coolant conditions are primarily in the high quality region during normal operation and during AOOs. In a BWR the flow near the fuel rod is annular with a fluid film on the rod surface and a mixture of steam and liquid droplets in between the fuel rods. Dry out occurs when the heat flux at the rod surface becomes high enough that the liquid film disappears or is entrained into the mixture in between the fuel rods.

The NRC Standard Review Plant Section 4.4 lists two acceptable means of meeting this SAFDL. These two options are as follows:

OPTION A – *For DNBR, Critical Heat Flux Ratio (CHFR), or CPR correlations there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs.*

OPTION B – *The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs.*

In the United States all of the PWRs use Option A above and all of the BWRs use Option B.

In addition, for BWRs Criterion 12 is applicable. This GDC states the core is designed to assure power oscillations, which could result in conditions exceeding acceptable fuel design limits are not possible or are detected and suppressed. The occurrence of power oscillations can lead to excessive localized power peaking, cyclic thermal fatigue, and or degraded fuel surface heat transfer that could exceed fuel design limits and lead to fuel failure. Compliance with this GDC prevents fuel overheating conditions from evolving during normal operation.

The other general design criteria are primarily involved in defining the characteristics of the AOOs and postulated accidents that provide boundary conditions for the evaluation of compliance to the thermal-hydraulic criteria listed above.

10 NUCLEAR DESIGN REVIEW (Jerry Holm and Sten Lundberg)

10.1 General

10.1.1 Introduction

Section 4.3 [NUREG-800, 2007] of the NRC SRP is an excellent background reference for the nuclear design aspects of the FA. The U.S. fuel vendors have developed their designs and design analyses to generally be compliant with the NRC perspective as described in this reference. The NRC guidance is consistent (though not identical) to that provided by regulatory agencies in other countries. The SRP describes an approach that the NRC considers acceptable to satisfy the regulations. Since the SRP is not a regulation itself the fuel vendors do not always comply literally with the SRP.

There are in general no specific regulatory limits related to the nuclear FA design. The SAFDLs that affect the nuclear design are related to the interaction of the nuclear design with the mechanical and thermal-hydraulic performance of the FA, and the criteria that are imposed in those areas. The SAFDLs are specified in Section 4.2 of the SRP [NUREG-800, 2007].

The SRP lists 9 items which are important to the nuclear design and which are related to the FA nuclear design. These elements will be the focus of the audit in the nuclear design of the FA. The primary items of importance to the nuclear design consist of:

1) The design bases required by the appropriate General Design Criteria (GDC) that are specified in 10 CFR 50 Appendix A. The SRP references GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28 in Section 4.3 Nuclear Design. The GDC are discussed in Section 6 of this report.
2) The generation of *power distributions* at various statepoints and exposures for both normal and accident conditions. This aspect includes the power distribution uncertainties and benchmarking of the analytical methods to previous reactor operation.
3) The determination of various *reactivity components* at various statepoints and exposures for both normal and accident conditions. The typical reactivity components are moderator (void in a BWR), Doppler and power coefficients. The moderator temperature coefficient is the change in reactivity due to a change in the moderator (water) temperature. The Doppler temperature coefficient is the change in reactivity due to a change in the fuel (pellet) temperature. The power coefficient is the reactivity change due to a change in power level (which is a combination of the moderator and Doppler temperature coefficients since a change in power affects the moderator and fuel temperatures). The fuel vendors characterize these in different manners depending on the specifics of the methodology used. This aspect includes the uncertainties in the reactivity components. Some of the vendors will assign uncertainties to each reactivity component while others will use conservative values for the reactivity components.
4) The evaluation of both long and short term *reactivity control* effects. This area includes the evaluation of the capability to shutdown the reactor as a function of exposure and initial statepoint conditions.
5) The evaluation of the impact of the allowed *control rod patterns*. The evaluation considers the impact of the control rod patterns on the power distributions and core reactivity.
6) An evaluation of the *reactor critical statepoints* during refuelling and in the new and SFPs.
7) An evaluation of the *stability* of the reactor.
8) An evaluation of the *applicability of the analytical methods* to the FA design.
9) An evaluation of the effect of the power distributions on the *irradiation of the pressure vessel.*

