Fuel Reliability

Authors

Peter Rudling
ANT International, Mölnlycke, Sweden

Charles Patterson
Clovis, California, USA

A.N.T. INTERNATIONAL®

Ecolabelled printed matter, 3041 0129

Disclaimer

The information presented in this report has been compiled and analysed by
Advanced Nuclear Technology International Europe AB (ANT International®)
and its subcontractors. ANT International has exercised due diligence in this work,
but does not warrant the accuracy or completeness of the information.
ANT International does not assume any responsibility for any consequences
as a result of the use of the information for any party, except a warranty
for reasonable technical skill, which is limited to the amount paid for this assignment
by each Project programme member.

Contents

1 Introduction

Although the performance of water reactor fuel has improved greatly relative to early experience, social, regulatory and operational incentives exist to both maintain the gains that have been achieved and to make further improvements in fuel reliability. The primary objectives in these areas are to increase fuel reliability by:

- Reducing the primary fuel failure frequency and minimize the consequences of fuel failures when they occur and

- Minimizing operational effects due to factors such as fuel assembly and channel bowing, that can affect thermal margins (LOCA, DNB, Dryout) and core control capabilities (control rod insertion).

In this report, reliability is considered in terms of:

- The ability of the cladding and end plugs comprising a fuel rod to isolate the fuel material and fission products from the primary coolant and to maintain the fissile material in the intended configuration. Failure is defined as the loss of the barrier between the rod interior and coolant such that fission products, fuel material or both are released to the primary coolant.

- Assuring that the fuel system[1] dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis.

Poor fuel reliability can have adverse effects on:

- Reduced thermal and safety margins,

- Power generation,

- Outage time,

- Chemistry and radiation monitoring costs,

- Personnel exposure,

- Handling, transportation, storage and reprocessing.

The cost per failure typically ranges from $1 500 000 to approximately $15 000 000 depending upon the type of reactor, the need for power suppression or a mid-cycle outage, reduced cycle length, the cost of replacement energy and the impact of the leaking fuel on subsequent core designs, operation and post-irradiation handling.

The loss of generating capacity is a significant component in the cost of fuel failure. Generating capacity can be lost in power suppression tests to locate leaking fuel assemblies, in power reductions to minimize the risk of degradation of a leaking fuel rod, in mid-cycle outages to remove leaking assemblies and in operating cycles that are cut short by control blade insertions or unacceptably high coolant or off-gas activity. Generating capacity can also be adversely affected in subsequent reactor cycles by non-optimal core loading or operating compromises as a result of action taken to manage a leaking fuel assembly.

---

[1] Fuel system consists of assemblies of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; holddown spring, connections, spacer grids and springs; end plates; channel boxes; and reactivity control elements that extend from the coupling interface of the control rod drive mechanism in the core.

While both BWRs and PWRs/VVERs are capable of operating with leaking fuel rods without adversely affecting safety, degradation of the affected rods after failure (secondary damage or degradation) can increase the release rate of gaseous, soluble and insoluble fission products. In the degradation process, enlargement of the leakage path and exposure of fuel pellets to flowing coolant can lead to oxidation and mass transfer (washout) of fuel particles from the damaged rod to the primary coolant system. These dispersed fuel particles, generally called "tramp uranium", can stick to the surfaces of fuel rods and other primary system components. The particles that deposit in the core continue to fission and release radionuclides directly to the coolant. Depending on the extent of degradation and the amount of fuel washout, long-term activity increases due to tramp uranium can be significantly larger than the steady-state increase from a leaking rod prior to degradation. The increase due to washout continues after the degraded rod has been removed from the core. Activity due to tramp uranium decreases in rough proportion with the fraction of the affected fuel assemblies that are discharged each reactor cycle; e.g., 6–10 years are typically needed to return to pre-degradation activity levels. So, another significant component of the failure cost arises from the long term increase in exposure levels and action necessary to mitigate their effects.

Maintaining and improving fuel reliability requires an understanding of the behaviour of fuel and materials as related to in-reactor conditions and the mechanisms that have been observed to cause fuel failures. A key factor in improving fuel reliability is the identification of the cause or causes of failure. Such information, in turn, requires the examination and analysis of irradiated fuel at reactor sites (poolside examinations), in hot cells and in related laboratories. Thus, to make progress toward ultra-high reliability fuel and to reduce the potential for post-failure degradation, it is imperative to examine both failed and non-failed (reference) fuel. The most cost efficient way to carry out these examinations is to begin with a good understanding of the known mechanisms of failure and degradation and of the principal methods for examining irradiated fuel. With such an understanding, fuel investigation and development programs can be focused on the likely causes of failure or degradation, while unnecessary costly and time consuming work can be minimized. One of the objectives of this Report is to provide such an understanding.

Issues related to fuel reliability are reviewed in this report. That is,

- Section 2 provides an introduction to water cooled reactor designs (PWR/VVER, BWR and PHWR/CANDU), the respective coolant chemistries, the fuel assembly structures and materials.

- Section 3 gives an overview of how the neutron irradiation in the core impacts the properties of fuel and zirconium-based alloy materials and affects the performance of fuel assembly components.

- Section 4 provides information about fuel design criteria and the type of data fuel vendors need for their fuel performance codes.

- Section 5 discusses fuel reliability and the failure characteristics and which type of examination techniques and supporting data are needed to assess the primary failure cause.

- Section 6 describes the mechanisms behind degradation of failed fuel, factors that contribute to degradation and action that can be taken to minimize the likelihood and mitigate the effects of degradation.

- Section 7 provides information on fuel reliability monitoring

- Section 8 guidelines on how to improve fuel reliability

- Section 9 contains a discussion and summary of fuel reliability issues

- Section 10 contains a list of references

2 Reactor and fuel characteristics

2.1 General

There are essentially five different types of commercial water cooled reactors. By design, the water cooled reactors in operation can be separated under thermodynamic aspects into Closed Cycle Systems (PWR, VVER, and CANDU) and Open Cycle Systems (BWR and RBMK) [Riess & Millet, 1994].

The separation into "Closed" and "Open Systems" has consequences when it comes to the interaction between radiation and the coolant. In a "Closed" system, the net decomposition of the coolant can be suppressed by the addition of hydrogen or ammonia, whereas in "Open Systems" radiolysis gas is produced and removed from the coolant.

The primary coolant in Closed Cycle System reactors operates under alkaline and reducing conditions with LiOH or KOH as the agents of pH control. This basic principle has been used for more than 30 years and has only been modified within the established framework of specified values for pH control. Such modifications are called "coordinated", "modified", "elevated", or "constant elevated" Li/B-chemistries.

The primary coolant in Open Cycle System reactors has historically operated under neutral and oxygenated conditions while keeping the cycle as clean as possible. However, based on material integrity concerns, the coolant chemistry conditions have been changed in some reactors to reducing conditions (Hydrogen Water Chemistry (HWC)).

The fuel in water reactors operates in a wide range of environments. Typical operating conditions for varying reactor types are shown in Table 2-1. Additional information on the thermal-hydraulic conditions in BWR and PWR cores is given in Figure 2-1. For reference, the number and relative generating capacities of each type of operating plant are shown Figure 2-2.

Table 2-1:   Design and operating conditions for water cooled reactors

Parameter	Western type PWR	VVER[4] (440/1000) MW	CANDU/ PHWR[1]	BWR	RBMK[2]
1. Coolant	Pressurized H_2O	Pressurized H_2O	Pressurized D_2O	Boiling H_2O	Boiling H_2O
2. Fuel Materials (Pressure tube materials)	Zry-4, ZIRLO[5], DUPLEX, M5, Inconel, SS[3]	E110, E635	Zry-4 (Zr2.5Nb)	Zry-2, Zry-4, Inconel, SS	Zr-alloy E110, (Zr2.5Nb)
3. Average power rating, (kW/l)	80–125	83/108	9–19	40–57	5
4. Fast Neutron Flux, Average, n/cm².s	6–9E13	5E13/7E13	1.5–2E12	4–7E13	1–2E13
5. Temperatures, °C					
Average Coolant inlet	279–294	267/290	249–257	272–278	270
Average Coolant outlet	313–329	298/320	293–305	280–300	284
Max Cladding OD	320–350	335/352	310	285–305	290
Steam mass content, %				7–14	14
6. System pressure, bar	155–158	125/165	96	70	67
7. Coolant Flow, m/s	3–6*	3.5/6	3–5	2–5*	3.7
8. Coolant Chemistry**					
Oxygen, ppb	<0.05	<0.1		200–400	<20
Hydrogen (D_2),					-
ppm	2–4	2.6–5.3	0.3–1	.05-.30	
cc/kg	25–50	30–60	3 to 10		
Boron (as Boric acid), ppm	0–2200	0–1400	–***	-	-
Li (as LiOH), ppm	0.5–6.0	0.05–0.5	1	-	-
K (as KOH), ppm	–	5–20		-	-
NH₃, ppm		6–30			
NaOH, ppm		0.03–0.35			

* Variation from lower to upper part of the core and from plant to plant.
** Zn in ppb quantities may be added for BWRs and PWRs; Pt and Rh in ppb quantities may be added for BWRs.

*** Not in coolant but in moderator

[1]Canadian Deuterium Uranium [CANDU 6 Program Team, 2005];Pressurised Heavy Water Reactor (PHWR), [2]Reaktor Bolshoi Mozhnosti Kanalov (RBMK), [3]Stainless Steel (SS), [4]Voda Voda Energo Reactor (VVER), [5]Zirconium Low Oxidation (ZIRLO), [6]Cladding tube consisting of an outer soft layer and inner layer with high strength - normally high Sn Zry-4

Figure 2-1:   Typical thermal-hydraulic conditions in BWRs and PWRs.
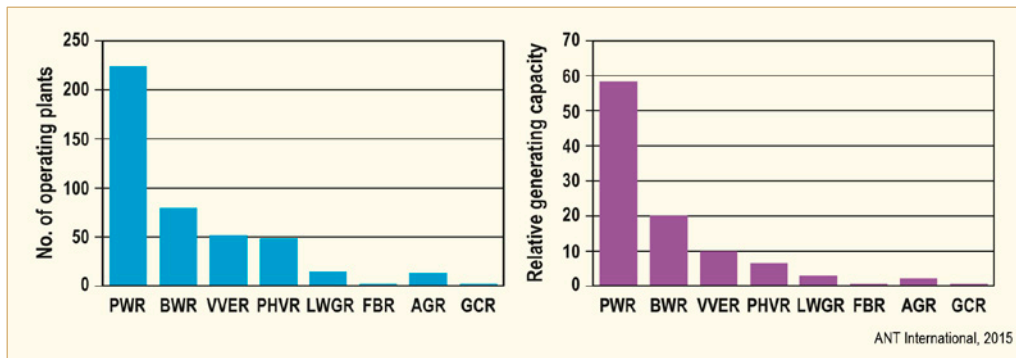
Figure 2-2:   Operating power plants, after [IAEA, 2015].

The reactor types, their characteristics and applied coolant chemistries are described more in the following subsections.

2.2   Reactor designs and coolant chemistries

2.2.1   BWR

One family of BWRs based on the designs of General Electric (GE) have been constructed in the USA, Japan, Taiwan, Sweden, Spain, Switzerland, Mexico and Germany. A second family of BWRs, the RBMK[2] reactors, were developed in Russia and have been constructed only in countries belonging to the former Soviet Union. This particular BWR design is reviewed in some detail in Section 3.1.4 of FMTR Vol. I, [Cox et al, 2006].

---

[2] Reaktor Bolshoi Mozhnosti Kanalov (in English Large Boiling Water Channel)

For power plants with BWRs, the main contribution from GE was the reactor or the nuclear island. The Balance of Plant (BOP) was normally constructed by contractors, who specialize on this part of the Nuclear Power Plant (NPP); e.g., turbines, generators, heat exchangers, etc. The reactor design has been changed and improved during the last five decades by GE itself, but additionally by the licence holders from other countries like Sweden, Japan, Taiwan and Germany. Big changes have been made concerning the recirculation system for the reactor cooling water, the construction materials, the Reactor Water (RW) purification systems, the water separators and steam dryers. The interested reader is referred to LCC7 STR on BWR Reactor Chemistry, Vol. I, [Cowan et al, 2011]. These differences in design can have a significant effects on water chemistry, which is discussed in detail in LCC8 STR on BWR Reactor Chemistry, Vol. II, [Cowan et al, 2012].

As the BOP is often constructed by international companies that normally build fossil-fired power plants, the differences in design of the BOP are much greater than those of the reactor system. However, the designers of the BOP had to keep in mind that the Feed Water (FW) that they supply has a direct influence on corrosion and activity built up in the reactor system.

The BOP is composed of several subsystems. The most important are the FW system, the steam lines, the turbine, and the condensate system. A general overview of a BWR plant is shown in Figure 2-3.
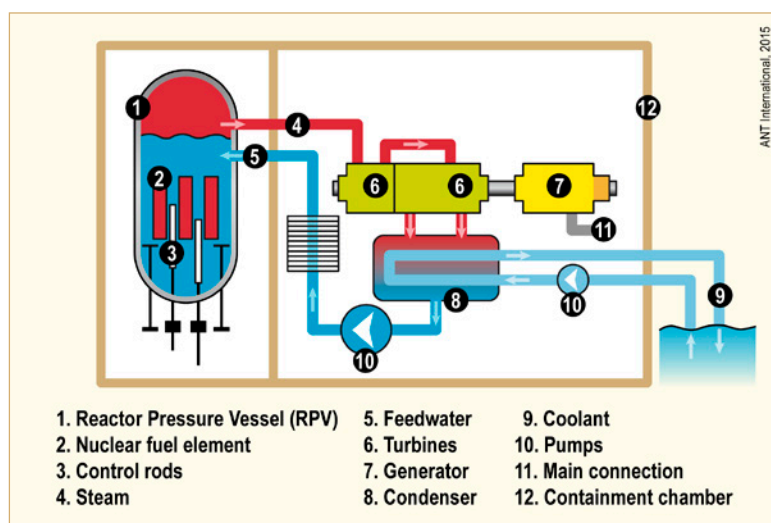
Figure 2-3:   Schematic of a BWR, general overview, after [Steffans, 2011].

There are currently 35 BWRs operating in the United States, 24 in Japan, 4 in Taiwan and 2 in Mexico designed by GE, [IAEA, 2014]. In Europe the 14 operating BWRs were designed by:

- GE-H (GE)[3]: Spain 1, Switzerland 2,

- WES AB (ASEA-ATOM): Sweden 7 and Finland 2,

- Areva (Siemens/KWU): Germany 2.

---

[3] Most of the plants were designed and fabricated before reorganization of the nuclear industry and, for notational convenience, are identified by the names of the respective companies when the plants were developed.

Water chemistry in U.S. and in European BWRs has four common main objectives. These are:

- Ensuring the integrity of the materials against any type of corrosion in all water and steam containing systems and components.

- Minimizing the formation and accumulation of radioactive corrosion products (CPs)

- Ensuring integrity of the fuel.

