Fuel Reliability Assessment Through Radiochemistry and Poolside Examinations

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Contents

1 Introduction

Nuclear power plants are capable of operating safely with leaking fuel in their cores and have done so on many occasions. Fuel reliability is increasing so that fewer fuel rods develop leaks that release fission products to their primary coolant systems (defined as failure in this report). In addition, increasing numbers of nuclear power plants (NPPs) are now operating without leaking fuel. So, what is the reason for a report on assessing in-core fuel reliability by means of radiochemistry?

Fuel failures affect three principal areas: radiological safety, core and plant operation, and costs. With increased fuel reliability (reviewed in Section 2.1), the effects of leaking fuel on each these areas are heightened. Reactors are now operating with lower radiation exposure fields due to changes such as:

- The addition of zinc to the primary coolant to entrain gamma-emitting corrosion products in the surface scale (crud)on fuel rods,
- The elimination or minimization of fuel and control assembly materials that produce corrosion products that emit high-energy gamma radiation,
- Reduction in the frequency and number of failed fuel rods and by
- The progressive elimination of fuel particles that were dispersed into the primary systems by fuel failures during previous cycles of operating cycles.

The first two items were driven primarily by the desire to reduce radiation exposure to values that are as low as reasonably achievable (ALARA), though zinc injections also help reduce stress corrosion cracking of primary system components. The second two items also address ALARA concerns. That is, the fission products released from leaking fuel rods increase radiation exposure while the failed fuel is operating. Fuel material that is released from leaking rods, generally following post-failure or "secondary" degradation, has the potential for increasing radiation exposure levels for 10-12 years after the degraded fuel is discharged. While it is now almost unheard of for an NPP to reduce power or shut down because of exposure to site personnel or the surrounding public, increases in radiation exposure due to leaking fuel are more clearly evident and are subject to proactive management to prevent the resulting activity levels from compromising plant operation or adding significantly to operating costs.

The issue of secondary degradation is particularly severe because of the large increases in radiation exposure that such events can cause in the primary coolant and clean-up systems. As discussed in Section 2.1.2, secondary degradation involves a combination of factors, one of which is the physical loading imposed on fuel cladding by power changes in leaking fuel rods. The knowledge that fuel has failed during an operating cycle and, when possible, the location of the leaking fuel are important considerations in how an affected core is operated. In BWRs, where the risk of secondary degradation is typically greater than in PWRs or VVERs, material improvements combined with the insertion of control blades in and sometimes around cells containing leaking rods has enabled operation without secondary degradation during the 18-24 month operating cycles used by most U.S. plants; [Schneider et al., 2016]. On-line detection and assessment of leaking fuel using radiochemistry are important factors in managing failed fuel rods to avoid forced outages than can come from secondary degradation.

The use of radiochemistry for the detection and tracking of leaking fuel will obviously not prevent failures from occurring, but can provide information for mitigating the radiation exposure, operational and economic consequences noted above.

The cost of operating with leaking fuel is significant for both the affected operating plant and the fuel vendor. Costs vary with the nature of the fuel failure and are generally greatest when secondary degradation takes place. Some of the significant costs associated with operating an NPP with leaking fuel have been estimated by Buechel et al., [2014] and are listed in Table 1-1. 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. The combined costs to the operating utility and fuel supplier range from \$1.4M to \$16.2M (1.2M to 13.1M €).

Table 1-1: Estimated cost of leaking core outage, modified according to [Buechel et al., 2014].

Cost category	PWR or BWR	Estimated cost ($K)			
		Utility		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

The use of radiochemistry for the detection and tracking of leaking fuel will obviously not prevent failures from occurring, but can provide information for mitigating the radiation exposure, operational and economic consequences noted above. Unfortunately, the resulting information is frequently confounded by the presence of more than one leaking rod, by the presence of fissile material in the primary coolant system from leaking fuel rods in prior reactor cycles, during the current cycle or a combination of all of these conditions. As a result, the application of radiochemical data in assessments of fuel reliability has historically involved empirical observations supported by varying degrees of scientific rigor.

With the operational and financial consequences of fuel failure and secondary degradation, the ongoing trend has been to supplement empirical indices, such as changes activity levels and changes in the ratios of radionuclide activities, with methods based more on the physical processes through which fission products and fissile materials are released to the primary coolant system. These phenomenological methods relate activity measurements to terms such as release or escape coefficients. Although intended to better represent the release of radionuclides, the model terms effectively become new empiricisms due to the simplifying assumptions needed in their construction. However, the combination of empirical indices and phenomenological methods is useful in both detecting the occurrence and assessing condition of leaking fuel during reactor operation, particularly when combined with some of the on-line monitoring methods that are now being used.

The objective of this report is to identify methods for detecting failed fuel and assessing conditions associated with leaking rod(s) during operation in an NPP. The report provides background for understanding the capabilities and limitations of the tracking and assessment methods. The causes and characteristics of fuel failures and secondary degradation in modern NPPs are reviewed briefly in Section 2. Sections 3, 4 and 5 review the methods used for detecting and assessing fuel failures during and after irradiation. Fuel reliability improvements are discussed in Section 6. Appendix A -provides more details of primary failure causes and degradation mechanisms.

2 Fuel failure causes and characteristics

Fuel reliability in this report refers to the maintenance of fuel rod hermeticity during and after in-reactor operation. That is, reliability is defined as the absence of a breach of the cladding wall, end plugs or end plug welds that releases fission products or fuel material from within an irradiated fuel rod. The broader sense, reliability also involves other measurable design and safety factors such as strain, corrosion and dimensional stability and other calculated criteria such as fuel temperature, margins to heat transfer limits and transient response to postulated (design basis) accidents. However, the focus in this report is on fuel rod breaches that release radionuclides from an affected fuel rod. The review that follows id divided into discussions of the causes and characteristics of primary failures and of post-failure (secondary) degradation.