10.1.2 General Nuclear Design Considerations

The reliable nuclear design and performance of the core and the individual assemblies is strongly dependent on the fuel cycle planned for the reload batch and for the individual assemblies in the reactor. This includes the:

- Cycle energy, length and exposure.
- Fuel management scheme and core loading pattern.
- Capacity factor.
- Peaking factor limits.
- Plant Technical Specifications applicable to the fuel and core design.

Functional requirements, such as these, should be defined in the contractual documents and provide bases for the thermal-hydraulic and mechanical design as well. Applicable limits for these parameters should be included in the vendor's design documentation. Any limits licensed with the regulator should be clearly noted as such.

The general nuclear design objectives defined in the contract should include:

- A clear statement of the nuclear performance requirements of the design.
- Fuel reliability and nuclear performance warranties as applicable.
- Applicable design policies including establishment and tracking of design and operating margins.
- Application of the vendor's QA Program and its conformance to a regulatory and national standard.
- All assumptions should be clearly stated.

The reliable nuclear performance of the FA and its components depend on the quantities and geometric distribution of the enrichment, BA and water moderator within and outside of the FA. These design parameters determine the reactivity and power distribution within the core and must maintain margins to the regulatory and Technical Specification limits to assure the reliable and safe operation of the fuel during normal operation as well as transients and accidents.

A variety of parameters that need to be considered in the nuclear design of the assembly and its components are listed here to provide an overview. The list below is a combination of both PWR and BWR related items:

- Design and as-built condition of the components:
  - Fuel enrichment and BA level and distribution within the assembly and the core.
  - Physical arrangement of coolant moderator within the assembly and the core.
  - Control rod (or blade for a BWR) worth.
  - Applicability of the methodology used to design the FA.

- Plant operation effects:
  - Determination of margins to local and general reactivity changes including the moderator void coefficient (BWR), moderator temperature coefficient (PWR), Doppler coefficient and power coefficients.
  - Long and short term reactivity effects including margins to hot and cold shutdown margins.
  - Core stability and its effect on power-flow limits (BWR).
  - Reactivity and criticality of the reactor during refuelling and in the SFP.
  - Power distributions as a function of exposure.
  - The effect of power distributions and the resulting irradiation effects on the pressure vessel embrittlement.
  - Source term for radiological analyses.
  - Flux gradients across assemblies and effects of neighbouring assemblies.
  - Dimensional changes such as bowing of the fuel rods, assemblies or channels (BWR) that change the moderator distribution and consequently the power distribution.
  - Maximum assembly and pin exposure.

The selection of the enrichment levels and distribution, BA levels and distribution and number of assemblies for a reload controls the nuclear behaviour of the core. In this sense the term nuclear behaviour is used to refer to the energy capability of the core, the power distributions and the reactivity characteristics (to moderator, void, Doppler and control rod worth) of the core. The nuclear design is an iterative process where the number of assemblies and their characteristics (enrichments and BAs) are chosen and then the nuclear behaviour is evaluated to determine if the nuclear design is acceptable. If the nuclear behaviour is such that the design is not acceptable then the enrichment levels and distribution, BA levels and distribution, and number of assemblies will be modified in an effort to make the nuclear behaviour acceptable.