- Avoiding water ingredients, which are unstable when exposed to radiation, can be activated in the radiation field or are volatile in steam.

The technical approach to water chemistry is different between US and most European BWRs: viz.,

- GE designed plants: HWC (Hydrogen Water Chemistry), Zinc Injection, NMCA/OLNC.

- Siemens and ASEA-ATOM designed plants: NWC

The approaches are driven by the materials of construction, have advantages and disadvantages but lead to similar behaviour of plant materials and comparable fuel reliability.

As there are no indications for sensitization in the core internals of the two German BWRs and as there are no external recirculation loops the "classical" water chemistry has been retained unchanged.

Chemistry issues and areas of concern are similar for all these plants. However, different design and materials concepts favor different chemistry conditions. In particular, the BWRs which are still operating in Germany were designed 1972 and were named "Type Series 72" or "Product Line 72", see (Table 2-2). They have a unique position regarding the materials and chemistry concept.

Table 2-2:   Product lines and station output for Siemens BWRs.

Station	Abbreviation	Product line	MWe	First commissioning
Gundremmingen-B[4]	KRB-II-B	72	1344	1984
Gundremmingen-C[5]	KRB-II-C	72	1344	1984
				ANT International, 2013

Regarding the Water Chemistry in Siemens/KWU and ASEA-ATOM designed plants, the material concept of the reactor part allows to operate with Normal Water Chemistry (NWC) as described below.

As in the case of PWRs, BWR water chemistry has to be optimized between sometimes conflicting requirements[6]. Water chemistry specifications have changed in response to the relative emphasis placed on each of these issues, (Figure 2-4).

---

[4] Planned shutdown 31.12.2017
[5] Planned shutdown 31.12.2021
[6] Degradation of structural materials, nuclear fuel performance and reliability concerns, and control of radiation fields in primary systems
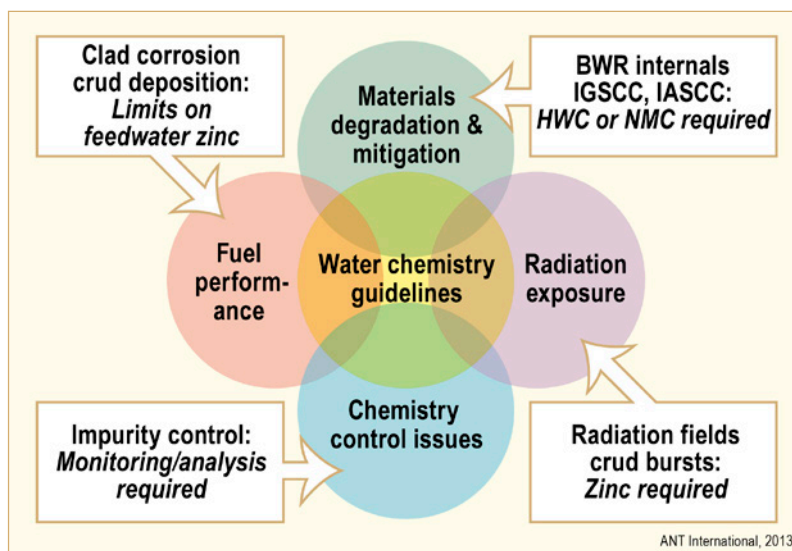
Figure 2-4: BWR chemistry interactions, after [Wood, 2008].

Plant-specific considerations sometimes influence or limit the options for controlling water chemistry, so chemistry specifications differ among plants. Design features and the effects of different operating strategies that have been developed in recent years are important factors in the differences among water chemistries. This is especially true internationally, where significant differences in chemistry strategies between countries exist and are noted below; i.e.,

- Firstly, a key issue facing BWRs in the USA concerns intergranular stress corrosion cracking (IGSCC) of reactor internals, resulting in the implementation of hydrogen water chemistry, with or without Noble Metal Chemical Addition (NMCA) or On-Line Noble Chemistry (OLNC), to ensure that extended plant lifetimes are achieved. German plants use different materials as mentioned before, and do not have the same level of concern. Some Swedish and Japanese plants have replaced reactor internals to minimize potential problems arising from stress corrosion cracking and typically operate with NWC.

- Secondly, BWRs in USA tend to have greater cobalt sources than plants in most other countries, despite strong efforts to replace cobalt-bearing materials. This produces higher out-of-core radiation fields, leading all US plants to implement zinc injection to control fields, whereas only a small number of plants of other designs use zinc.

- Thirdly, the move to longer fuel cycles and increased fuel duty at US plants, while having major economic benefits, has led to new constraints on chemistry specifications in order to avoid fuel performance issues.

Figure 2-5 depicts the changing chemistry strategies over the past 30 years, showing the focus on improving water quality in the early 1980s, the move to reducing chemistry to control IGSCC, which in turn resulted in increased radiation fields, subsequently controlled by zinc injection.
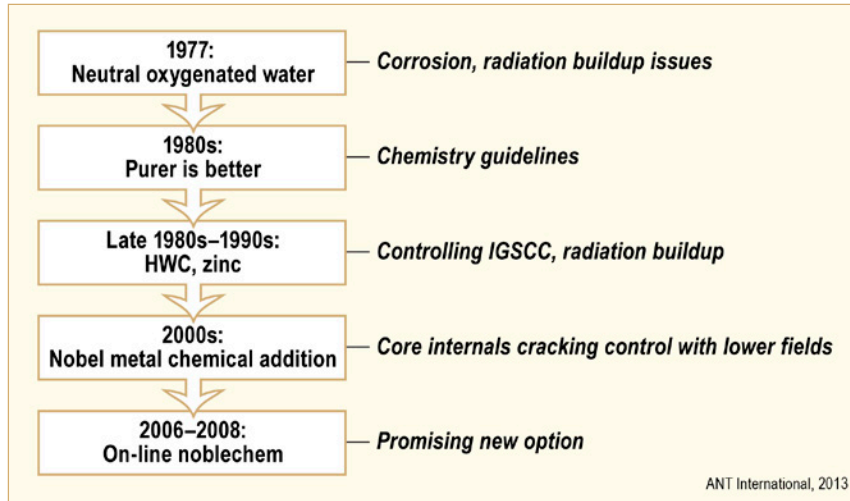
Figure 2-5:   Evolution of BWR chemistry options from 1977 to 2008, after [Wood, 2008].

As mentioned above, concerns about core internals cracking led to the injection of hydrogen at rates which increased over time. Such injections shift the oxygen potential of the core coolant so that the concentration of ^{16}N in the steam and off-gas lines increase in rough proportion to the injection rate. The resulting gamma activity from the decay of ^{16}N (7.13 sec half-life; β^- decay with 6.1 and 7.1 MeV gamma emissions) led to the introduction of NMCA to reduce the need for hydrogen and the resulting operating radiation fields. Figure 2-6 shows the time-based implementation of HWC, zinc and noble metals (NMCA), and on-line noble metal addition (OLNC). The basis, rationale and implications of these developments are discussed in e.g. [Cowan et al, 2011] and [Cowan et al, 2012].

Figure 2-6:   Implementation of zinc injection, hydrogen water chemistry, noble metals chemical addition and on-line noble metal at U.S. BWRs, after [Wood, 2008].

For the reactor part of BWRs, there are in principle three types of design.

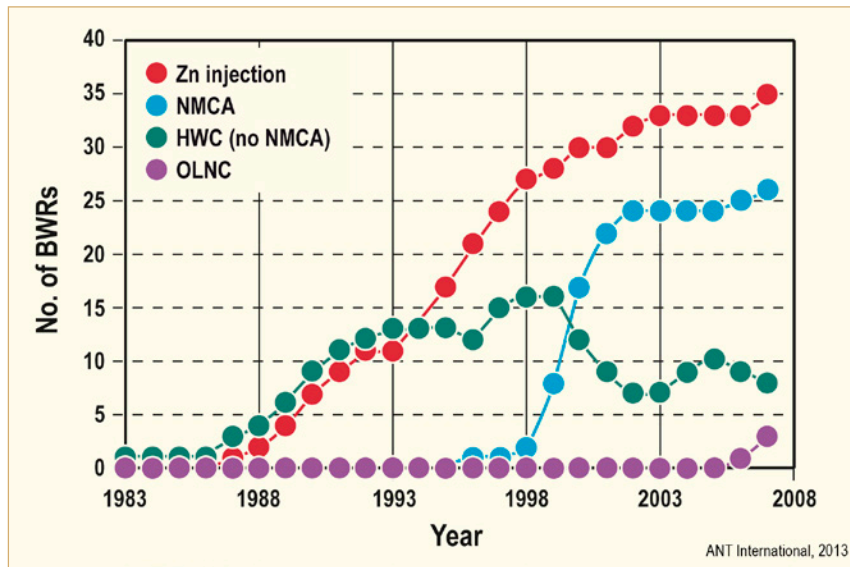- The first generation of BWRs introduced by GE had external recirculation loops. A similar type of BWR design was later used by ASEA-ATOM for Swedish plants. The number of recirculation loops varies between four and six. The main drawback with this type of design from a safety perspective is the size (large-diameter), number and length of recirculation lines in the containment that may be subject to a pipe breaks causing a LOCA situation.

- The jet-pump design was later introduced by GE in order to reduce the number of recirculation loops inside the containment. Typically 1/3 of the core flow is passing the recirculation loops in the containment, and the rest of core flow is driven by the jet-pumps located in the down-comer inside the reactor vessel. The reduced flow rate means that the number of recirculation loops normally is reduced to only two. The dominant fraction of BWRs in US and Japan is of the jet-pump design.

- The third generation of BWRs with internal recirculation pumps was introduced by ASEA-ATOM and Siemens at the end of the 70s. Big recirculation loops in the containment are avoided by installing recirculation pump impellers in the reactor pressure vessel at the bottom of the region between the vessel wall and the outer surface of a shroud which surrounds the core; i.e., the "down-comer" region. The most recent design of GE, the ABWR, has a similar design as the older ASEA-ATOM and Siemens internal pump plants[7]. Some ABWRs have been built and put in operation in recent years in Japan and are under construction in Taiwan. The number of internal pumps varies normally between six and eight dependent on size of the reactor.

Another difference between GE-designed BWRs and other western type BWRs is the materials of construction. In GE plants the inner surface of the reactor system is manufactured from 304 or 316 type of austenitic material. In Siemens/KWU units, all stainless steel components are manufactured from alloys of the 347 type of austenitic material, which includes titanium or niobium as stabilizing elements. However, operating experience revealed that susceptibility to stress corrosion cracking existed in the titanium stabilized material but not in the niobium stabilized material. As a result, all titanium-stabilized material has been replaced with niobium-stabilized material.

In addition to the changes in stabilized 347 material, coolant chemistry was optimized to improve the performance (reliability) of the system materials.

As stated above, the most important difference between US-plants or GE designed BWRs and some European stations (Germany, Sweden and Finland) is the selection of the niobium stabilized 347 material for components to be built from austenitic stainless steel. This material allows the operation of BWRs with NWC. However, it has to be mentioned that the manufacturing of this material as well as the installation, which includes specific welding methods and surface finishing treatments, are important to its performance. That is, both the manufacturing and installation must be controlled to avoid residual stresses or cold work to reduce the tendency for SCC.

---

[7] ABWRs are of similar designs are also offered by Toshiba and Hitachi by way of technology rights associated with corporate realignments.

The materials comprising the turbine condenser tubing and other feed-water heaters also play an important role plant performance, particularly with regard to fuel reliability. Initially, many plants, including all German BWRs, utilized brass tubing consisting of 71% Cu, 28% Zn and 1% Sn. Coolant impurities (copper and zinc) from the corrosion of such tubing was involved in the formation of surface scale (CRUD) on fuel rods which affected heat transfer to produce localized corrosion that penetrated the fuel cladding. These failures resulted from what is identified as the CRUD-Induced Localized Corrosion (CILC) process and led to the replacement of condenser tubing with alternate materials. For example, the tubing of the two German units belonging to type series 72 was changed to SS.

The US industry started developing water chemistry guidelines 25–30 years ago. The early editions of these guidelines presented impurity specifications and required actions if limits were exceeded. When advanced water chemistries were developed and qualified, the guidelines evolved into providing a menu of options, within an envelope of specifications. Guidance is now provided on how to select the optimum plant-specific water chemistry strategy.

The interested reader of this topic is referred to the LCC Reports on BWR Coolant Chemistry [Cowan et al, 2011] and [Cowan et al, 2012].

2.2.2 PWR

Currently, there are, [World Nuclear Association, 2015]:

- 66 PWRs operating in the United States,

    - 48 designed by Westinghouse (W),

    - 12 by Combustion Engineering (CE) and

    - 6 by Babcock and Wilcox (B&W).

    There are several differences between the designs, most notably B&W units having once-through steam generators (SGs) whereas the W and CE units have re-circulating U-tube SGs. In contrast to secondary chemistry issues, these design differences have little effect on primary chemistry strategies.

- In Western Europe there are currently 80 PWRs. The principle design of these stations is from W. They were built by W itself or by the two former licensees which were Framatome and Siemens.

- In Asia 76 PWRs are in operation.

Differences in design and operational experience between the PWRs in the different continents mentioned above are explained in LCC7 STR, [Riess et al, 2011].

In the very early days of PWR operation, heavy CRUD build-up on fuel cladding surfaces was caused by the transport of Corrosion Products (CPs) from the SGs into the reactor core, as for example in the first cycle of the Obrigheim NPP. As a result, activated CPs caused high radiation fields on out-of-core surfaces (Figure 2-7), fuel performance was compromised and even coolant flow issues were observed in some plants.
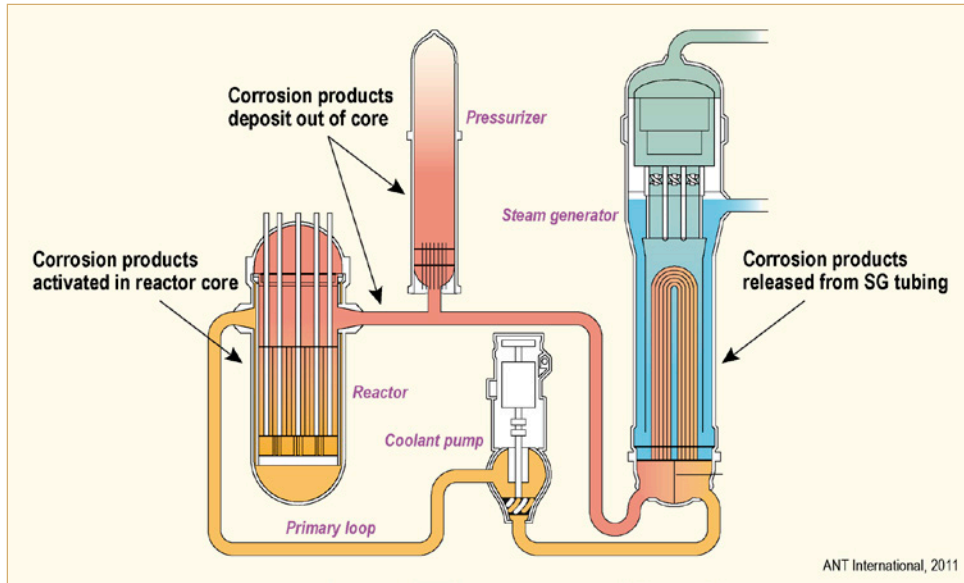
Figure 2-7:   Transport and activation of CPs in PWR primary systems, after [Wood, 2008].