2.1 Primary failure

2.1.1 Observed fuel failure mechanisms and rates- needs to be updated

The causes of fuel failures in water-cooled reactors vary slightly among reactor types but, with the exception of plant or reload-related upsets, tend to be similar for a type of NPP within a country due to the commonality of fuel designs, materials and operating conditions. The mechanisms by which fuel rods have failed in the past and are continuing to fail are summarized in Table 2-1 for PWRs and Table 2-2 for BWRs. These tables are based on 5-year intervals because the number of fuel failures has become small and leads to large fluctuations in the calculated rates when considered on an annual basis. The information available on fuel failures in VVER and PHWR/CANDU reactors does not permit the construction of similar tabular summaries, but is reviewed below.

Note that the mechanism percentages in Table 2-1 and Table 2-2 are based on the failures for which the root cause has been determined. The sum of the failure proportions for the known mechanisms is 100% in each time interval. The proportions in the "Unknown/undetermined" category are based on the total number of failures in each time interval, which causes the total for each time interval to exceed 100%; i.e., 100% for the known causes plus the percentage of unknown causes. This treatment differs from typical failure tabulations that are based on the total number of failures, but provides estimates of the failure fractions due to each cause that are unbiased by the unknown category. Normalizing the known failure proportions for each mechanism to the column total will give the customary proportions relative to the total number of failures in the respective time interval; e.g., normalizing the 57.9% fraction of observed PWR failures due to grid-to-rod fretting in the 2011-2015 interval to the column total, 140.2%, gives 41.3% due to this cause relative to the total number of failures.

Table 2-1: Summary of world-wide PWR fuel rod failure mechanisms for the interval 1987-2015, modified according to [IAEA, 2019].

Failure mechanism	Proportion of fuel failures for each mechanism (%)						
	1987-1990	1991-1994	1995-1998	1999-2002	2003-2006	2007-2010	2011-2015
Grid-to-rod fretting	16.6	42.7	73.0	87.6	78.0	58.4	57.9
Debris fretting	55.6	46.7	14.5	7.1	13.9	19.5	33.7
Fabrication issue	20.8	6.7	9.5	3.4	7.2	16.4	8.4
Crud or corrosion	0.0	0.0	2.2	1.5	0.0	0.0	0.0
PCI/SCC	0.0	0.0	0.0	0.0	0.9	5.7	0.0
Handling	2.8	3.8	0.8	0.4	0.0	0.0	0.0
Baffle jetting	4.2	0.0	0.0	0.0	0.0	0.0	0.0
Unknown/undetermined	50.0	48.0	26.7	14.6	33.2	28.4	40.2

Table 2-2: Summary of world-wide BWR fuel rod failure mechanisms for the interval 1987-2015; [IAEA, 2019].

Failure mechanism	Proportion of fuel failures for each mechanism (%)						
	1987-1990	1991-1994	1995-1998	1999-2002	2003-2006	2007-2010	2011-2015
Debris fretting	17.5	50.5	39.6	53.4	32.2	58.6	66
Crud or corrosion	42.3	4.4	46.8	23.1	52.9	3.9	13.2
PCI/SCC	27.7	34.1	9.9	11.5	14.2	36.2	18.9
Fabrication issue	10.1	11	3.7	11.5	0.7	1.3	1.9
Handling	2.4	0	0	0	0	0	0
Unknown/undetermined	1.62	32.5	24	27.8	13.6	29.2	26.4

For PWR fuel rods, failures during the past 10 years have been the result of grid-to-rod fretting, debris-induced fretting, manufacturing defects and a limited number of pellet-cladding interaction (PCI) events. The leading causes of PWR fuel failures continue to be grid-to-rod fretting and debris fretting. Failures due to crud, corrosion, baffle-jet effects and handling have not been observed. The fraction of unknown causes is in the range of 30-40% during this interval.

For BWR fuel rods, failures during the past 10 years have resulted from debris fretting, crud, corrosion, PCI and manufacturing defects. The leading causes of BWR fuel failures are debris fretting, PCI, and crud or corrosion. The fraction of failures due to unknown causes is in the range of 10-30% for the past 10 years.

The information on fuel failures in VVER fuel is sparse. From tabulations by the IAEA, the leading, recurrent cause of failure is debris fretting; [IAEA, 2010], [Inozemtsev &Onufriev,2013] and [IAEA, 2019].Instances of failure due to grid-to-rod fretting, crud or corrosion and manufacturing defects, including primary hydriding are also reported.

Failure data on PHWR/CANDU fuel indicate the leading mechanisms are debris fretting and fabrication defects (primarily weld issues). Isolated failures are reported to have been caused by pellet chips (PCI due to chipped pellets), micro cracks in fuel cladding and handling damage.

The fuel rod failure rates are shown by year and country for each reactor type in Figure 2-1 through Figure 2-4. These failure rates are based on the number of fuel assemblies found to be leaking after discharge by sipping and/or post-irradiation inspection combined with the average number of leaking rods observed historically in each type of fuel relative to total number of discharged assemblies and the number of fuel rods in each assembly. The numbers of failed rods in each leaking fuel assembly assumed in the IAEA, [2019] assessment are 1.6 for PWRs and VVER-1000 reactors and 1.1 for VVER-440s, CANDUs and BWRs.