The nuclear design has a number of obvious impacts on the nuclear behaviour. An increase in the enrichment for an individual assembly relative to the other assemblies will result in an increase in the radial and total peaking factor for that assembly at Beginning Of Cycle (BOC). A decrease in enrichment will have the opposite effect. An increase in the number of BA rods in an assembly relative to other assemblies will decrease the assembly power at BOC but may result in an increase in its relative power after the BA burns out. An increase in the BA (gadolinia) content in a fuel rod at BOC will not impact the BOC relative power but it will change the exposure at which the BA is depleted and thus the exposure at which the peak power in the assembly will occur. These generalities about the relationship between the nuclear design characteristic and nuclear behaviour are used by the nuclear design engineer to develop the cycle specific nuclear design. The relationships are sufficiently approximate that a detailed evaluation of the nuclear behaviour is required for the final design to assure that the behaviour is acceptable and will not result in unacceptable results when the mechanical and thermal-hydraulic evaluations are performed later in the reload design process.

11 CLADDING PERFORMANCE UNDER ACCIDENT CONDITIONS (Peter Rudling and Alfred Strasser)

11.1 LOCA

11.1.1 Summary Description of the LOCA

The design basis LOCA is a break in a pipe that provides cooling water to the reactor vessel. Analyses are performed for a variety of break sizes and locations to demonstrate that the ECCS can maintain the fuel in a coolable geometry. The limiting break is typically in one of the cold, main coolant pipes of a PWR or one of the intake pipes to the recirculation pump of a BWR.

The LOCA process starts by the decrease and ultimate loss of coolant flow at the same time that the reactor is depressurized, Figure 11-1. The loss of coolant flow decreases heat removal from the fuel, increasing the fuel temperature and causing a significant temperature rise of the cladding. The decrease in system pressure causes an outward pressure differential and a hoop stress in the cladding wall. The result is the plastic deformation, or *ballooning* of the cladding. Ballooning may also result in *fuel relocation*[81] that may impact the cladding temperature as well as the Equivalent Cladding Reacted (ECR) in the later phase of LOCA.

Ballooning of the fuel rods may result in *blockage* of the coolant sub-channel that in turn may impact the fuel coolability. If large fuel clad burst strains occur at the same axial elevation, *co-planar deformation*, in the FA, the coolability may be significantly degraded. Specifically, the clad azimuthal temperature gradient will strongly impact the burst strain. The extent of the ballooning is also dependent on:

- Creep strength of the cladding.
- Stress in the cladding and the corresponding strain rate.
- Temperature and the rate of temperature increase.

Depending on the temperature, the cladding ductility and the rod internal pressure, the cladding will either stay intact or may burst which will allow steam to oxidize the fuel clad inner surface. In addition, some of the hydrogen released by the water/zirconium corrosion reaction inside the burst fuel may be picked up by the cladding resulting in very high local hydrogen concentrations (1000-3000 wtppm H). A fuel cladding with such high hydrogen concentrations will be very brittle even though the cladding is not oxidised at all, i.e. ECR is 0. The fuel clad axial temperature distribution will determine the axial elevation of the ballooned and burst fuel rods in the assembly. The axial and azimuthal fuel clad temperature distribution is a result of the heat transfer mechanisms at the surfaces of the cladding.

[81] Fuel relocation may occur, if during LOCA a section of the fuel rod experiences ballooning, by slumping of fuel fragments from upper location in the ballooned section.

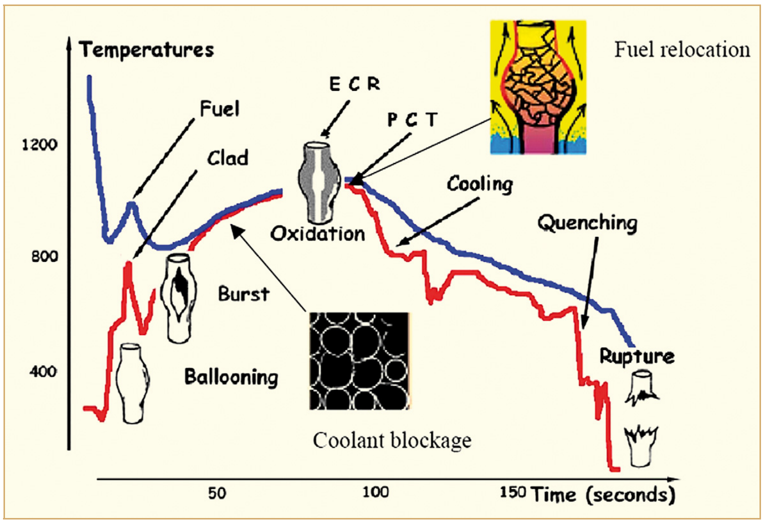

Figure 11-1: Typical LOCA in a PWR.