These problems related to CPs were initially mitigated by imposing a hydrogen overpressure on the primary system, to reduce the electrochemical corrosion potential, and raising the primary chemistry pH. Materials degradation in primary systems was then not a major concern, with most of the emphasis focused on secondary side corrosion issues in the SGs. Commercial PWR power plants use a steadily decreasing concentration of boric acid as a chemical shim throughout the fuel cycle, which results in the use of lithium hydroxide to control pH. Some 30 years ago, the concept of "coordinated boron and lithium" was developed, whereby the concentration of LiOH was gradually reduced in line with the boric acid reduction to maintain a constant pH. Furthermore, it was determined that heavy fuel CRUD build-up was avoided if a constant pH of at least 6.9 was maintained. This was possible with 12 month fuel cycles, but fuel cladding corrosion concerns limited the maximum LiOH concentration to 2.2 ppm. As a result, plants often started the fuel cycle with pH below 6.9, which resulted in radiation fields remaining relatively high. Although research and plant demonstrations showed that the 2.2 ppm limit was excessively conservative, the move to higher Li concentrations was initially slow. However, detailed fuel examinations from various plant demonstration programmes have indicated that Li can be raised to as high as 6 ppm without any fuel corrosion issues. In parallel optimizations of the fuel cladding material allow the operation at enhanced pH-values.

About 25 years ago, Pressurised Water Stress Corrosion Cracking (PWSCC) of Alloy 600 Steam Generator (SG) tubes were observed in a few plants, leading to studies on mitigating this effect. Following successful demonstration of zinc injection in BWRs, initial field tests at European and U.S. PWRs revealed, that radiation fields were reduced, and laboratory studies in U.S. indicating that PWSCC was reduced, these facts were eventually confirmed. As a result, zinc injection is now being implemented at an increasing rate, although concerns about fuel performance at high duty plants (plants with significant subcooled boiling) have not been completely resolved. Most recently, especially in the U.S. plants, build-up of boron-containing CRUD in areas of sub-cooled nucleate boiling leading to localized flux depression (Axial Offset Anomaly - AOA[8]) has encouraged the use of higher Li concentrations to minimize CP transport. However, in particular in German PWRs these problems were not experienced because of more iron-rich CRUD. Concerns about the potential adverse effects of zinc deposited in high-CRUD regions has resulted in several highly-rated plants applying in-situ Ultrasonic Fuel Cleaning (UFC) to remove CRUD from the fuel before implementing zinc injection.

The identification of PWSCC in reactor vessel penetrations in the last 15–20 years has encouraged the use of zinc injection, but has also focused attention on the effects of dissolved hydrogen, for which the recommended range has remained in 25–50 ml/kg for 30 years. It now appears that raising hydrogen will reduce PWSCC propagation rates, while lowering it may delay initiation of PWSCC. The interactions of materials, radiation fields and fuels in PWR primary chemistry and optimization issues covered in the Water Chemistry Guidelines, are depicted in (Figure 2-8).
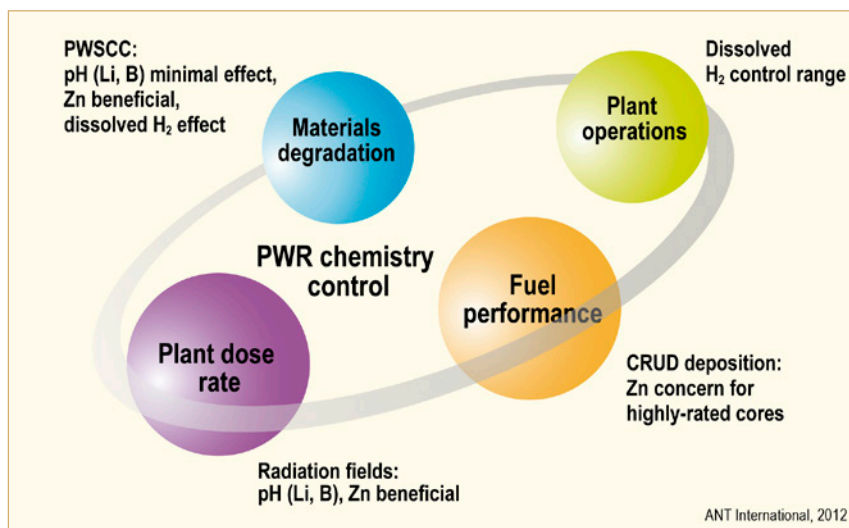
Figure 2-8:   PWR primary chemistry optimization, after [Fruzzetti & Perkins, 2008].

---

[8] AOA, also called Crud Induced Power Shift (CIPS), is a phenomenon, which is caused by boron absorption in PWR fuel crud in the upper part of the core. The boron absorbed causes a reduction of neutron flux resulting in power reduction in the upper core region. In order to maintain overall power, the flux shifts downward in the core, resulting in uneven power distribution in the core. The following three conditions are required for the occurrence of AOA:
- SNB at the fuel clad surface,
- crud deposits in the boiling regions of the fuel rods, and
- boron absorption in these fuel crud deposits.
For more information, see e.g. Section 5.2.3.4 in LCC7 STR on PWR/VVER Primary Side Coolant Chemistry, [Riess et al, 2011].

The evolution of water chemistry control in PWR primary systems in USA over the last 30 years is shown in (Figure 2-9) whereas in Europe the "modified" chemistry prevails, which consists of a constant Li- concentration of max 2.2 ppm at Beginning of Cycle [Sauer & Besenböck, 2008] till reaching a $pH_{300°C}$ of 7.4, which is kept till End Of Cycle (EOC).
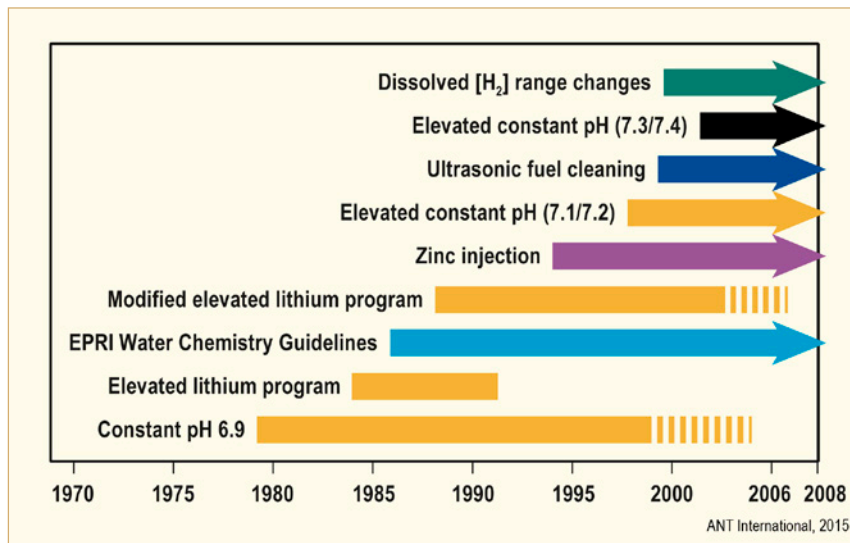
Figure 2-9:   PWR primary chemistry changes at US plants, after [Wood, 2008].

The concerns of the early years of operation are summarized in (Figure 2-10).
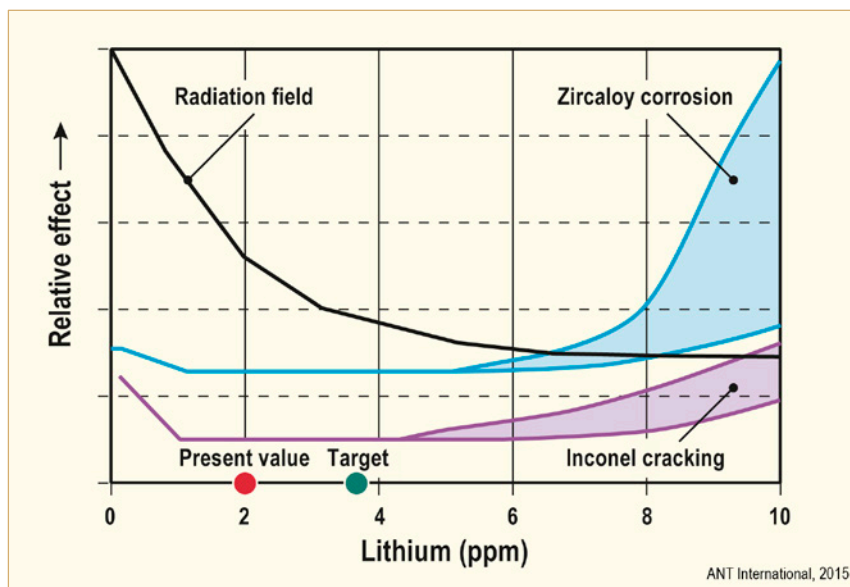
Figure 2-10:   PWR primary coolant chemistry effect of Li-concentration (schematic), after [Riess & Millet, 1994].

The three main factors -pH control, Zinc injection and dissolved hydrogen control- that have dominated PWR primary chemistry strategies in the past, continue to do so today, [Fruzzetti & Perkins, 2008]. Each of these factors is considered from the viewpoint of materials degradation, radiation field control and fuel performance concerns.

In recent years PWR primary chemistry has become complicated by demands of longer fuel cycles, causing, for example, higher initial Boron concentration. In addition, increased sub-cooled boiling and material- as well as fuel- concerns required further optimization of the coolant chemistry.

The key problem, caused by the above mentioned demands, is the existence of a CRUD inventory, which creates problems due to its volume or mass of material and the complex chemical composition of the participating elements. From the fuel element perspective the resulting main area of concern is AOA or CRUD Induced Power Shift (CIPS) especially in the U.S. plants with its use of nickel based alloys as SG tubing material. From a plant operational view the material integrity and the radiation exposure situation may obtain highest attention.

There is a consensus worldwide that an operational success depends on three key factors, which are: Design, material, and chemistry. If one of these factors are deficient, operational problems can result.

With respect to the design of PWRs, the majority of the plants are built on the basis of the Westinghouse design concept. The exceptions are the PWRs built by Combustion Engineering and Babcock & Wilcox. The CE units are different by using SGs that have about twice the size of the heat transfer surface of W SGs and that of its licensees. The B&W design differs by using Once-Through Steam Generators (OTSG), as noted earlier in this section, with a surface area comparable to CE SGs. Despite the differences in design there is a common understanding that it has no consequences for the selection of the plant chemistry.

The chosen material has a great influence on details of the chemistry to be applied. The greatest evidence for such an influence is the material used for SG tubing. This can be either Nickel based Alloys, like Alloy 600 MA[9]/TT[10] and Alloy 690 TT or Alloy 800 NG[11] in some European and Canadian stations, especially in Germany. Details of the differences between the two types of materials are mentioned in Section 2.4.

Early operating experience both in the U.S. and Europe confirm that insufficient control of pH and redox potential leads to enhanced CRUD formation and its negative consequences like fuel CRUD deposits, fuel cladding corrosion, and Axial Offset Anomaly.

The interested reader of this topic is referred to the LCC Reports on PWR/VVER Coolant Chemistry [Riess et al, 2011] and [Riess & Odar, 2012].

2.2.3   VVER

VVER is an acronym for Voda Voda Energo Reactor (VVER) which means a water-cooled, water moderated energy reactor (WWER in English). The VVER reactors belong to the family of the pressurized water reactors (PWRs). The coolant in the primary circuit is kept under pressure to keep it sub-cooled during plant operation. The thermal-hydraulic design of the primary reactor system rules out nucleate boiling in the fuel assemblies.

The Russian VVERs are operating with $KOH/NH_4OH/LiOH$ water chemistry, (LiOH build up is not allowed to exceed 0.5 ppm Li). They nominally operate with a hydrogen over pressure, but the precise hydrogen level is not clearly specified. In the Russian VVERs, NH_3 is added which decomposes radiolytically to N_2 and H_2. In VVERs hydrogen is kept between 2.6 and 5.3 ppm (30–60 cc/kg) and the NH_3 content is typically >5 ppm, [Yurmanov et al, 1997]. Boric acid is added to control reactivity. KOH is added to neutralise the resulting acidic coolant conditions.

---

[9] Milled Annealed
[10] Thermal Treated
[11] Nuclear Grade

The Russian VVERs use carbon steels as material for components and pipes lines while stainless steel is used in the SG tubing. Due to the material selection, the total Ni content in the coolant is very low (0.01–0.1 ppb), [Dickinson et al, 2002]. Also, the total Fe content is normally low in VVERs (0.2–3 ppb), [Dickinson et al, 2002].

Today 50 VVER reactors are in operation in Russia, Ukraine, Hungary, Slovakia, Czech Republic, Bulgaria, Armenia, Finland and China. There are two main designs: 1) the 6-loop designs rated at 440 MWe (1375 MWth) and 2) the later 4-loop designs rated at 1000 MWe (3000 MWth).

The first VVER was a 100 MW reactor in Rheinsberg, East Germany. It was used to test primary circuit coolant technologies. This reactor had zirconium fuel cladding, and used ammonia water chemistry without KOH and without boric acid. Here water chemistry with ammonia injection without direct hydrogen injection was first tested. Stainless Steel was used for both piping and SG pipes. After the technology was tested, 200 MW reactors were built and commissioned at the Novovoronezh NPP, which then served as a foundation for the development of standardized 440 and 1000 blocks.

There are two basic VVER-440 designs. These are the first generation VVER-440s, which includes the initial V-179 design, the V-230 design and the V-270 design with enhanced seismic features. The second generation VVER-440 is standard V-213 design with a full accident confinement system. The primary circuits of the VVER-440, V-179, V-230 and V-213 designs have a Reactor Pressure Vessel (RPV) and six loops, each consisting of a hot leg, a horizontal SG, and a cold leg in which is mounted a Main Circulating Pump (MCP). Two isolation gate valves are fitted to the hot and cold legs of each loop, one between the RPV and SG and one between the RPV and the main coolant pump (RCP). These enable individual loops to be drained for inspection and repair, whilst circulation is maintained in the other loops. There is no separate residual heat removal system and decay heat is removed via the SGs. A pressurizer (compensator tank) is connected to the cold leg of one of the loops and a spray line to the cold leg of the same loop. Typical operating conditions are 297°C (T_{hot}), 267°C (T_{cold}) and 12.3 MPa. All primary circuit surfaces in contact with the primary coolant are either made from SS (main loop pipework, main coolant pumps, SG tubing, SG tube headers (collectors), gate valves and auxiliary systems pipework), from low alloy steel (reactor pressure vessel) or carbon steel (pressurizer, type-22K carbon steel) weld clad with SS. Stainless steel components, pipework including SG tubing and the pressurizer clad are normally made from the Russian-type titanium stabilised SS 08Cr18Ni10Ti, (08X18H10T equivalent to AISI[12] (ANSI[13]) 321). The reactor pressure vessels are made from low alloy steel (15Cr2MFA; Loviisa 12Cr2MFA), weld clad internally with two SS layers.