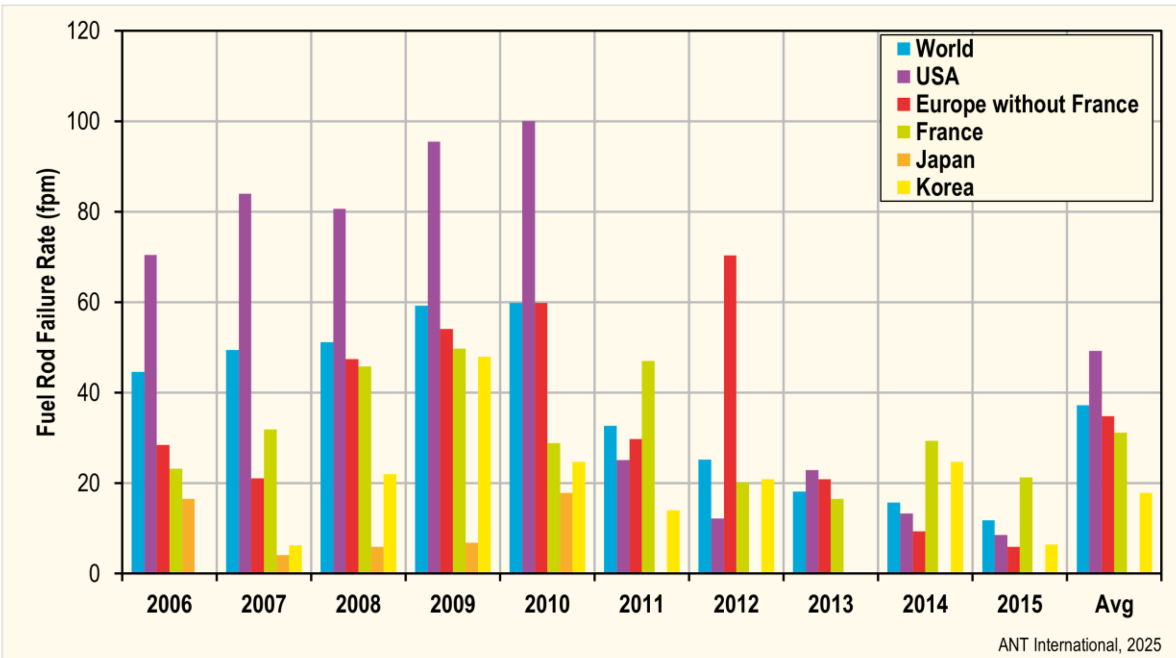


Figure 2-1: PWR fuel rod failure rates relative to year of fuel discharge and country, modified according to [IAEA, 2019].

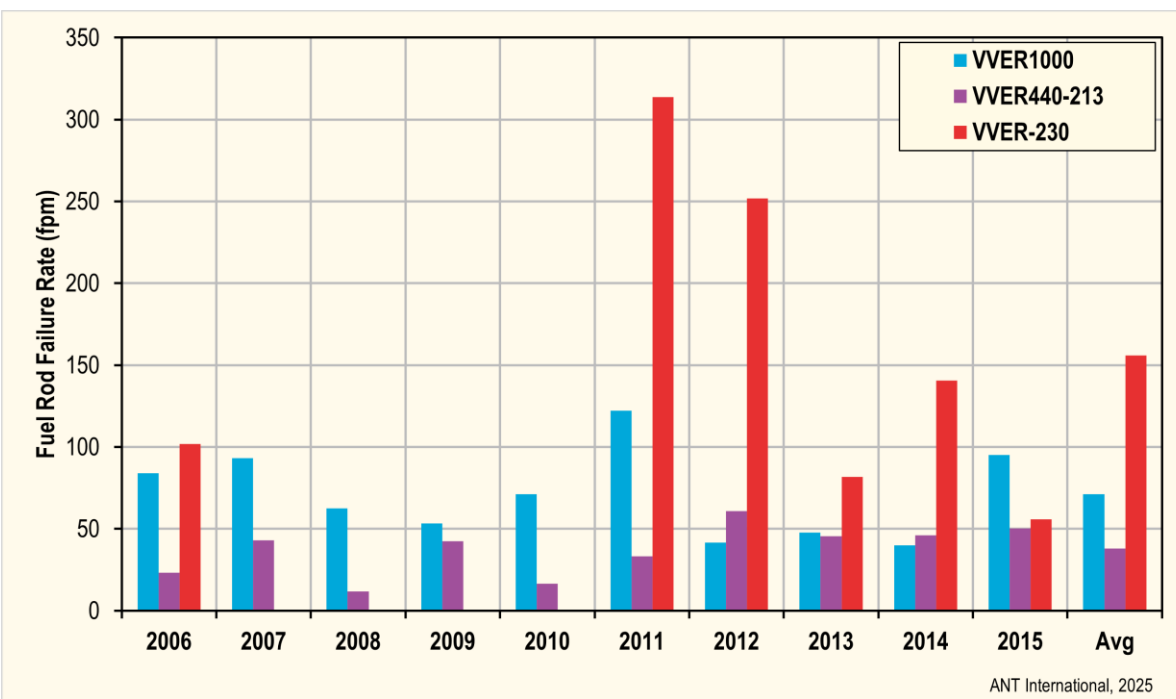


Figure 2-2: VVER fuel rod failure rates relative to year of fuel discharge for all countries, modified according to [IAEA, 2019].