The increasing temperatures and presence of steam will cause the intact cladding to oxidize on the OD and the burst cladding to oxidize on both the OD and ID (two sided oxidation) until the ECCS is activated and the water quenches the cladding. The oxidation process at the high LOCA temperatures will increase the oxygen and hydrogen content in the cladding, reducing its ductility and resistance to rupture. The process and final structure of the cladding after a LOCA cycle is shown on Figure 11-2:

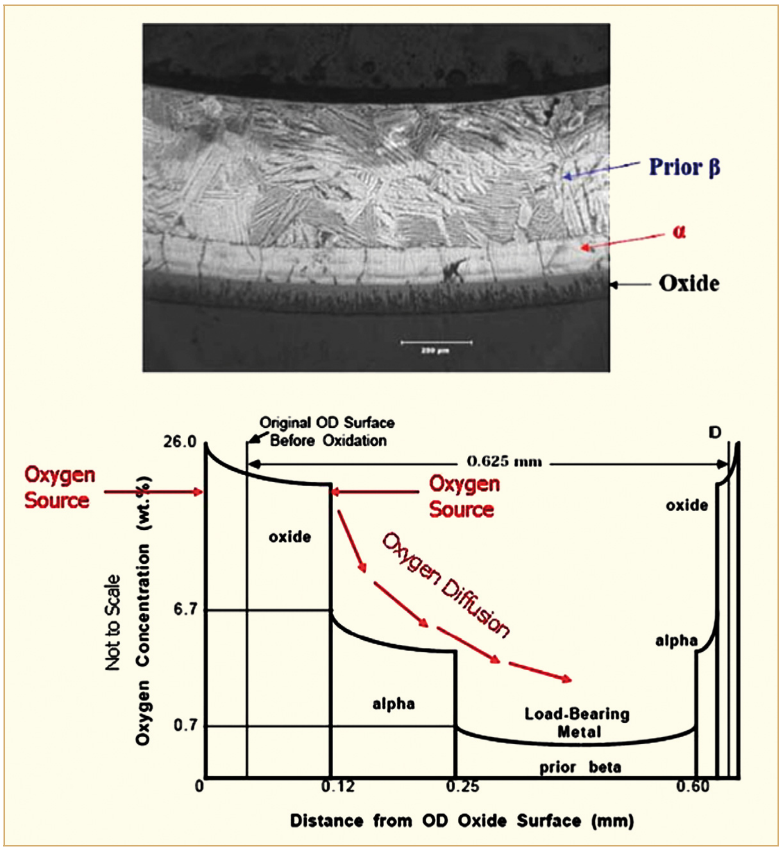

Figure 11-2: Structure of oxidized cladding, [Meyer, 2005].

- First, the high water and steam temperatures increase their reaction rates with the cladding and increase the conversion of the cladding surface into thicker ZrO_2 films.
- As the LOCA temperature passes the levels where α→β transformations start and finish, the resulting structure consists of:
  – The growing ZrO_2 layer.
  – A brittle zirconium alloy layer with a very high oxygen content which stabilizes the α phase, formed by diffusion of oxygen from the oxide layer.
  – The bulk cladding, which is now in the β phase, has a high solubility for hydrogen; the hydrogen picked up by the cladding from the water-metal reaction increases the solubility of oxygen in the β layer.
- The ZrO_2 and oxygen stabilized α layers grow with continued diffusion of oxygen and hydrogen from the water reaction. The increasing amount of oxygen convert some of the β phase to oxygen stabilized α phase with the concurrent shrinkage of the β phase. The remaining β phase cladding wall thickness is transformed to α phase, or "prior β phase", on cooling and is the only structural part of the cladding that can insure its integrity.