---

[12] American Iron and Steel Institute
[13] American National Standards Institute

Thirty-two VVER-1000 MWe units have been completed [World Nuclear Association, 2015], Figure 2-11. There are a number of VVER-1000 variants. All the VVER-1000 units have a full containment building [IAEA, 2007]. The VVER-1200 (or NPP-2006 or AES-2006) is an evolution of the VVER-1000 being offered for domestic and export use. The primary circuits of all VVER-1000 designs have a RPV and four loops, each consisting of a hot leg, a horizontal SG, an intermediate leg, a main coolant pump (RCP) and a cold leg. A pressurizer (compensator tank) is connected to the hot leg of one of the loops and the spray line to the cold leg and auxiliary sprays are connected to the charging line beyond the regenerative heat exchanger. VVER-1000 operating conditions are 322°C (T_{hot}), 290°C (T_{cold}) and 15.7 MPa. As in most VVER-440, all primary circuit surfaces are either made from, or are clad in, SS. 08X18H10T SS (08Cr18Ni10Ti, AISI321) is used for the core structures, main coolant pumps and SG tubing, whilst the main loop pipe work and SG collectors are made from type 10GN2MFA carbon steel, clad internally with 08Cr18Ni10Ti SS. The reactor pressure vessel is made from the low alloy steel 15Cr2MNFA, also clad with an inner layer of 07Cr25Ni13 SS and two layers of the niobium stabilised SS 04Cr20Ni10Nb (again similar to AISI 347). Small amounts of other grades of SS and ferritic SS are also present in the core internal structures, but no Stellite hard facing alloys are present in the primary or auxiliary circuits.
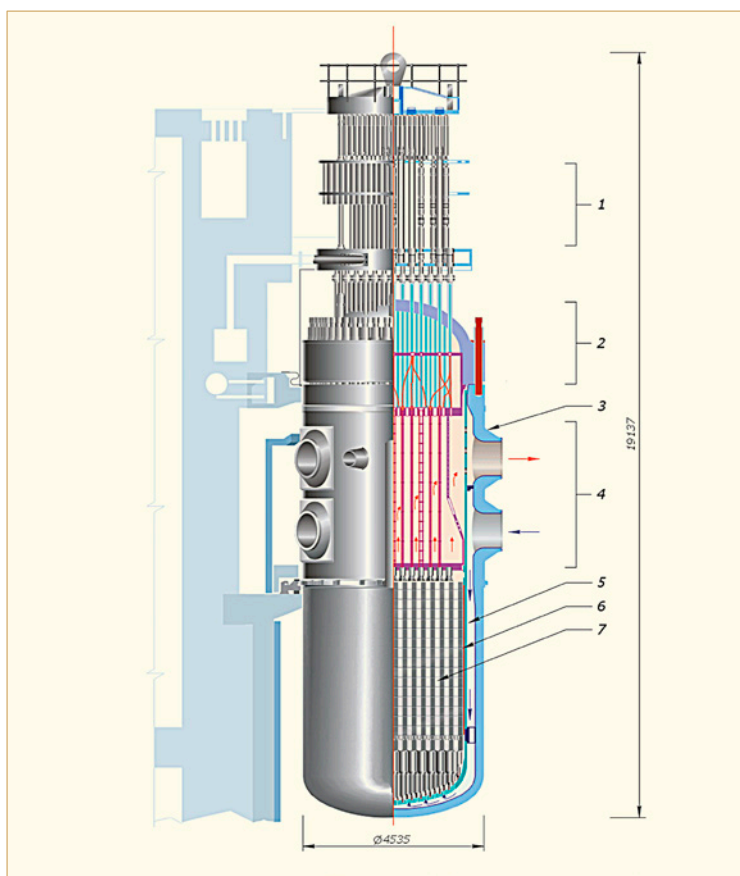

Figure 2-11   VVER-1000 (or VVER-1000 as a direct transliteration of Russian ВВЭР-1000) is a 1000 MWe Russian nuclear power reactor of PWR type [Panther, 2015].

The interested reader of this topic is referred to the LCC Reports on PWR/VVER Coolant Chemistry [Riess et al, 2011] and [Riess & Odar, 2012].

3 Irradiation effects

3.1 Background

The irradiation effects of interest in most water-reactor fuel stem from neutron capture by ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu or interactions with other structural materials; e.g., Zr-alloys. For the fissile nuclides ^{235}U, ^{239}Pu and ^{241}Pu, the nucleus resulting from the capture of low energy (thermal) neutrons is unstable. Upon fissioning, such nuclides generally divide into two fission products and liberate energy. The fission product yield varies slightly with fissile nuclide and energy level, but typically spans the periodic table from about 70 to 170 atomic mass units with statistical distributions centered around mass numbers 95 (a light group) and 140 (a heavy group). The unstable fission and daughter products generally decay by a β$^-$ process. A fission event produces 2–3 additional neutrons, beta particles, gamma rays, neutrinos and thermal energy. The recoverable yield of thermal energy from fissioning and decay is ~200 MeV/f. This corresponds to an average fission density in a fuel pellet of ~2.5E19 f/cm^3 per GWd/MTU.

For the fertile isotope ^{238}U, the capture of intermediate-energy (epithermal) neutrons in a LWR leads to a series of transmutation and decay processes which produces the fissile isotopes of plutonium, Figure 3-1.
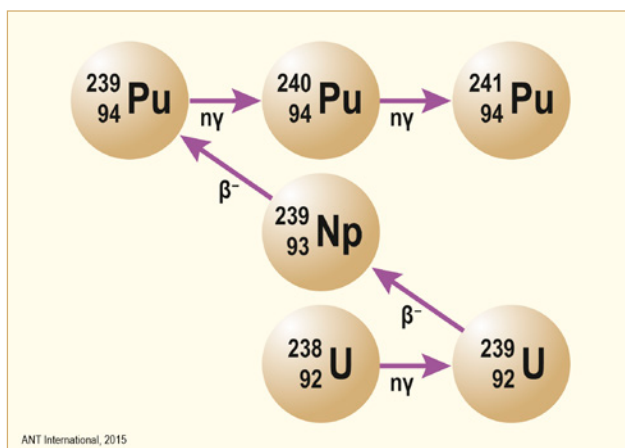
Figure 3-1: Conversion of ^{238}U to fissile isotopes of plutonium in LWR fuel.

This conversion is an important aspect of the design and operation water reactor fuel; it enables fuel in power-generating reactors to operate to greater burnups than are achievable with the limit on fuel enrichment in Asia, Europe and the Americas; i.e., <5% ^{235}U. LWR fuel is now designed to optimize this conversion process by considering water-to-fuel ratios along with other nuclear, thermal and mechanical constraints.

The neutrons produced by fissioning in LWRs span a wide range of energy. As shown in Figure 3-2, the number of neutrons within a given energy range varies, with a peak at ~0.8 MeV and a tail extending beyond 10 MeV. Low energy (thermal) neutrons, ≤0.025 eV, are important to fissioning in thermal spectrum (water) reactors. High energy neutrons contribute to radiation damage of fuel and core materials, with atomic displacements in zirconium alloys beginning at about 40 eV. Damage, which is the displacement of atoms from their normal positions, varies with energy level and cumulative exposure so that a basis for assessing radiation damage is the time-integrated neutron flux (neutron fluence) at energy levels greater than a damage threshold. The energy level for the onset of radiation damage varies among materials and among those studying material behaviour. A common energy threshold in assessments and models of water reactor behaviour is >1 MeV.
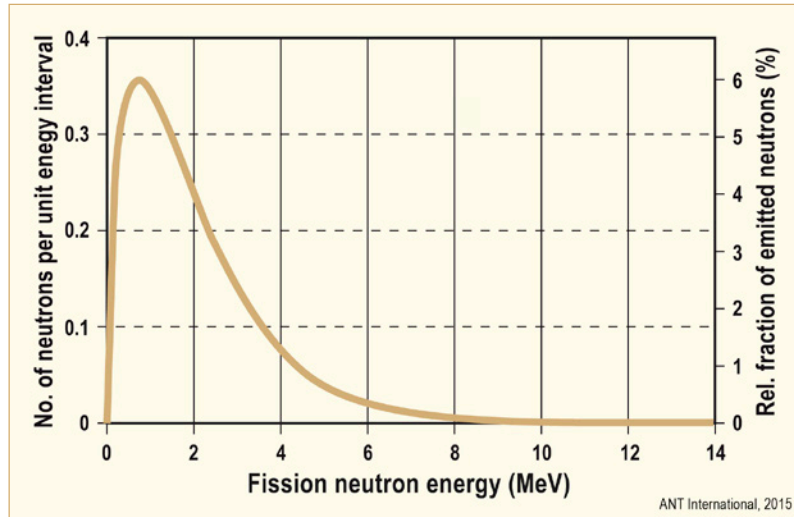
Figure 3-2:   Typical energy distribution from fissioning in a LWR, after [Glasstone & Sesoneke, 1967].

The flux of high-energy neutrons (E>1 MeV) in LWRs varies with core location and conditions, but is approximately 1E14 n/(cm^2 s). This corresponds to a fast neutron fluence of ~1E22 n/cm^2 in LWR fuel cladding at a burnup of 50 GWd/MTU or approximately 20 displacements per atom (dpa), [Adamson et al, 2009]. "Displacements per atom" is an alternate measure of radiation damage which combines the effects of neutron energy, fluence and the response of the affected material. For reference, a fuel pellet will have undergone approximately 1E21 f/cm^3 at 50 GWd/MTU. This fission density corresponds to more than 1000 dpa and is significantly greater than in the fuel cladding due to the combined effects of fission spikes, displacement cascades and other fission-related phenomenon in addition to neutron-induced displacements.

The energy produced by fissioning leads to temperature gradients and heat transfer from the location of fissioning in the fuel to the reactor coolant. For steady-state conditions, the distribution of temperatures across the radius of a fuel pellet is given by

Eq. 3-1:
$$\frac{1}{r}\frac{d}{dr}\left(\lambda\, r\, \frac{dT}{dr}\right) + Q = 0$$

in which

r	= Radial location in pellet at a given axial elevation,
Q	= Heat generation, Q(composition, fission rate, radius, burnup, …),
T	= Temperature
λ	= Thermal conductivity, λ(composition, T, burnup, …).

The temperature distribution is nearly parabolic, but is sufficiently nonlinear that Eq. 3-1 is evaluated by means of nonlinear, numerical methods.

One of the nonlinear factors in the temperature equation is neutron capture at the edge of fuel pellets, a condition identified as "self-shielding". Self-shielding leads to increased fissioning at the surface and decreased fissioning toward the centre of a fuel pellet. This variation depends on a number of factors, including the initial isotopic composition and its evolution during irradiation; e.g., initial and exposure-varying ^{235}U enrichment along with the conversion of ^{238}U to fissile isotopes of plutonium and their consumption by fissioning. As shown in Figure 3-3a, self–shielding in UO_2 under typical LWR conditions increases during operation, with progressively greater fissioning and power generation at the outer ~10% of the pellet radius than in the balance of the pellet. Self-shielding also exists in MOX fuel, but is slightly different than in UO_2 fuel. As shown in Figure 3-3b, fissioning is depressed at the interior of MOX fuel at the beginning of irradiation to a greater extent than in UO_2 fuel. With increasing burnup, the radial distribution in MOX fuel flattens and shifts towards the outer pellet surface. These changes in UO_2 and MOX fuel lead to variations in the power, exposure and temperature across a fuel pellet and to the variations in pellet composition and structure that are discussed later.
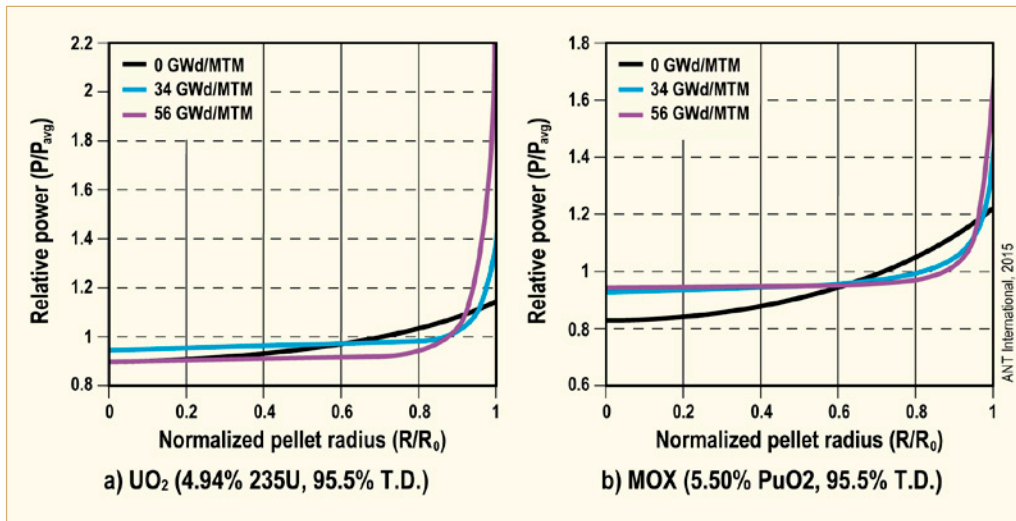
Figure 3-3:   Variation of local power relative to pellet radius and exposure in LWR fuel; after Lemehov et al. in [IAEA, 2004].

Another nonlinear factor in heat transfer and the distribution of temperature across a fuel pellet is the dependence of its thermal conductivity on temperature and local structure; i.e., radiation damage and the accumulation of fission products. The effects of such changes are illustrated in Figure 3-4, which shows UO_2 temperature relative to pellet radius and burnup over the life of a BWR fuel rod. The key points to note in this figure are the steep temperature gradient across the pellet radius and the increases in temperature due (largely) to the effects of irradiation.
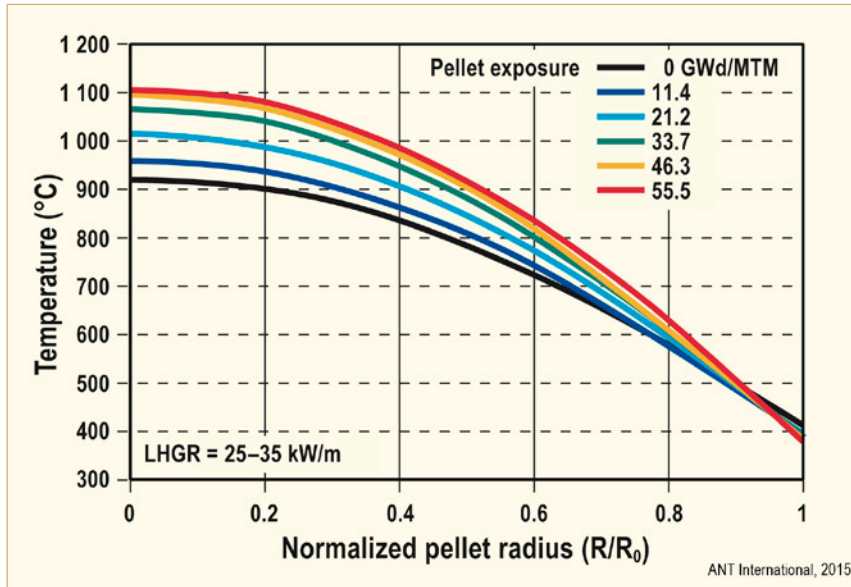
Figure 3-4:   UO$_2$ temperature relative to pellet radius and exposure, after Lemehov et al. in [IAEA, 2004].