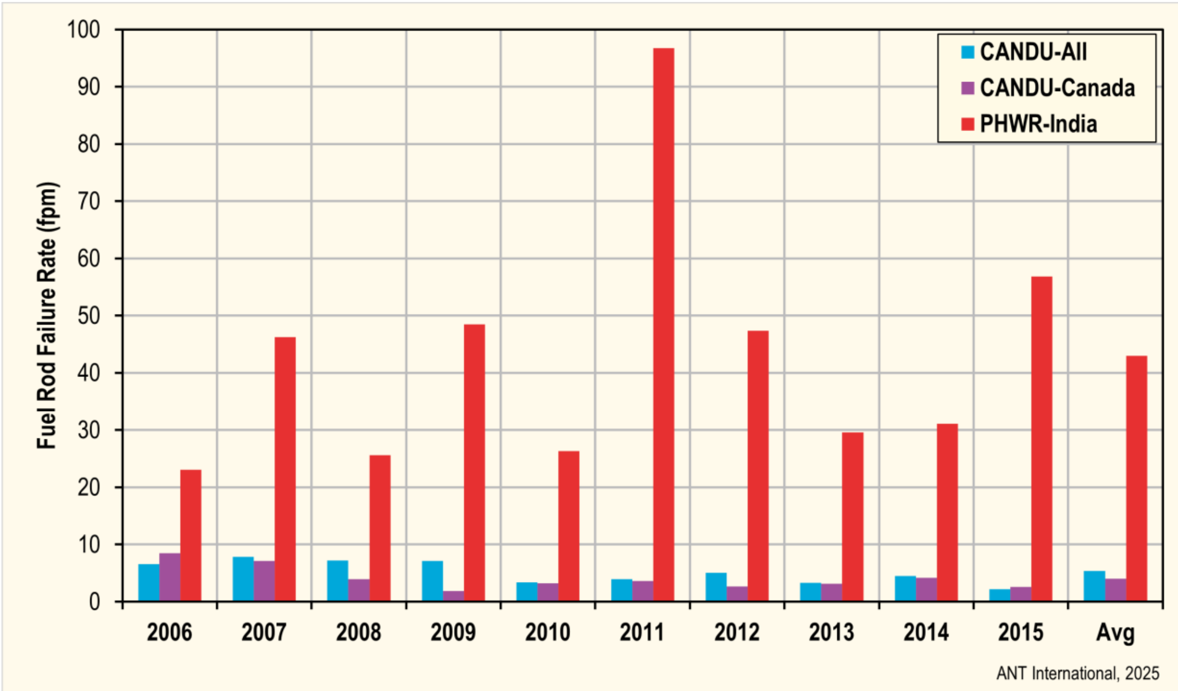


Figure 2-3: CANDU and PHWR Fuel rod failure rates relative to the year of fuel discharge, modified according to [IAEA, 2019].

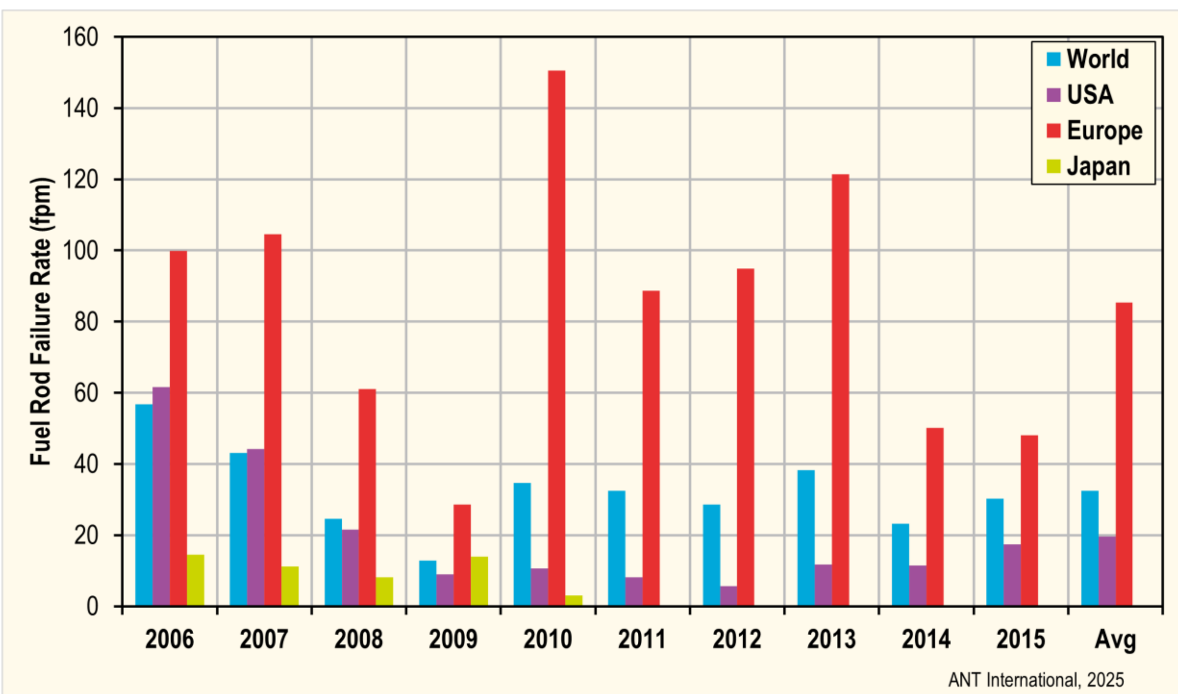


Figure 2-4: BWR fuel rod failure rates by year of fuel discharge and country, modified according to [IAEA, 2019].

The failure rates have decreased with time. Over the past 5 years, CANDU fuel continues to exhibit the lowest rates with 2-4 failures per million (fpm) discharged fuel rod. During this interval, PWR and BWR rods are failing at a rates of 10-100 fpm and 10-150 fpm, respectively, with differences among countries.

For more information about fuel failures causes and degradation mechanisms see Appendix A -.

2.1.2 Characteristics of fuel rod failure mechanisms

As indicated in the preceding section, recent fuel rod failures have been caused by grid-to-rod fretting, debris fretting, fabrication defects and sporadic instances of crud or corrosion and PCI. The initial leakage paths resulting from these failure mechanisms provide varying degrees of resistance to the release of radionuclides from the affected rod to the reactor coolant system. These leakage paths can change (almost always increase in effective size) during post-failure operation, particularly due to the potential effects of secondary degradation. However, the characteristics of these leakage paths and their evolution during operation enable the use of radiochemical analyses for the detection of fuel failure(s) and the estimation of conditions such as the number of leaking fuel rods, their approximate power and burnup, and the state of the leakage path. The characteristics of fuel failures are discussed below as background for the review of radiochemical analyses.