During the LOCA oxidation, the first oxide formed is adherent and protective. However, after an extended exposure, *breakaway oxidation* may occur manifested by the change from parabolic to quasi-linear oxidation kinetics and a dramatic increase in HPUF. High temperature oxidation studies have shown that the Zr-Nb materials M5, ZIRLO have similar LOCA oxidation kinetics compared to that of Zry-4, i.e., an intact black oxide layer is preferentially formed which is associated with low HPUF. However, the Russian Zr-Nb alloys E110 and E635 (with similar chemical compositions to that of M5 and ZIRLO) show a much larger tendency to breakaway oxidation and high hydrogen pickup. It appears that the reason for this difference in behaviour is related to that different source material for E110/E635 (iodide/electrolytic Zr) and M5/ZIRLO (Zr sponge) are used. Also, another reason for the difference in behaviour between these two groups of materials is that the clad outer surface is HF pickled for E110/E635 materials while belt-polishing is used for M5/ZIRLO materials.

The sources and role of hydrogen in the embrittlement of the cladding includes hydrogen from the corrosion reaction during normal operation and hydrogen from the reaction with steam during the LOCA. In addition hydrogen increases the solubility of oxygen and diffusivity of oxygen in the β phase at high temperatures. Oxygen, in combination with hydrogen, are the two major elements that cause cladding embrittlement by the growth of the α layer and the shrinkage of the structural, prior β layer.

Integrity of the cladding is based partly on the properties of the former β zone, since the ZrO_2 and oxygen stabilized α zones are too brittle to sustain a load. The embrittlement criteria are based on properties of the prior β layer measured on post-simulated LOCA tests of unirradiated Zircaloy-4, by ring compression tests [Hobson & Rittenhouse, 1972] and related to oxidation, or ECR, calculated by the Baker-Just equation. It should be noted that the oxide thickness of the samples were never actually measured during those tests. The United States Nuclear Regulatory Commission (USNRC) accepts ECR calculations by the Cathcart-Pawel equation or others that may be submitted for approval.

12 REFERENCES

10 CFR Part 50 Appendix A, "*General Design Criteria for Nuclear Power Plants*", U.S., Government Printing Office, Washington, 1990.

10 CFR Part 100.11, "*Determination of Exclusion Area, Low Population Zone, and Population Centre Distance*", U.S. Government Printing Office, Washington, 1990.

Adamson R. B. and Rudling P., "*Dimensional Stability of Zirconium Alloys*", ZIRAT7/IZNA2 Special Topics Report, ANT International, Skultuna, Sweden, ANT International, 2002/2003.

Aksenov V. I., Molchanov V. L., Panyushkin A. K., Samoilov O. B., Lupishko A. N., Zheleznyak V. M., Kuul V. S., "*Results of alternative FA development and operation in Kalinin NPP-1 VVER-1000 reactor*", Proc: ENS, Top Fuel, Stockholm, Sweden, May, 2001.

ANF-89-98PA Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

Baleon J. P., Burtak F., Peyran J. C. and Urban P., "*Framatome ANP Fuel Experience and Development*", ENS/TopFuel, Stockholm, 2001.

BAW-10179PA Revision 6, *Safety Criteria and Methodology for Acceptable Cycle Reload Analysis*, AREVA NP, December 2004.

Bryant J. and Wilburn L., "*Handbook of Software Quality Assurance Techniques Applicable to the Nuclear Industry*", NUREG/CR-4640, August. 1987.

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

Clifford P., "*The US Regulatory Commision's Strategy for Revising the RIA Acceptance Criteria*", Int. LWR Fuel Performance Conference, San Francisco, CA, 2007.

Douglas F., "*Software Quality Assurance Program and Guidelines*", NUREG/BR-0167, February 1993.