Temperatures in Figure 3-4 are calculated at the end of each of five, annual LWR cycles. The values include the effects of differences in power, which ranged from 25–35 kW/m and can be as large or larger than the effects of irradiation. Similar temperature data are shown in Figure 3-5 for two heat generation rates at the beginning of operation. The upper curve is representative of the peak pellet power in typical LWR fuel. The lower curve is representative of the average over the life of a fuel rod. The variations in temperature due to heat generation rates combine in a nonlinear manner with those due to irradiation to produce temperatures to which fuel rods are exposed during normal operation.
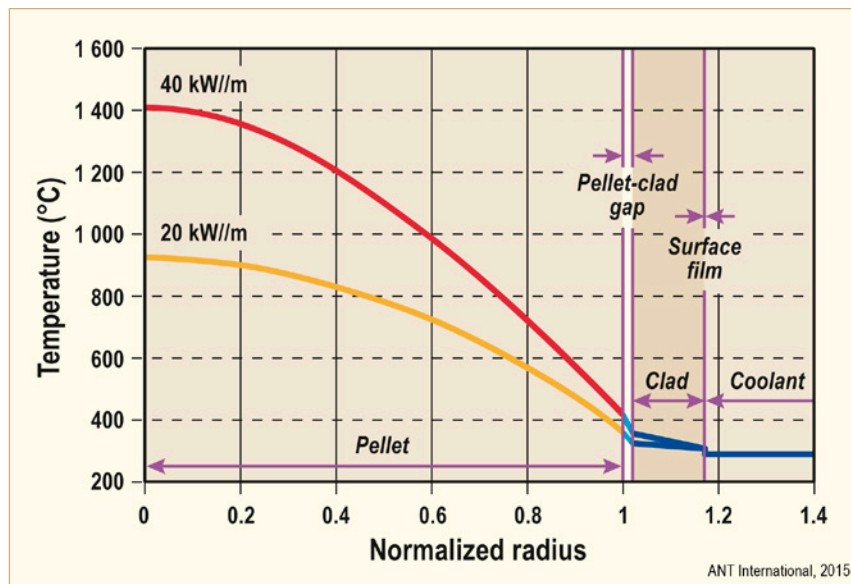
Figure 3-5:   Temperature relative to location across the radius of a LWR rod and average power, based on FRAPCON-3 relationships, after [Geelhood et al, 2011].

4   Fuel design and licensing

The principal aspects of fuel performance which enter into design and operation include:

- Power generation, reactor cycle energy, duration and economics,

- Heat transfer, temperature and stored energy,

- Power margins to fuel melting as related to local heat generation rates and operating history,

- Fission product generation and release, internal gas pressure and margin to limits on gas pressure within a fuel rods or on cladding strain rate relative to pellet swelling (cladding lift-off),

- Dimensional changes of fuel pellets (pellet densification, swelling, cracking, relocation and thermal expansion) as related to pellet-cladding mechanical interaction (PCMI) and cladding deformation,

- The combined effects of temperature, fission product release, PCMI and localized cladding stress on iodine stress-corrosion cracking, cadmium-caesium liquid-metal embrittlement and cladding integrity; i.e., the PCI failure process,

- Fuel stability, corrosion and restructuring related to the release of radionuclides and fuel material to the primary system in the event of cladding failure or under accident conditions such as LOCA or RIA.

As shown in Figure 4-1, these processes are interrelated and depend on the behaviour of the fuel pellets, fuel cladding and a fuel rod as an integrated system.
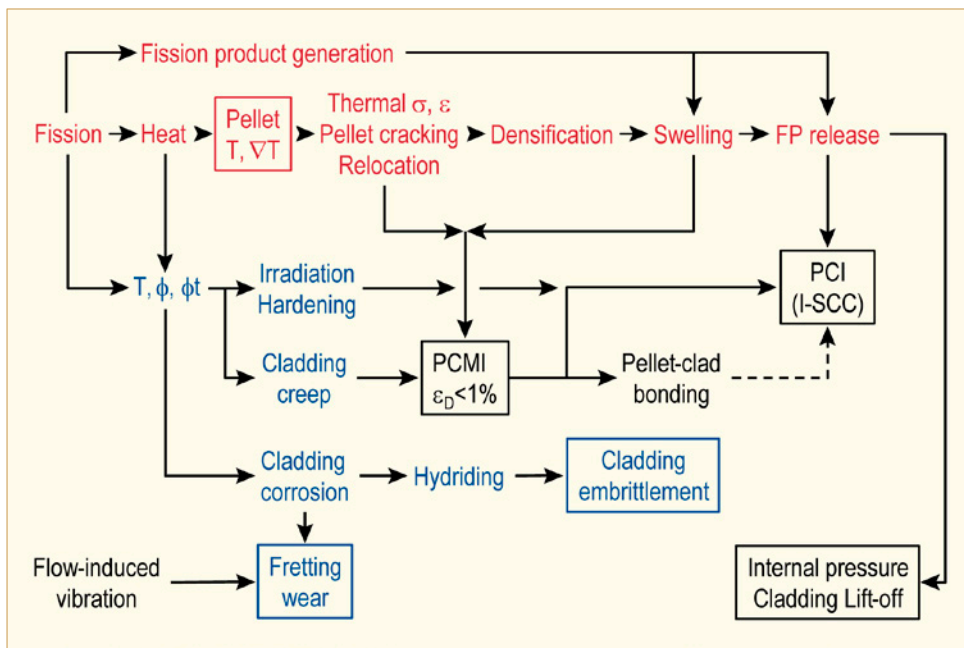
Figure 4-1:   Principal internal, fuel rod processes and their primary interactions. Fuel-related issues are shown in red, cladding-related issues in blue and fuel rod issues in black. Key design and reliability criteria are enclosed in a box, after [Mohr et al, 1976].

5 Fuel reliability

5.1 Failure statistics

The cost of operating with leaking fuel in the core is significant for both the operating plant and the fuel vendor. Table 5-1 summarizes some of the most relevant costs associated with operating a plant with leaking fuel. This table identifies different categories of cost and assigns an estimated minimum and maximum range to the cost in U.S. dollars to the plant operator (utility) and the fuel vendor.

Table 5-1   Estimated cost of leaking core outage, [Power Engineering, 2015].

| Cost category	PWR or BWR	Estimated cost ($K)			
		Utility (1)		Vendor	
		Min	Max	Min	Max
Diagnosis, tracking, analysis during cycle	Both	5	10	10	20
Cost of sipping during offload	Both	60	100	110	250
Cost of Ultra Sonic Testing (UT) of leaking fuel	Both	40	60	150	250
Cost of repair of leaking fuel	Both	100	150	250	500
Cost due to higher dose rates during outage	Both	60	150	0	0
Cost due to additional maintenance of resin beds	Both	120	180	0	0
Cost due to extended outage	Both	0	3500	0	0
Cost due to unplanned shutdown	BWR	0	3500	0	0
Cost of isolating leaker and suppressing	BWR	40	80	0	0
Cost of redesign of core	Both	75	125	0	250
Value of lost energy in fuel	Both	0	150	0	0
Root cause analysis	Both	60	120	50	100
Corrective action implementation	Both	10	50	20	2500
Root cause field exams	Both	10	30	250	500
Hot cell exams (2)	Both	0	500	0	1500
					ANT International, 2015

[Beale, 2014] reported on the information in Fuel Reliability Data Base maintained by the Electric Power Research Institute (EPRI) which includes all 100 US plants (65P/35B), 24 European (9B/7P/6VVER), 45 Asian (15B/30P), 2 South African (P) and 2 South American plants (P). In terms of reactor cycles the data base includes over 700 US and about 100 non-US PWR cycles and 400 US and about 100 non-US BWR cycles [Beale, 2014]. This is currently the most comprehensive and updated fuel reliability database in the industry.

The major cause of fuel failures in PWRs continues to be grid-to-rod fretting (GTRF), a cause that has been sharply reduced by improved spacer and lower nozzle designs. Once all plants have transitioned to GTRF resistant components, this failure mode is expected to be eliminated. A very low level of failures due both to debris fretting and fabrication related failures have occurred (Figure 5-1).
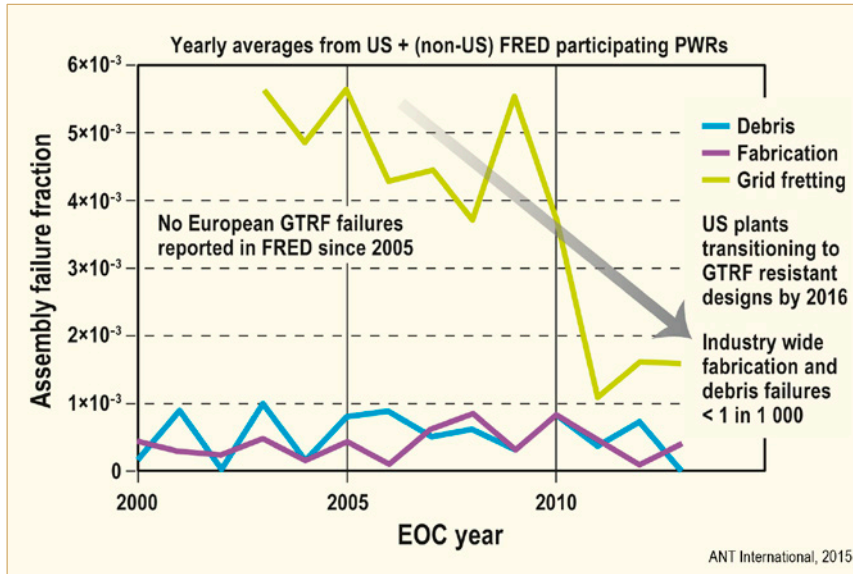
Figure 5-1:   Fuel failure cause trends in FRED base PWRs, after [Beale, 2014].

The fuel failures in BWRs have been almost exclusively due to debris fretting during the past year, at a rate higher than in PWRs (Figure 5-2). An interesting comparison of debris failures in forward pumped plants with failures in full-flow demineralizer plants indicated a forward pumped plant is almost 3 times more likely to shut down with a debris failure (Figure 5-3). The forward pumped plants drain about 1/3 of the feed-water from the turbine directly into the feed-water system without going through the condenser and demineralizer in order to improve the thermal efficiency. The demineralizers are designed to remove ionic impurities in solution, but act as mechanical filters as well, so that they assist in removing some debris. Erosion of the forward pumped drain piping could also add debris to the coolant.

The FRED data base also shows that the probability of a debris failure is 50% higher in a cycle with a power uprate in either plant type. The debris failures tend to increase in the first cycle of the fuel and then plateau out in the second cycle [Beale, 2014].
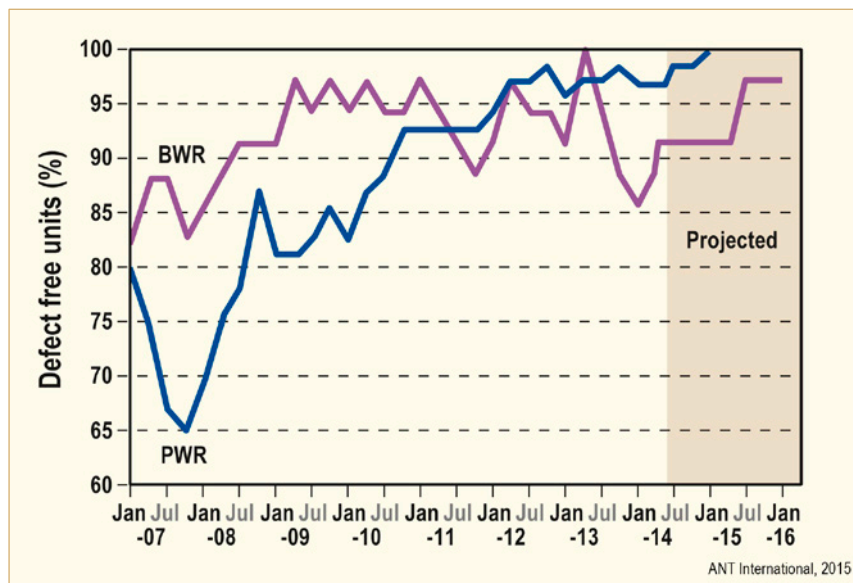
Figure 5-2:   Comparison of debris failure free PWR and BWR units, after [Beale, 2014].
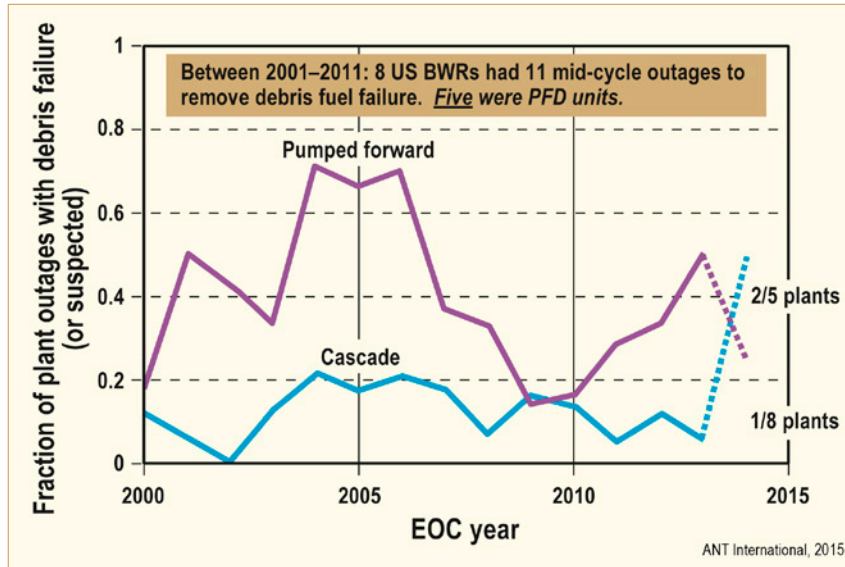
Figure 5-3:   Comparison of debris failures in forward pumped and full flow demineralizer plants, after [Beale, 2014].