2.1.2.1 Fretting

Fretting is a wear process which involves the combined effects of abrasion (mechanically induced erosion) and corrosion at the metal-oxide surface. It can affect Zr-alloys in an aqueous environment when subjected to sliding contact due to flow induced vibrations. Corrosion affects the process because, the rate of oxide growth is very rapid initially and then slows with increasing oxide thickness. When oxide is removed by sliding contact with another metal surface, the oxide growth rate reverts to the initial rapid rate. As a result, the repeated process of oxide removal followed by re-oxidation can breach the cladding wall of a fuel rod under conditions where the sliding contact alone would produce little or no metal loss.

The same fundamental mechanism is involved in grid-to-rod fretting (GTRF) and debris-induced fretting. The scar produced by fretting depends on the shape of the contacting material and the direction or directions of relative motion. An example of GTRF is shown in Figure 2-5 in which the metal loss corresponds to sliding between the fuel cladding and the grid spring and supporting projections (dimples). Note that the wear in this example was produced in an ex-reactor test rig which induced vibrations normal to an assembly consisting of Zry-4 grids and Zry-4 clad fuel rod. The example illustrates that fretting can result from contact between similar metals. Examples of GTRF between dissimilar metals (Zr-alloy cladding and stainless steel grids) during reactor operation are shown in Figure 2-6.

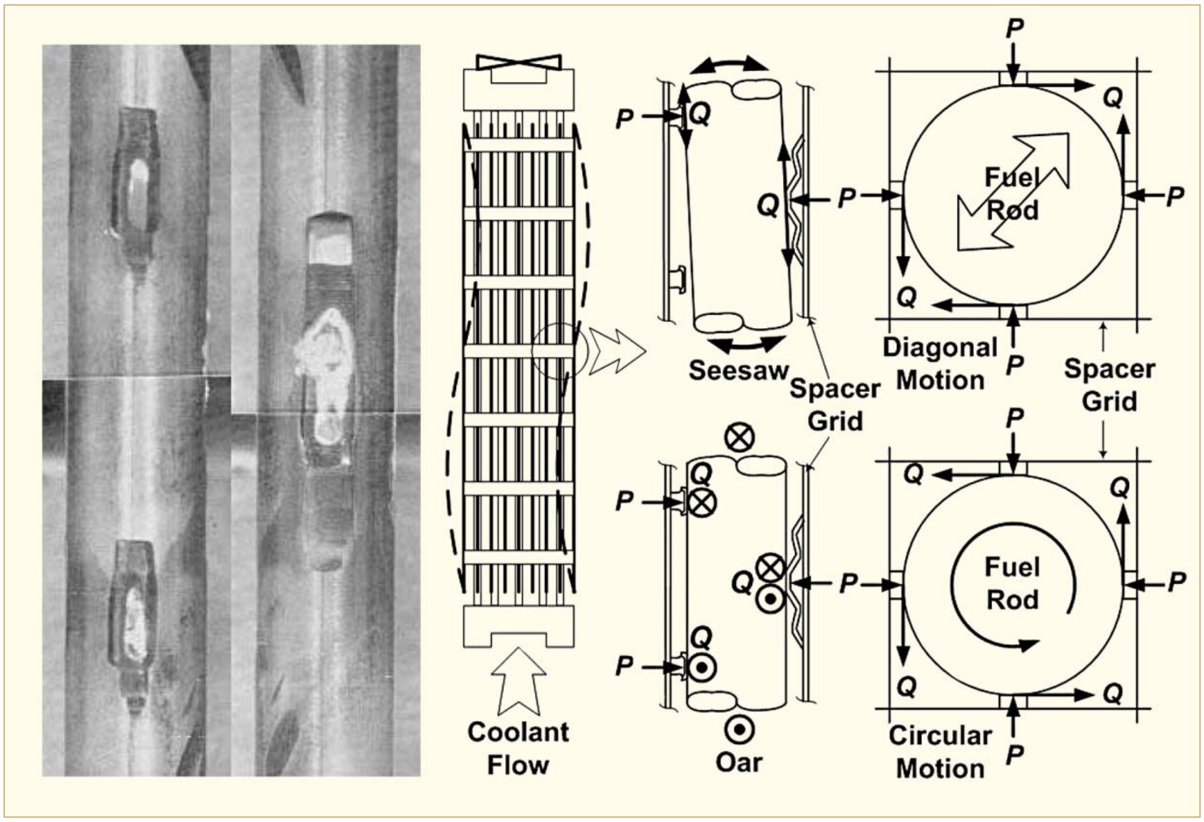


Figure 2-5: Fretting of fuel rods with schematic diagrams of the grid-to-rod interaction that contributed to the wear, modified according to [Kim &Lee, 2007].