Dragunov Yu. G., Seleznev A. V., Vasilchenko I. N., Kobelev S. N., Makarov V. V., Afanasyev A. V. and Puzanov D. N., "*Experimental and Analytical Study of Rigidity and Deformation of Nonirradiated FA for WWER-1000*", Proceedings of the 2004 International Meeting on *LWR* Fuel Performance Orlando, Florida, September 19-22, 2004.

EMF-92-116PA Revision 0, *Generic Mechanical Design Criteria for PWR Fuel Designs*, Siemens Power Corporation, February 1999.

EPRI Technical Report 1008102, "*PWR Axial Offset Anomaly (AOA) Guidelines, Revision 1*", June 2004.

EPRI Technical Report 1015499, "*Fuel Reliability Guidelines: PWR Fuel Cladding Corrosion and CRUD*", March 2008a.

EPRI Technical Report 1015032, "*Fuel Reliability Guidelines: Fuel Surveillance and Inspection*", March 2008b.

Fuketa T., Hochreiter L. E., Montgomery R. O., Moody F. J., Potts G., D., Preuitt W., Rashid J., Rohrer R. J., Tulenko J. S., Valtonen K. and Wiesenack W., "*Phenomenon Identification Ranking Tables (PIRTs) for Power Oscillations Without Scram in Boiling Water Reactors Containing High Burnup Fuel*," NUREG/CR-6743, LA-UR-00-5079, Los Alamos National Laboratory, 2001.

Garzarolli F. et al., "*Corrosion Phenomena in Zr Alloy Fuel Claddings at High Burnups*", 10th Int. Conf, Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Lake Tahoe Nevada., August 5-9, 2001.

Gorman J. A. and Lipsey G. W., "*An Assessment of BWR Fuel Channel Lifetimes*", EPRI-Report NP-2483, 1982.

Hobson D. O. and Rittenhouse P. L., "*Embrittlement of Zircaloy Clad Fuel Rods by Steam during LOCA Transients*", ORNL-4758, Oak Ridge National Laboratory, January 1972.

IAEA Report No 379: "*Design and Performance of VVER Fuel*", IAEA, Vienna, 1996.

IAEA, "*Waterside Corrosion of Zirconium Alloys in Nuclear Power Plants*", IAEA-TEC-DOC-996, 150, 1998.

INPO document, "*Fuel Integrity Excellence: Chemistry and Corrosion Control Methods for Ensuring Fuel Integrity*", April 2007.

Ishijima K. and Nakamura T., "*Transient elongation of a fresh fuel rod under reactivity initiated accident conditions*", J. Nucl. Sci. Technology, 33(3), pp. 229-238, 1996.

Karoutas Zeses E., File Penney A. and Martin Michael L., "*Advanced Fuel Implementation at Calvert Cliffs 1 and 2*", Proceedings of the 2004 International Meeting on *LWR* Fuel Performance Orlando, Florida, September 19-22, 2004.

Kim Yong Hwan, Jeon Kyong Lak and Kim Kyu Tae, "*Advanced Nuclear Fuel Development and Fuel Rod Fretting Wear Evaluation*", Proc. Water Reactor Fuel Performance Meeting, Kyoto, Japan, October 2-6, 2005.

Landry R., "*Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance*", USNRC Memo, NRR/DSS/SNPB, January 19, 2007.

Lespiaux D., Noirot J. and Menut P., "*Post-test examinations of high burnup PWR fuels submitted to RIA transients in the CABRI facility*", Proc. ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, March 2-6, 1997, pp. 650-658, 1997.

Limbäck M., Dahlbäck M., Tägtström P, Jourdain P, Massih A., Sihver L, Andersson T and Witt P., "*Westinghouse Optimised BWR Cladding and Channel Materials*", Proc: ENS, Top Fuel, Stockholm, Sweden, May, 2001.