Fuel can be damaged and made inoperable without experiencing a clad breach. A summary of these events in PWRs and BWRs by Beale, indicate a very low level of occurrences for the current and past year. That is, assembly bow and mechanical damage occurred in PWRs as well as one crud induced power shift (CIPS). BWRs had debris and fabrication issues and in addition experienced persistent channel bow resulting in control rod interference (Figure 5-4).
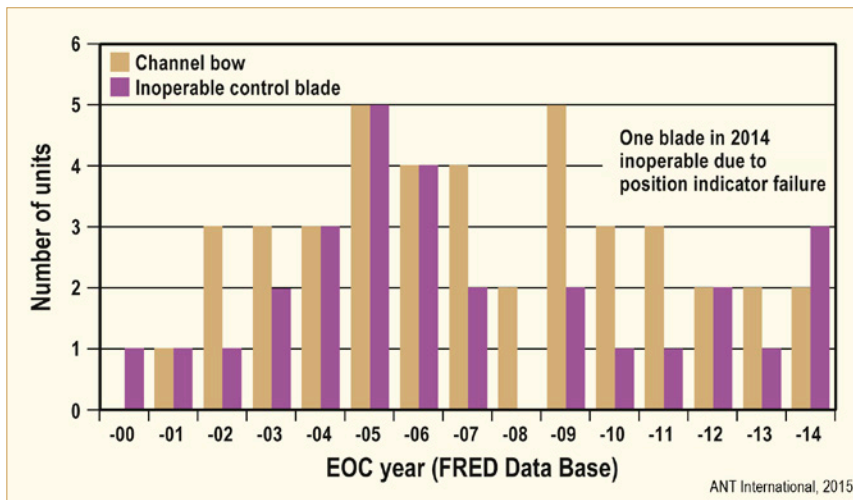
Figure 5-4:   BWR channel bow and inoperable control assembly incidents, after [Beale, 2014].

In [IAEA, 2010] fuel reliability in different types of reactors are reported and a summary is given in the following.

Figure 5-5 and Figure 5-6 show an overall downward trend in the number of fuel leaks in PWRs. It is noteworthy that Japan has a leak level of practically zero for the entire period 1994–2006. Figure 5-7 shows that grid to rod fretting is the dominant fuel rod leaker mechanism in PWRs worldwide. The second most common cause is debris related failures. The large fraction of unknown failure causes is due primarily to the absence of failure investigations. *However, the recommendation of the authors of this report is to always try to assess the failure cause to enable actions to improve fuel reliability.*
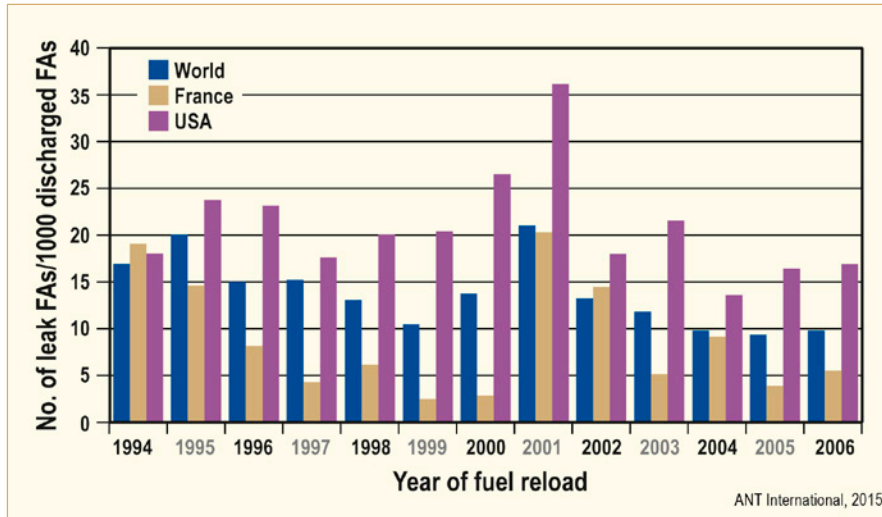
Figure 5-5   PWR fuel assembly leak rate for France, the United States of America and worldwide, after [IAEA, 2010].
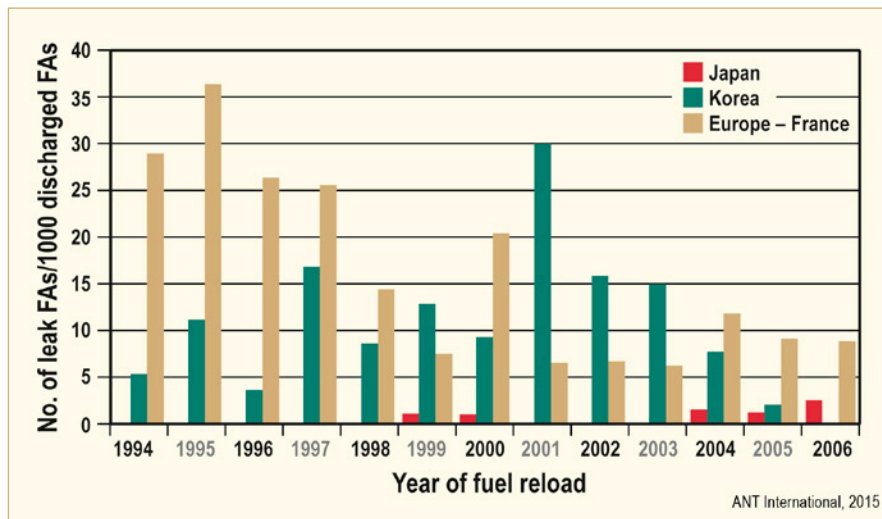
Figure 5-6   PWR FUEL ASSEMBLYleaker rate for Europe with the exception of France, Japan and Republic of Korea, after [IAEA, 2010].
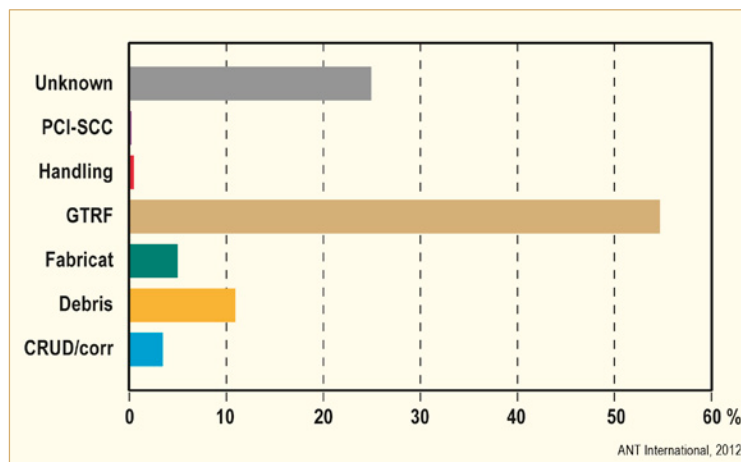
Figure 5-7   PWR fuel leak causes worldwide in 1994–2006, after [IAEA, 2010]

6 Secondary degradation of failed fuel

A failed LWR fuel rod may continue to operate without degrading beyond the initial source of leakage (primary failure) or may degrade by developing long axial cracks, splits[33], transversal breaks, loss of sections of cladding due to hydriding or combinations of these damage modes. Such post-failure damage is identified as secondary degradation and is a topic of concern because it may result in elevated primary-system activities due to fuel washout. This may occur if the rod degrades to such a point that the water gets in contact with the fuel pellets. Steam will not result in fuel washout while the water phase can. Normally, utilities are much more concerned about fuel washout than high iodine and noble gas release, because it may take up to 10 years to clean PWR and BWR cores of the tramp uranium resulting from the fuel washout. On the other hand, the high iodine and noble gas activities released directly from the failed rod will be eliminated at the point when the failed rod is extracted from the core.

Two principal types of degradation have been identified, namely the development of:

- Transversal breaks (also called guillotine cuts or circumferential break) occurring in *BWR*s, *PWR*s and *VVER*s, Figure 6-1 and,

- long axial cracks (axial splits), which can occur in *BWR*s due to the movement of control blades but may also occur in *PWR*s that are subjected to significant control rod movements during operation, Figure 6-2.

Both types of degradation can also involve the loss of sections of cladding by hydriding that is more extensive than that shown in the right image in Figure 6-1. Note that such hydriding can take place at locations away from the primary failure site due to reactions of steam with the cladding first to oxidize the inner surface near the point of ingress and then form hydrides at locations far enough from the point of in-leakage for the oxygen-to-hydrogen ratio in the steam to be insufficient to support a stable oxide film.
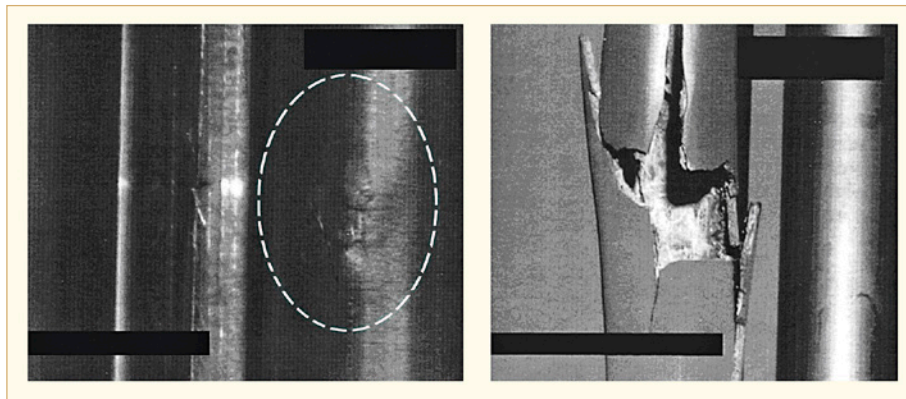
Figure 6-1:   Visual inspection of failed rod. Left picture shows the debris fretting failure at spacer no.6, the right picture shows the transversal break between spacer no. 1 and 2.

---

[33] *Axial split* is a term introduced by *GNF* and represents a failed rod that either has an off gas level larger than 5000 µCi/s or a total crack length that is larger than 152 mm (6 inches).

7   Fuel reliability monitoring

Reliability monitoring provides a means of assessing the state of the fuel during operation, managing the core to minimize the risk of secondary degradation, configuring the next core loading and planning the coming refueling outage. As indicated by the range of tasks and the associated costs in Table 5-1, the effects of fuel failures extend throughout plant operation both during and after the cycle in which failure occur, with combined costs to the affected utility and fuel vendor of $1.4 M to $12.3 M. Reliability monitoring during reactor operation is intended to minimize these effects and is based on analyses of the activities of radionuclides in the primary system (primarily xenon, krypton and iodine) with respect to composition and trends relative to time and operating conditions.

On-line monitoring of fuel reliability is performed at different levels of detail. The measurement and assessment of activity levels relative to radiological limits is a high level form of monitoring. The "fuel reliability indicators" (FRI) of industry organizations such as the Institute of Nuclear Power Operations (INPO) and the World Association of Nuclear Operators (WANO) are another form of high level monitoring. These indices are based on the activity of iodine in the primary coolant for PWRs and offgas activity for BWRs with corrections for the effects of core power, coolant cleanup and background activity (tramp sources; discussed below). Such industry indices are designed to indicate the presence of leaking fuel in operating reactors, for comparisons with license limits and comparisons among utilities and reactors. Other, broader-scope monitoring methods also indicate the presence of leaking fuel and can provide varying degrees of additional insight as to the number of leaking rods, their power, exposure, the condition of such fuel and its evolution during operation. The focus of this section is on the later class of monitoring methods.

The composition of fission products in the primary coolant and offgas changes with the occurrence of a leak in a fuel rod and varies with the condition of the leak and with operating conditions. Failure by any of the mechanisms discussed in Section 5.2 produces a leakage path between the primary coolant system and the interior of the affected rod. At the time of perforation, coolant enters a rod and converts to the vapor phase or gas escapes to the primary system to establish equilibrium between pressures inside and outside of the rod. If the leakage path is small, gasses continue to escape from the rod void volume (open volume within fuel rod) by diffusion, while additional water enters the rod to maintain pressure equilibrium as vapor in the rod is consumed by reactions with the fuel cladding. The escaping gases include a mixture of xenon and krypton with an isotopic composition that depends on their source, rate of production and release, half-life and decay time. If the leakage path is large, gases, soluble fission products and coolant are exchanged by a combination of diffusion and mass transfer. This progression is shown schematically in Figure 7-1. As the size of the leakage path increases, the rate at which coolant flows into and out of a leaking rod increases. Similarly, the time needed for radionuclides to escape to the coolant decreases and the concentration of soluble nuclides such as iodine and cesium in the primary coolant increases. As a result, the isotopic composition of the fission products in the coolant reflect conditions associated with a leaking fuel rod. Note also that plant maneuvers such as shut downs, restarts or power changes also affect the release of radionuclides to the primary system and can provide additional insight regarding the presence, location and state of a leaking rod.
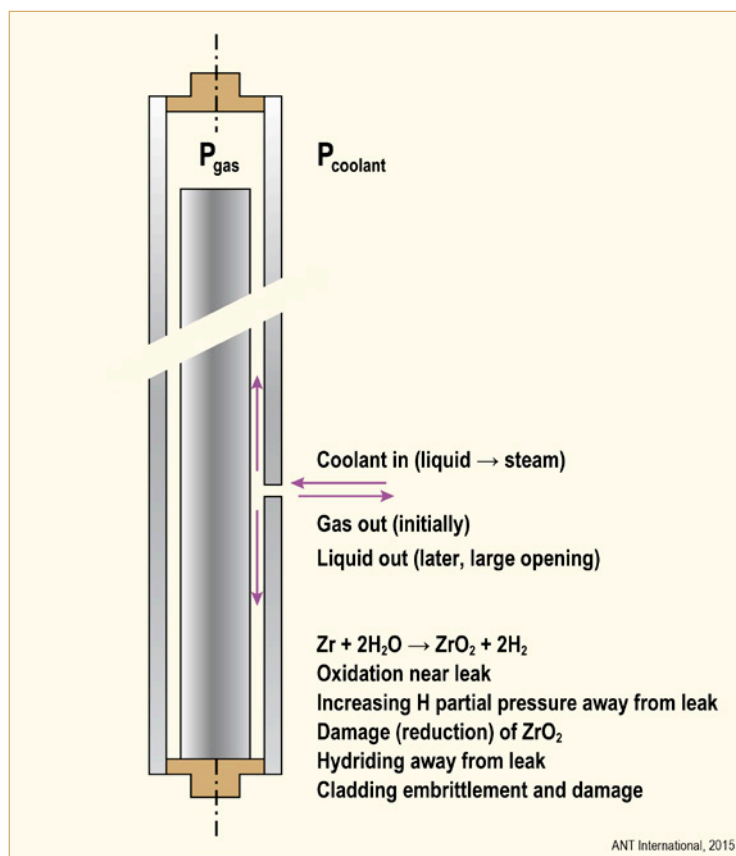
Figure 7-1:   Schematic view of exchange of primary coolant and radionuclides in a leaking fuel rod.