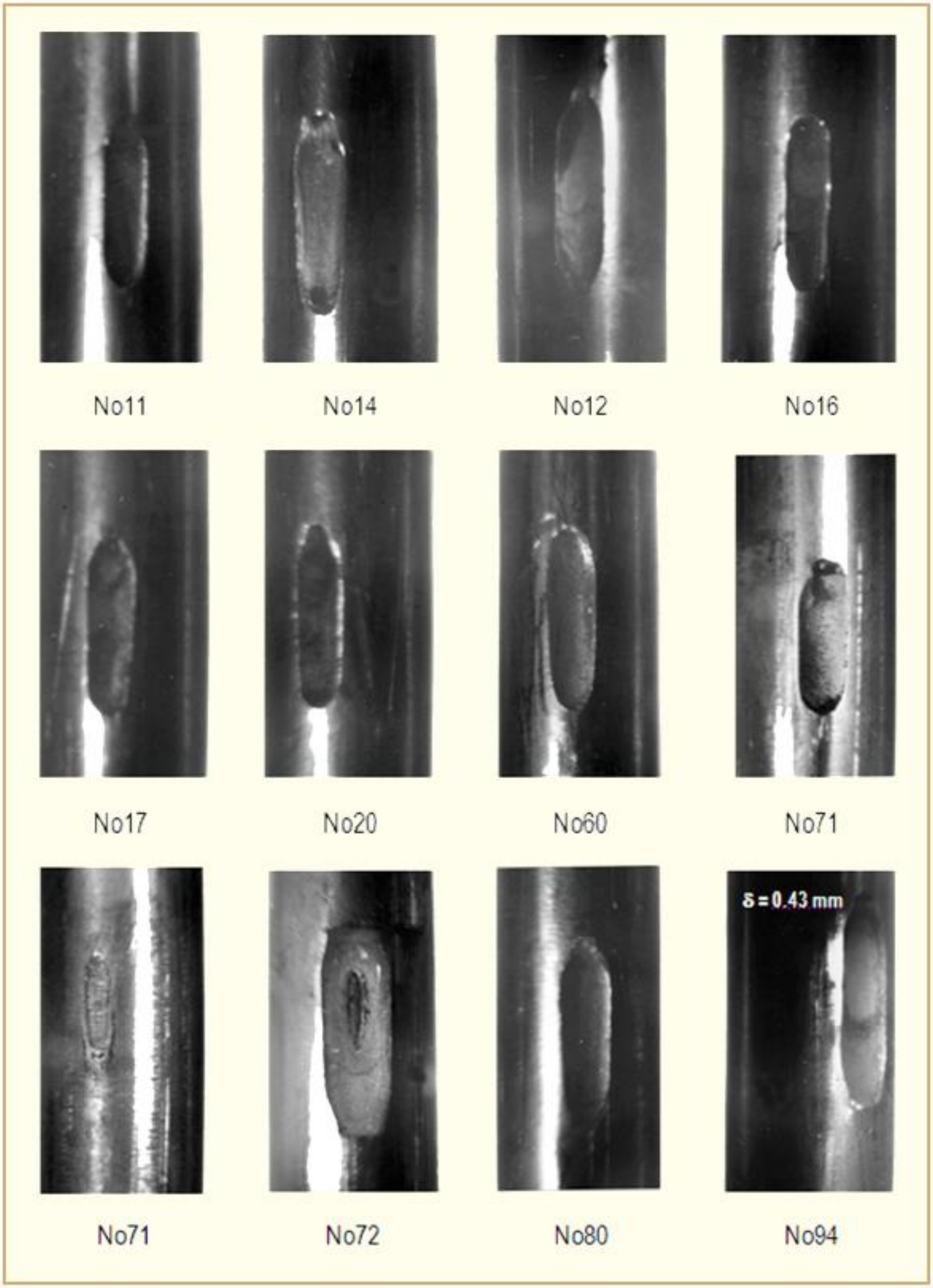


Figure 2-6: Visual appearance of cladding wear scars due to grid-rod fretting at the places of contact with 1st and 2nd spacer grids in VVER 440 fuel, modified according to [Smirnov et al., 2002].

The wear scars due to debris fretting differ in appearance from those due to GTRF and tend to reflect the nature of the debris. As an example, the wear and cladding perforations shown in Figure 2-7 correspond to pieces of wire or metal turnings that lodged in the lower nozzle and in spacer grids adjacent to a fuel rod. Cladding perforation resulted from flow-induced vibration of the trapped debris by way of cyclic abrasion-corrosion of the affected cladding surface.

Fretting scars like those shown in Figure 2-7 tend to be observed after modifications or maintenance work on the primary coolant system that involves machining operations. With increased awareness of the effects of debris on fuel reliability, the amount of such contamination has been reduced. Similarly, the size of debris has also been reduced. The reduction in debris size has shifted the fretting to higher elevations in affected fuel assemblies, generally produced smaller (harder to locate) debris scars, Figure 2-8 and Figure 2-9, and frequently involved the loss of debris from the affected assembly prior to poolside inspection; i.e., debris fretting is inferred by the wear scar and not the presence of debris.