MacDonald P. E., Seiffert S. L., Martinson, Z. R., McCardell R. K., Owen D. E. and Fukuda S. K., "*Assessment of light-water reactor fuel damage during a reactivity-initiated accident*", Nucl. Safety 21(5), pp. 582-602, 1980.

Meyer R. O., "*An assessment of fuel damage in postulated reactivity-initiated accident*", Preprint to be published, 2005.

Montgomery R. O. and Rashid Y. R., "*Review and Analysis of RIA-Simulation Experiments on Intermediate and High Burnup Test Rods*", ANS-97 Proceedings, pp. 711-720, 1997.

Montgomery R., Waeckel N. and Yang R., "*Bases for RIA Fuel Failure and Core Coolability Acceptance Criteria*", Proc: MIT Colloquium on High Burnup Fuels for LWRs, Cambridge, MA, USA, 2003.

Nakajima I., Teshima H. and Yamada M., "*Improvement and Innovation of Mitsubishi PWR Fuel*", Proceedings of the 2005 Water Reactor Fuel Performance, Kyoto, October 3-7, 2005.

Nakamura J., Nakamura T., Sasajima H., Suzula M. and Uetsuka H., "*NSRR Tests under Simulated Power Oscillation Conditions of BWRs*", Proc: Enlarged Halden Group Meeting, Gol, Norway, 2002.

NEDO-24011PA, *General Electric Standard Application for Reload Fuel GESTAR II*, Global Nuclear Fuel, September 2005.

NEI-96-07 Revision 1, *Guidelines for 10 CFR 50.59 Implementation*, NEI, November 2000.

NUREG-0800, U.S. NRC Standard Review Plan, Section 4.3, Fuel System Design, Revision 3-March 2007.

O'Donnel W. J. and Langer B. F., "*Fatigue Design Basis for Zircaloy Components*", Nucl. Sci. Eng. 20, 1, 1964.

Rudling P., Adamson R. and Garzarolli F., ZIRAT9/IZNA4 Special Topics Report on "*LOCA and RIA in BWRs and PWRs*", ANT International, Skultuna, Sweden, 2004/2005.

Seibold A., Garzarolli F., and Manzel R., "*Material development for Siemens fuel elements*", Proceedings: International Topical Meeting on Light Water Reactor Fuel Performance, American Nuclear Society, April 10-13, Park City, Utah, 2000.

Smirnov A. V., Kusmin V. I., Smirnov V. P., Dubrovin K. P., Bibilasjvili Yu. K. and Saletnih B. A., "*WWER-1000 and WWER-440 Fuel Operation Experience*", Proc. ANS Int. Topical Meeting on LWR Fuel Performance, pp. 31-44, West Palm Beach, Florida 1994.

Strasser A. and Rudling P., "*Fuel Fabrication Process Handbook*", ANT International, Skultuna, Sweden, 2004.

Strasser A., Sheppard K., Epperson K. and Armstrong B. "*Thermal-hydraulic Bases for Fuel Cycle Designs to Prevent AOAs*", EPRI TR-108781, December, 1997.

Troyanov et al., Proceedings of the 2004 International Meeting on LWR Fuel Performance, Orlando, Florida, September 19-22, 2004.

US Nuclear Regulatory Commission Standard Review Plan, NUREG-0800, Office of Nuclear Reactor Regulation, 1981.

Vitanza C., Kolstad E. and Graziani U., "*Fission Gas Release from UO_2 Pellet Fuel at High Burnup*", ANS Topical Meeting on LWR Fuel Performance, Portland, 1979.

WCAP-9272PA, *Westinghouse Reload Safety Evaluation Methodology*, Westinghouse, July 1985.

Wiesenack W. and Tverberg T., "*Thermal performance of high burnup fuel*", ANS Topical Meeting on LWR Fuel Performance, Park City, 2000.

Wiesenack W., "*High Burn-up Issues and Experimental Results*", MIT Colloquium on High Burnup Fuels for LWRs: Benefits and Limitations, January 23, 2003.