Monitoring fuel reliability typically involves sequential, radiochemical measurements of the primary coolant, gas extracted from the primary coolant and the steam line in BWRs. Measurements of the gross (total) beta or gamma activity from the primary coolant or offgas by means of sensors built into the reactor system provide early indications of fuel failures and secondary degradation, but are generally insufficient for monitoring fuel reliability. Detailed assessments are usually based on spectrographic measurements of gamma activity to determine the composition and activities of radionuclides with respect to magnitude and trends relative to time, burnup and plant maneuvers. The sources of such data are generally discrete samples of the primary coolant and offgas (BWR) or extracted-gas (PWR) that are processed in a gamma activity measurement laboratory (counting room). In some cases, on-line measurements are performed with flowing samples routed through a γ-spectrometer located near the coolant or offgas streams; e.g., [Sihver et al, 1999]. The radionuclides of interest include xenon and krypton in the coolant and gas samples, iodine in the primary coolant and other nuclides that appear in the coolant with a large leak or after secondary degradation; e.g., Cs, Sr, Np, Rb, etc.

An example of monitoring fuel reliability in a BWR is shown in Figure 7-2. This figure is a time-based plot of reactor power and the γ-activities of a long lived fission gas, ^{133}Xe(5.25d) and the sum of six fission gases with relatively long half-lives (Sum-of-Six; discussed later). In this case, the reactor was operating at full power when spikes in the ^{133}Xe and Sum-of-Six activities indicated the failure of a fuel rod. These activity data came from spectrographic measurements of periodic samples. The initial spikes are due to the release of fission gas stored in the void volume of the leaking rod prior to failure. Variations in the relative magnitudes of ^{133}Xe and the Sum-of-Six activities represent changes in the state of the leaking fuel. The large initial increase in ^{133}Xe activity was due to its longer half-life and the inventory of the gas in the rod void volume. The decrease of ^{133}Xe activity relative to the Sum-of-Six activity reflects depletion of the initial inventory of the long-lived nuclide and the progression toward a mixture more representative of the production rates of the six gaseous nuclides; i.e., greater concentrations of the shorter-lived isotopes. The subsequent increases in activity while at constant or decreasing power resulted from what was ultimately found to be the formation of a long axial crack in the fuel cladding and the transport (washout) of fissile material through the crack, Figure 7-3. Power suppression tests were conducted to locate the fuel assemblies with leaking rods. The leaking assemblies were removed during a refueling outage at the normal end of the operating cycle. The plant began the next cycle with elevated coolant and offgas activity due to fuel material that had been transported from the leaking rod and was subsequently entrained in the CRUD on fuel rods that were in the core when washout occurred and then continued operation in the next cycle.
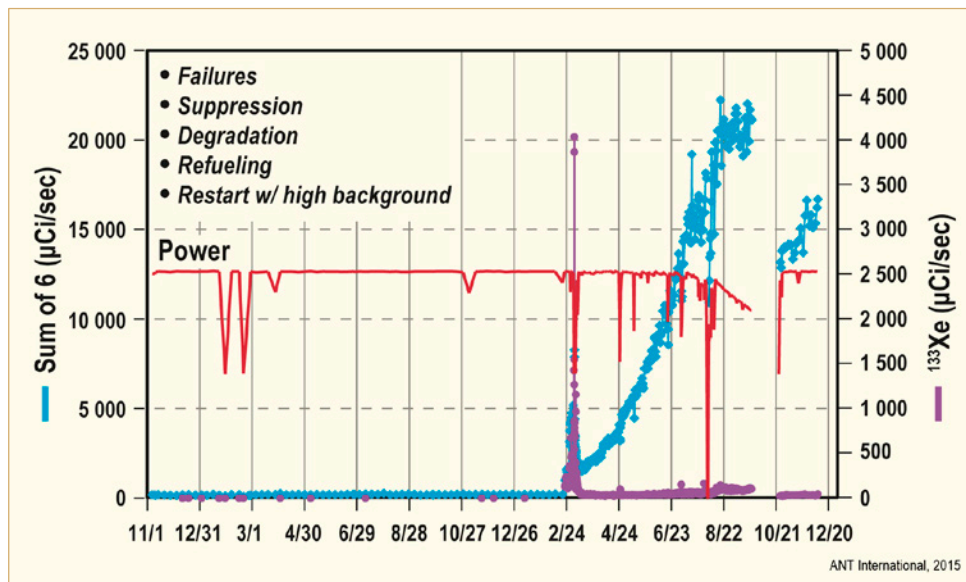
Figure 7-2:   BWR fuel failure and resulting changes in offgas activity, after [Yeager & Schneider, 2005].
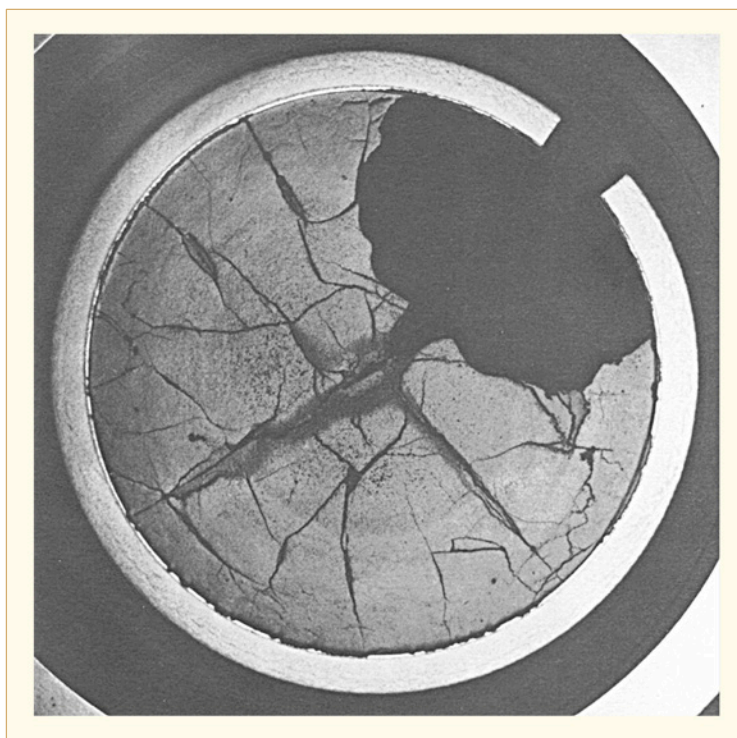
Figure 7-3:   Secondary degradation of a Zr-lined, BWR rod with a long axial cladding crack, [Yeager & Schneider, 2005].

The fission products most useful in monitoring fuel performance are identified in Table 7-1. This table lists fission gas and iodine isotopes in the order of their decay half-lives; i.e., shortest half-life to longest-half-life. It also includes their production rates and approximate steady-state inventories.

Gamma activities of fission gases generally provide the first sustained indication of a new failure because they can escape through small leakage paths. The fission gas isotopes ^{138}Xe(14.1 m) through ^{133}Xe(5.25 d) decay slowly enough to be collected and measured in counting rooms. The sum of the activities of these isotopes constitute the Sum-of-Six mentioned above. The activities of shorter-lived isotopes, ^{89}Kr(3.15 m) and ^{137}Xe(3.82 m), are sometimes measured in addition to those of the longer-lived nuclides by means of γ-spectrometers connected to flowing sample lines. They decay too rapidly for assessment in discrete samples, particularly since the transport times to normal sample points are in the range of 3–6 minutes.

Soluble fission products such as the isotopes of iodine listed in Table 7-1 are used alone and in conjunction with fission gas to identify and track the state of leaking fuel rods. The presence of iodine and other soluble nuclides indicates mass transfer of coolant into and out of one or more leaking fuel rods. Soluble fission produces are generally observed only with large leakage paths, such as those associated with secondary degradation. The exception is the occurrence of spikes in iodine activity which can occur during shutdown, startup or other transients involving pressure changes. When observed in measurements of coolant activity, the range of decay rates of the five isotopes starting with ^{134}I(52.6 m) through ^{131}I(8.04 d) provide a means for assessing the state of leaking fuel that is similar to the gaseous radionuclides. That is, the long-lived isotope ^{131}I is observed in greater relative proportion to the shorter-lived isotopes in a new failure or in the case of a restrictive leakage path. The ratio of long-to-short lived isotopes decrease with depletion of the inventory of iodine in the rod void volume and with increasing rates of release to the coolant.

8 Fuel reliability improvement

8.1 Introduction

Key to improve fuel reliability is to determine the fuel failure cause even for discharged fuel. Only a complete understanding of the failure root causes may improve fuel reliability. The various poolside and hot cell examinations to determine the fuel failure root causes are provide in [Mahmood et al, 2014].

Reasons for fuel failure are as outlined below and described in Table 8-1:

- Poor design (including operation outside the experience base of the design) – this failure mode can be eliminated by a fuel design review. Effective Fuel Design Reviews are treated in the ANT International Fuel Design Review Handbook, [Strasser et al, 2010b].

- Poor manufacturing process – this failure mode can be eliminated by good fuel fabrication practice. See ANT International Fuel Fabrication Handbook to minimize fabrication related defects [Strasser et al, 2014].

- Changes of coolant chemistry and/or duty outside the regime in which the fuel has been qualified for – this can be eliminated by

    - Healthy fuel examinations (HFE) to get early warning of an emerging fuel performance issue

    - Following the trends in the industry of fuel issues in other reactors related to new designs, materials or fuel failures due to fabrication/design flaws. Also the knowledge of fuel failure mechanisms impacting the fuel failure process may help to eliminate the problems.

Table 8-1:   Fuel failure causes

Failure cause	Short description	Reason
Accelerated uniform nickel base corrosion, flow-assisted corrosion or IASCC	Stainless steel and nickel base alloy components in fuel assemblies may fail due to excessive stresses and/or poor heat treatment during manufacturing resulting in poor microstructure or inadequate surface treatment (Ni-based alloys)	Poor design or fabrication process
PWR Fuel Assembly bowing	In a fuel assembly the hold-down spring force should provide an adequate hold-down force together with gravity[1] of the fuel assembly to counteract the uplift forces due to hydraulics[2] as well as buoyancy[3]. The net force considering all four forces should be such that the fuel assembly does not lift-off from core plate[4]. However, if the net hold-down force is too large excessive fuel assembly bowing may occur. Significant fuel assembly bowing may lead to decreased thermal margins (LOCA, dnb) and difficulties to load the fuel assemblies without damaging the spacer grids as well as inserting the RCCA	Poor design
BWR Channel Bowing	Channel distortion and bulging may impede CR insertion while channel distortion may also decrease thermal margins (dry-out and LOCA)	Poor design (material choice), manufacturing process or core management
Excessive Zr alloy corrosion	An accelerated corrosion process results in cladding perforation. This corrosion acceleration can be generated by e.g., CRUD deposition (CILC)[a], Enhanced Spacer Shadow Corrosion, (ESSC)[b], (in BWRs), dry-out due to excessive fuel rod bowing.	Poor design or fabrication process or coolant impurities (in BWRs)
Localised hydriding	Fuel may fail in hydrided regions fractured under tensile loading that arise with accumulation of exposure during the course of normal operation (BWRs)[c].	Cladding composition, heat treatment or coolant impurities
Manufacturing defects	Non-through-wall cracks in the fuel cladding developed during the cladding manufacturing process. Defects in bottom and/or top end plug welds. Primary hydriding due to moisture in fuel pellets and or contamination of clad inner surface by moister or organics. Too large a gap between the fuel rod and the spacer grid supports (poor spacer grid manufacturing process) leading to excessive vibrations in PWR/VVER fuel causing fretting failures. Chipped pellets may result in PCI failures both in liner and non-liner fuel.	Poor manufacturing and quality assurance practices
PCI	PCI—an iodine assisted SCC phenomenon that may result in fuel failures during rapid power increases in a fuel rod.	Poor design or poor fabrication process
Cladding collapse	This failure mechanism occurred due to pellet densification. This failure mode has today been eliminated by fuel design changes and improved manufacturing control.	Poor design
Fretting	This failure mode has occurred due to: Debris fretting in BWR and PWR/VVER. Grid-rod fretting – Excessive vibrations in the PWR/VVER fuel rod causing fuel failures. This situation may occur for example due to different pressure drops in adjacent Fuel assemblies causing cross-flow. Baffle jetting failures in PWRs – Related to unexpectedly high coolant cross-flows close to baffle joints.	Poor design or fabrication process in combination with debris

1) Due to the mass of the FA.

2) The coolant flowing upwards will result in frictional forces at all FA surfaces that will result in an upward force.

3) Buoyancy is the upward force on the FA produced by the surrounding coolant due to the pressure difference of the coolant between the top and bottom of the FA. The net upward buoyancy force is equal to the magnitude of the weight of the coolant displaced by the FA combined with the force due to coolant flow. This force enables the FA to seem lighter than its actual weight.

4) If this happens, there is a potential risk that the FA will not be seated correctly afterwards on the bottom core plate, which would then impose difficulties to insert the control rods.

ANT International, 2015

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.

Fuel design review and fuel fabrication review are discussed in the following sections.

8.2　　　Fuel design review

The overall objective of the design review is to assure that the fuel assembly 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. 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.

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 such as those of of 10 CFR 50[35], 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.

---

[35] Title 10 of the U.S. Code of Federal Regulations Part 50

- 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. Guidance is provided regarding items that have the greatest influence on fuel performance and prioritize the audits by [Strasser et al, 2010b].

8.3 Fuel fabrication process review

The fabrication methods and parameters have a significant effect on the properties and microstructure of the Zr alloys, and these in turn on the performance of the Zr components in the reactor. The effect of the major fabrication steps on the properties, the effect of the properties on performance and the effect of the reactor environment on performance are shown in Figure 8-1. As an example, this figure can be interpreted as follows: The melting operation determines the Chemical Composition and Homogeneity of the alloy in the large ingot, properties that do not change in subsequent fabrication steps. The Chemical Composition affects every property of the alloy, while Homogeneity affects the Corrosion resistance and Ductility of the alloy. These in turn are affected by most of the environmental parameters during their exposure.

Neutron - Flux	Neutron - Fluence	Power History	Burn Up	Heat Transfer	Clad Temperature	Water Chemistry	Material Properties / Process Parameter	Chem. Composition	Homogeneity	Precipitates	Grain Size	Dislocation Density	Texture	Surface Condition
	■	■			■		Strength	■			■	■	■	
	■	■			■		Ductility/Fracture Toughness	■	■		■	■	■	■
■	■	■			■		Creep	■			■	■	■	
	■	■			■		Growth	■			■	■	■	
■		■	■	■	■	■	Corrosion	■	■	■		■	■	■
		■	■	■	■	■	Hydrogen Pickup Fraction	■		■				■
	■	■			■		Stress Corr. Cracking	■	■		■		■	■
							Melting	■	■					
							Hot Forming		■					
							Beta Quenching		■	■				
							Pilgering				■	■	■	
							Intermed. Annealing			■	■			
							Final Annealing				■	■	■	
							Surface Finish							■

ANT International, 2015

Figure 8-1: The relation between material properties, operation parameters, material characteristics and manufacturing processes.