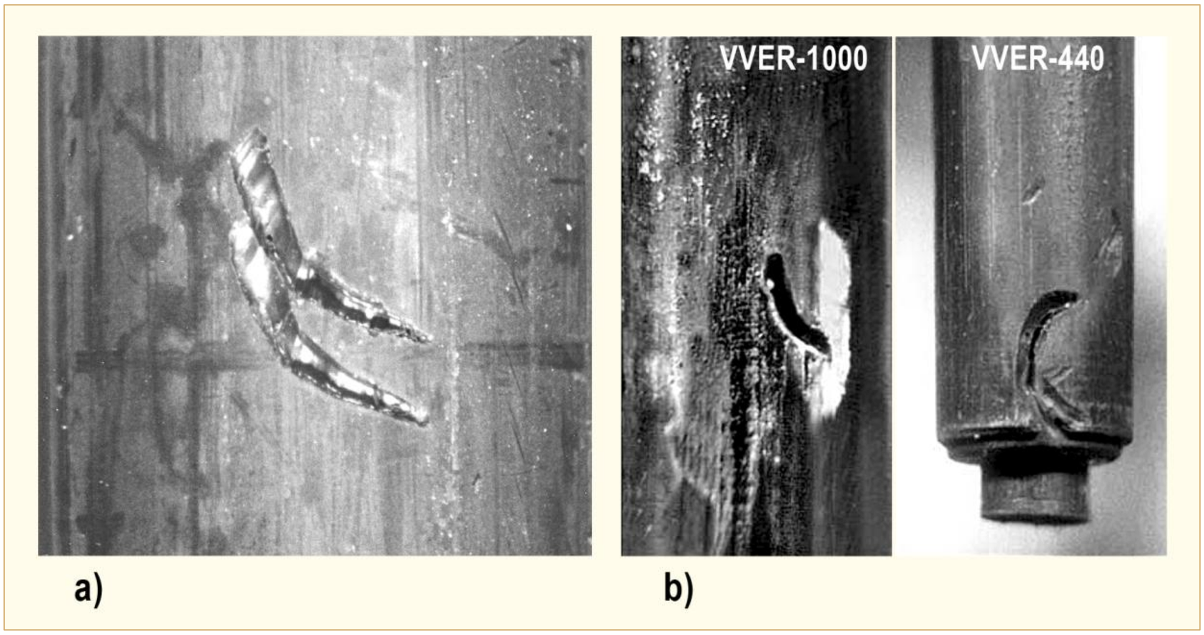


Figure 2-7: Examples of fuel failures due to debris fretting, a) modified according to [Lysell et al., 2008], b) modified according to [Smirnov et al., 2005].

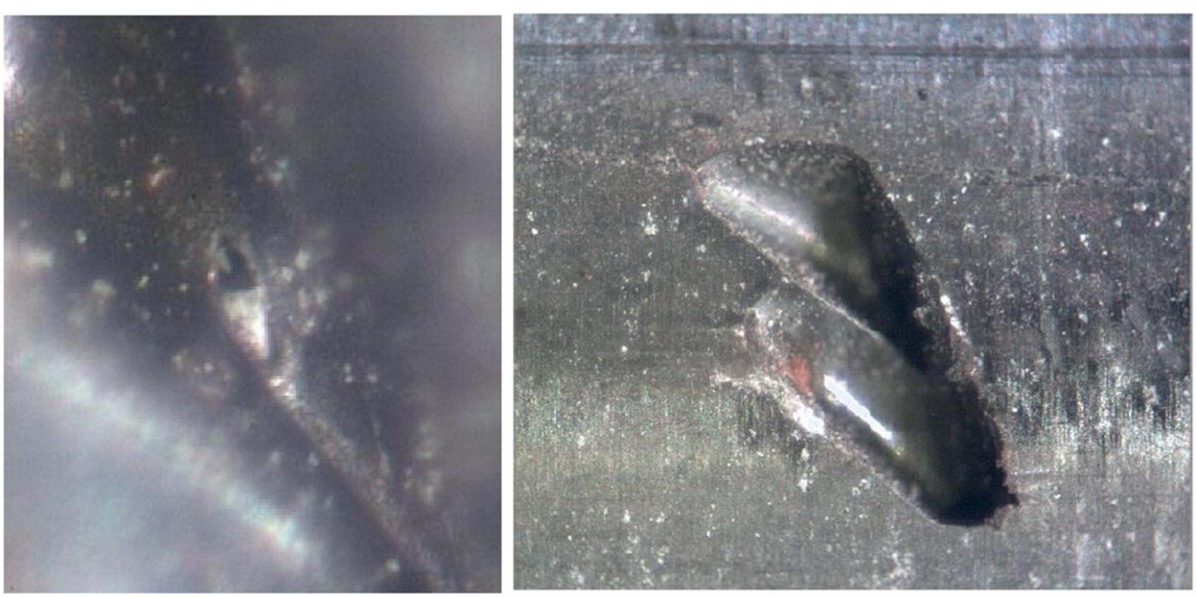


Figure 2-8: Example of a small fretting scar at the first (lowest) spacer location of a BWR fuel rod from the Ringhals-2 BWR, modified according to [Lysell et al., 2008].

3 Online assessment of fuel reliability using radiochemistry

3.1 Overview of section

On-line monitoring of fuel reliability is performed at different levels of detail. As occurrence of a failure during a cycle corresponds to a sudden presence (or an increase) of fission product concentrations in the primary circuit, measurement and assessment of primary 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 isotopes in the primary coolant and offgas activity for BWRs, with corrections for the effects of core power, coolant cleanup rate and background activity (e.g. distributed sources coming either from previous failures, or from contamination by tramp uranium still present on primary circuit walls or rod cladding; discussed below). Such industry indices are designed to indicate the presence of leaking fuel in operating reactors, for comparisons with license limits or internal specifications, 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 average linear heat generation rate, burn-up, the condition of such fuel and its expected evolution during operation. The focus of this section is on the later class of monitoring methods.

3.2 Effects of failure on in-core behaviour

3.2.1 Primary failure

Primary failure results from the formation of one or more leakage paths through the cladding, end plugs or welds of an affected fuel rod. A brief review of the types of primary failures relevant to modern water reactor fuel is given in Section 2.1. The initial leakage paths due to such failures are typically small enough that only the fission products that are in the gaseous state or that are soluble in reactor coolant are released to the primary coolant system; fuel material is typically retained in leaking rods. As a result, activity release rates generally increase at the time of failure, as the stored inventory of long-lived gaseous and soluble fission products (mainly Xe, Kr, I, Cs) are released, and then continue at rates that depend on the local power in the leaking rod and of the effective size of the leakage path. In the absence of factors that increase the size of leakage paths, like cladding corrosion or local hydriding, the release rates from primary failures vary with the power of the leaking fuel rod(s) in many cases[4] and can be limited by reducing the local power in the leaking fuel, preferably at or near the location of the leak. Such action is generally limited to BWRs as it requires local variations in power by means of control blade movements with activity measurements either before and after each blade move or continuously during the moves to identify the leaking fuel. Although the magnitude of activity release rates, or the total number of leaking rods (when a generic cause affects a lot of rods) can pose issues with respect to exposure, radiation limits for workers and safety authorizations, the significant characteristic of primary failures is that the activity release returns to pre-failure values when the leaking fuel is removed from the core.