9 Discussion and summary

The reliability of fuel in power-generating water reactors has improved and is now high relative to earlier experience. Fuel rod failure rates are slightly different among reactor types and countries. The latest tabulation by the IAEA shows, however, the average number of leaking fuel rods per discharged assembly is about 53 fpm for PWRs, 60 fpm for VVERs, 3 fpm for CANDU/PHWRs and 3 fpm for BWRs; [Inozemtsev & Onufriev, 2013]. These failure rates are for the 2004–2010 reporting interval and are similar to those of the previous 5 year interval.

Fuel rod reliability is affected by the performance of integral fuel assemblies as well as individual fuel rods. That is, the leading cause of failures in PWR fuel is grid-to-rod fretting, which is responsible for ~40% of the fuel failures. Debris fretting, which differs from GTRF, is observed in all types of fuel assemblies; e.g., ~13% of PWR failures, ~35% of BWR failures. Debris fretting is also observed in VVER and CANDU/PHWR fuel at unspecified rates. Although greatly reduced as a mechanism of recurrent failures, PCI/SCC accounts for ~3% of PWR failures and ~24% of BWR failures. Fuel fabrication issues also affect all types of fuel, with ~11% of PWR and 1% of BWR failures due to manufacturing issues. The remaining 30–40% of the fuel failures are due to unknown or indeterminate causes; i.e., the failed fuel has not been inspected or the cause cannot be determined from the inspections that have been performed.

Grid-to-rod fretting issues are being addressed by changes in the designs of lower nozzles and grids. Nozzle improvements focus primarily on reducing the turbulence and non-uniformity of coolant flows in the lower elevations of a fuel assembly. Grid improvements include larger bearing areas between the fuel cladding and the structural supports in the grids; i.e., springs and dimples. Some of the design changes also involve the use of relaxation-resistant, Ni-based spring materials. Although these changes are similar to those implemented in earlier designs, the likelihood of actually reducing the GTRF failure rates appears to be increasing because of ongoing improvements of capabilities for modeling fluid-structure interactions and the incorporation of modeling and test results into PWR fuel assembly designs.

Efforts to reduce the rate fuel failure due to debris fretting involve both the fuel suppliers and operating utilities. Debris exclusion practices have been established in manufacturing, transportation, storage and reactor facilities. Debris removal programs have been implemented in plants with debris in their primary coolant systems. Debris strainers have also been installed to reduce the amount of debris that is transported into the core by the primary coolant. More importantly, lower nozzles and tie plates have been developed to exclude or trap the small debris (metal turnings, chips, wires) observed to cause fretting of fuel cladding. Spacer grids are being redesigned to capture debris at elevations where flow-induced vibrations are low (entrance to a fuel assembly) and minimize the capture probability at elevations where vibrations are high (upper half of a fuel assembly).

Failures due to the PCI/SCC process have been observed in PWRs, but are more common in BWRs. This failure mechanism was essentially eliminated by the use of Zr-liner cladding and is now observed only at a very small rate; viz., rod failure rate of <1 fpm. The cause of these failures is unknown, but is likely to involve the duty cycles under which BWR fuel now operates and the effects of pellet chips and missing cylindrical surface on cladding stress. High-energy fuel and core designs require a large number of fuel rods and large amounts of fuel within each rod to operate closer to the power-exposure envelope. As a result, more fuel is exposed to the conditions needed for PCI/SCC of Zry-2 cladding. In addition, the presence of missing pellet surface or pellet chips in the pellet-cladding gap increases cladding stress and has been observed to cause PCI failures under operating conditions where none should have occurred. Although all fuel suppliers are working to eliminate missing surface and pellet chips, the brittle nature of sintered UO_2 hinders these efforts and increase the importance of quality control and audit programs.

10 References

Adamson R. B., *Irradiation Growth of Zircaloy*, Zirconium in the Nuclear Industry; Third Conference, ASTM STP 633, pp. 326, 1977.

Adamson R., *Effects of Neutron Irradiation on Microstructure and Properties of Zircaloy*, Zirconium in the Nuclear Industry; Twelfth International Symposium, Eds. Sabol, G., Moan, G., Toronto, Canada, American Society for Testing and Materials, vol ASTM STP 1354, pp 15-31, 2000.

Adamson R. B., Lutz D. R., and Davies J. H., *Hot Cell Observations of Shadow Corrosion Phenomena*, Proceedings Fachtagung der KTG-Fachgruppe, Brennelemente und Kernbautelle, 29 Februar/1 Marz 2000, Forschungszentrum Karlsruhe, 2000.

Adamson R. B. and Rudling P., *Dimensional Stability of Zirconium Alloys*, ZIRAT7/IZNA2 Special Topic Report, ANT International, Mölnlycke, Sweden, ANT International, 2002.

Adamson R., Cox B., Garzarolli F., Strasser A., Rudling P. and Wikmark G., *Corrosion of Zirconium Alloys*, ZIRAT7/IZNA2 Advanced Nuclear Technology International, Uppsala, Sweden, 2002.

Adamson R. and Cox B., *Impact of Irradiation on Material Performance*, ZIRAT10/IZNA5, Special Topic Report, Advanced Nuclear Technology International, Surahammar, Sweden, 2005.

Adamson R., Cox B., Davies J., Garzarolli F., Rudling P. and Vaidyanathan S., *Pellet-Cladding Interaction (PCI and PCMI)*, ZIRAT11/IZNA6 Special Topic Report, Advanced Nuclear Technology International, Skultuna, Sweden, 2006.

Adamson R., Garzarolli F., Cox B., Strasser A. and Rudling P., *Corrosion Mechanism in Zirconium Alloys*, ZIRAT12/IZNA7, Advanced Nuclear Technology International, Skultuna, Sweden, 2007a.

Adamson R., Garzarolli F. and Patterson C., *In-Reactor Creep of Zirconium Alloys*, ZIRAT14/IZNA9, Special Topic Report, Advanced Nuclear Technology International, Skultuna, Sweden, 2009.

Andersson T., Almberger J. and Björnkvist L., *A Decade of Assembly Bow Management at Ringhals*, International Meeting on LWR Fuel Performance, Orlando, Florida, USA, American Nuclear Society, 2004.

Andersson B., Bergmann U., Dahlbäck M., Hallstadius L., Haskins W. and Limbäck M., *Advantages Gained by Introducing Low Tin ZIRLO™ as BWR Channel Material*, TopFuel 2015, Reactor Fuel Performance, Zurich, Switzerland, Eoropean Nuclear Society, pp 430-437, 2015.

Andriambololona H., Bosselut D. and Massin P., *Methodology for a numerical simulation of an insertion or a drop of the rod cluster control assembly in a PWR*. Nuclear Engineering and Design, 237, pp 600-606, 2007.

Averin S. A., Panchenko V. L., Kozlov A. V., Sinelnikov L. P. Shishov V. N. and Nikulina A. V., *Evolution of dislocation and precipitate structure in Zr alloys under long term irradiation*, Zirconium in the Nuclear Industry: Twelfth International Symposium, ASTM STP 1354, G. P. Sabol and G. D. Moan, Eds., American Society for Testing and Material, pp. 105-121, West Conshohocken, PA, 2000.

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

Baranowski R., *Investigation on debris related fuel failures in BWR supported by CFD simulations*: Division of Nuclear Reactor Technology. Stockholm, Sweden, Royal Institute of Technology, Master of Science, 2014.

Beale J., *Fuel Reliability Update, FRED 4.0*, International Utility Fuel Performance Conference, St. Louis, Missouri, September, 2014.

Bieli R., Ledergerber G., Zbib A., Blavius D. and Muench C.-J., *Abnormal bow of Zircaloy-2 fuel channels without shadow corrosion influence*, Jahrestagung Kerntechnik 2011, 17-19 May, Berlin, Germany, p 870, 2011.

Blat-Yrieix M., Ambard A., Foct F., Miguel A., Beguin S. and Cayet N., *Toward a better understanding of dimensional changes in Zircaloy-4: What is the impact induced by hydrides and oxide layer?*, Journal of ASTM International, Vol. 5, No. 9, Paper ID JAI101321, 2008.

Bossis P., Thomazet J. and Lefebvre F., *Study of the Mechanisms Controlling the Oxide Growth Under Irradiation: Characterization of Irradiated Zircaloy-4 and Zr-1Nb-O Oxide Scales*, Zirconium in the Nuclear Industry: Thirteenth International Symposium: ASTM STP 1423, G. D. Moan and P. Rudling, Eds., ASTM International, pp. 190-221, West Conshohocken, PA, 2002.

Buechel R., Reparaz A., Bradfute J. and Norell J., *Fuel Reliability: How it affects the industry, and one fuel vendor's journey to flawless fuel performance*: Nuclear Power International. Tulsa, Oklahoma, USA, Penn Wall Corp., vol 7, pp 22-24, 2014.

CANDU 6 Program Team, *CANDU 6, Technical Summary*, Vol Reactor Development Business Unit, AECL, Mississauga, Ontario, Canada, 2005.

Cantonwine P., Lin Y.-P., Lutz D., White D., Ledford K., *BWR Corrosion Experience of NSF Channels*, TopFuel 2013: Light Water Reactor Fuel Performance Meeting, Charolotte, North Carolina, USA, American Nuclear Society, 2013.

Cantonwine P., Schneider R., Dunavant R., Ledford K. and Fawcett R., *GNF Fuel Performance 2015 Update*, TopFuel 2015: Light Water Reactor Fuel Performance Meeting, Zurich, Switzerland, European Nuclear Society, pp 295-304, 2015.

Capps N., Montgomery R., Sunderland D., Spencer B., Pytel M. and Wirth B., *Evaluation of Missing Pellet Surface Geometry on Cladding Stress Distribution and Magnitude*, U.S. Department of Energy, Consortium for Advanced Simulation of LWRs (CASL), 2015.

Causey A. R., Cheadle B. A., Coleman C. E. and Price E. G., *Improving the Service Life and Performance of CANDU Fuel Channels*, Presented at IAEA Tech Comm Meet on Advances in Heavy Water Reactors, Bombay, India, Jan. 29 - Feb 1, 1996, AECL 11543, 1996.

Chalmers B., *Physical Metallurgy*. New York, New York, USA, John Wiley & Sons, 1962.

Chen J., Lindberg F., Belova L., Forsgren B., Gott K., Lejon J. and Jasiulevicius, A., *High Resolution Electron Microscopy Study on Oxide Films Formed on Nickel-Base Alloys X-750, 182 and 82 in Simulated High Flow Velocity BWR Water Conditions*, 15th Symposium on the Environmental Degradation of Materials in Nuclear Power Systems, Virginia Beach, VA, August, 2009.

Chung H., Strain R. V., Bray T. and Billone M. C., *Progress in ANL/USNRC/EPRI Program on LOCA*, Proceeding of the Topical Meeting on LOCA Fuel Safety Critera, Aix-en-Provence, 2001, NEA/CSNI/R(2001)18, 2001.

Cole S., Garner N., Graebert R.-F. and Mollard P., *ATRIUM™ 11 – Validation of Performance and Value for BWR Operations*, TopFuel 2015, Reactor Fuel Performance, Zurich, Switzerland, European Nucler Society, pp 325-334, 2015.

EPRI, *Post Irradiation Evaluation of BWR Fuel From Hope Creek Reactor*, Electric Power Research Institute, Palo Alto, California, USA, 1996a.

EPRI, *Operational Recommendations for Failed Fuel - An EPRI Perspective*, Electric Power Research Institute, Palo Alto, California, 1996b.

EPRI, *CHIRON - A Fuel Failure Prediction Code Version 3.0*, Electric Power Research Institute, Palo Alto, California, USA, 1998.

EPRI, *BWR Channel Distortion Database*, Ed. E. Mader, Electric Power Research Institute, Palo Alto, California USA, 2012.

Etoh Y. and Shimada S., *Neutron Irradiation Effects on Intermetallic Precipitates in Zircaloy as a function of Fluence*, Journal of Nuclear Materials, Vol. 200, pp. 59-69, 1993.

Evans P. M., Conner M. E. and Lu R. Y., *Investigation of Vibration Related Fractures in PWR Grids*, Water Reactor Fuel Performance Meeting, Chengdu China, September, 2011.

Flanagan M., Oberländer B. and Puranen A., *Fuel Fragmentation, Relocation and Dispersal under LOCA Conditions: Experimental Observations*, Top Fuel 2013, Charlotte, North Carolina, USA, American Nuclear Society, pp 660-668, 2013.

Fleck R. G., Elder J. E., Causey A. R. and Holt R. A., *Variability of Irradiation Growth in Zr-2.5Nb Pressure Tubes* Zirconium in the Nuclear Industry: 10th Int'l Symposium, ASTM STP 1245, A. M. Garde and E. R. Bradley, Eds., American Society for Testing and Materials, Philadelphia, 168-182, 1994.

Franklin D., Lucas G., Bement A., *Creep of Zirconium Alloys in Nuclear Reactors*, Vol ASTM STP-815, American Society for Testing and Materials, Phidelphia, Pennsylvania, USA, 1983.

Fruzzetti K. and Perkins D., *PWR Chemistry: EPRI Perspective on Technical Issues and Industry Research*, VGB NPC '08 Water Chemistry Conference, Berlin, September 14-18, 2008.

Fujimoto K., Iwamura T., Suzuki S., Kobayashi T. and Kikuchi Y., *SCC Growth rate of cold worked austenitic stainless steel in PWR environment*, Proceedings of AECL/COG/EPRI Workshop: Effects of Cold Work on Stress Corrosion Cracking of Materials in Water Cooled Nuclear Plants, EPRI Report 1016519, Toronto, Canada, 2007.

Garde A., Smith G. and Pirek R., *In-PWR Iradiation Performance of Dilute Tin-Zirconium Advanced Alloys*, Zirconium in the Nuclear Industry: Thirteenth International Symposium, Eds. G. Moan and P. Rudling, Annecy, France, American Society for Testing and Materials, vol ASTM STP 1423, pp 490-506, 2002.

Garlick A., *Stress Corrosion Cracking of Zircaloy in Iodine Vapour*, Vol UK Report, TRG Memo 4336(W), 1968.

Garlick A., *A critical review of available information on fuel failures in Douglas Point and some AECL loop experiments*, Vol UK Report, TRG Memo 5002(W), 1969.

Garzarolli F., Manzel R., Peehs M., and Stehle H., *Observations and hypothesis on pellet-clad interaction failures*, Kerntechnik 20 Jahrestagung, vol No. 1, pp 27-31, 1978.

Garzarolli F., Von Jan R. and Stehle H., *The main causes of fuel element failure in water cooled power reactors*, IAEA Atomic Energy Review, Vol. 17, pp. 31-128, 1979.

Garzarolli F., Dewes P., Maussner G. and Basso H-H., *Effects of high neutron fluences on microstructure and growth of Zircaloy-4*, Zirconium in the Nuclear Industry: Eighth International Symposium, ASTM STP 1023, pp. 641-657, ASTM, 1989.

Unit conversion

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

MASS	
kg	lbs
0.454	**1**
1	2.20

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

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

STRESS INTENSITY FACTOR	
MPa√m	ksi√inch
0.91	**1**
1	1.10

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