3.2.2 Secondary degradation

Secondary degradation, reviewed in Section 2.1.2, involves changes to a fuel rod subsequent to its initial failure that enlarge the effective size of the leakage path or produce a second (and often larger) path and thereby increase activity release rates. Although the fission products released directly from a rod can be

[4] Small cladding breaches have been observed to be sufficiently restrictive that only long-lived fission gas is released to the primary system. In such cases, the amount of release and its composition are essentially independent of power. Failed fuel rods with this behaviour have been identified as “weak leakers” and are discussed later in this section.

large enough to pose activity issues, the most significant factor associated with secondary degradation is the dispersal of fuel into the primary system.

As noted earlier, fuel particles that are released from a damaged rod and transported by the coolant, add to the activity of the primary coolant system while the source rod is in-core. When fuel particles are released to the primary coolant and circulate through the core at each coolant rotation, a supplementary release of fission products occurs (mainly very short half-life isotopes). Moreover, the fuel particles from the failed rod, fuel washout, is quickly deposited on tubes and walls in contact with the coolant. Part is deposited under neutron flux, mainly on the cladding of neighbouring rods. Part also deposits on the steam generator tubes or PWRs and VVERs (they offer the largest contact surface) or on the steam separators of BWRs. So, the deposited fuel particles subjected to a neutron flux can continue to release fission products after the leaking / damaged rod is removed from the core. Large increases in the activity release rates during and after secondary degradation can adversely affect plant operation and almost always add to the cost of running a nuclear plant as noted in Section 1.

Although practices have changed over time, emphasis on operating nuclear plants well within their allowable limits have caused utilities to assess the risk of degradation and sometimes undertake mid-cycle outages to remove leaking fuel if large increases in activity could occur. Such outages are more frequent in BWRs because of their open primary systems and because the oxidizing nature of BWR coolant is conducive to the post-failure corrosion and hydriding that leads to the degradation. By comparison, both the likelihood of secondary degradation and its consequences are smaller in PWRs because of the closed primary system and the reducing nature of the coolant. Adverse effects arise in PWRs, but tend not to be as severe as in BWRs and are not generally severe enough to force mid-cycle outages.

An important aspect to successful NPP operation is detecting fuel failures as soon as they occur, assessing the condition of the leaking fuel and estimate the activity release evolution. It is specifically important to minimise fuel washout since the resulting tramp uranium will increase the background radiation level in the core for up to 10 years. These topics are reviewed in the sections that follow.

3.3 On-line monitoring

3.3.1 Summary of physical processes activated in a leaking fuel rod

The composition and concentration levels of fission products in the primary coolant and offgas streams changes with the occurrence of a leak in a fuel rod and varies with both the condition of the leak and its operating conditions. Failure by any of the mechanisms discussed in Section 2 produces a leakage path between the interior of the affected rod and the primary coolant system. At the time of perforation, coolant enters into the rod if the internal gas pressure, due to manufacturing filling with helium and the fission gases released out of the fuel material, is under the primary coolant pressure. Moreover, in some countries, a safety specification on the rod design imposes that its inner gas pressure shall be systematically under the primary pressure, even at the end-of-life for the most powerful rod. In these cases, the pressure discrepancy provokes ingress of primary water in the inner free volume of the rod (upper or lower plenum, fuel-clad gap). In other countries, the pressure of gas in the rod intraspace is allowed to exceed the coolant pressure and gas can be released to the coolant at the time of failure. Depending on the local temperature at the defect level during rod operation, different situations can occur with intraspace pressures less than the coolant pressure:

- If the local temperature is lower than the saturation temperature at the primary pressure, the entered water remains in a liquid phase and fills progressively the accessible volume. This is the case for failures occurring at rod extremities (e.g. plug welding, failure due to a debris blocked under the bottom grid…) which operate at lower linear power (or without power). The water quantity can be large, especially if it fills a plenum.
- If the local temperature on every surface is higher than the saturation temperature at the primary pressure, the entered water is transformed into steam. This is the general case along the fuel stack length (except a few tens centimeters at upper and lower extremities of PWR fuel

or the lower extremities of BWR fuel). Then steam propagates axially through the free volumes of the rod and can condense at rod extremities, for the same reason as above.

- When the radial temperature gradient authorizes at the same elevation both a liquid phase (e.g. on the inner surface of the clad) and steam (e.g. on the fuel outer surface), situation is more complex because a vaporization-condensation mechanism can be activated and maintained, as long as the water still enters. This mechanism, called "two phase regime", can be agitated and can provoke successive sudden ejections of vapor out of the rod. It concerns a PWR rod with a moderate local linear power (e.g. at high burn-up) and is often the common case when the reactor is operated at partial power.

After a while depending on the leak size, pressure equilibrium is reached between the inner free volume of the rod and the primary coolant pressure. This can last several days (and even few weeks) if the leakage path is small or if the fuel-clad gap is closed (as for high burnup rods). Due to the fuel cladding creep down and fuel rod swelling the pellet-cladding gap decreases with burnup and at about 20MWd/kgU there is fuel – cladding contact. This equilibrium can be quickly reached for end-of-life failures, i.e. when the inner pressure is already near the external pressure. After this step, gas or liquid starts to escape into the primary system because the rod inner pressure becomes higher than the external pressure. At constant power level and for a primary failure, the fission product release rate is continuous and presents an approximately constant level. It is due mainly to three main processes:

- steam continues to be formed as long as liquid is present,
- water and steam are radiolyzed by neutrons, gammas and recoil fission products, and finally
- fission gases continue to escape from the fuel materials and contribute also to the pressure increase.

The generation and release of fission products from the fuel pellets proceeds somewhat as described in Section 4 except that temperatures, oxygen potentials, physical states and chemical forms change due to the entry of water into the rod inner free volume.

The escaping gases include a mixture of xenon and krypton, with a routing gas which is probably hydrogen. Fission gases have an isotopic composition that depends on their source, rate of production and release, half-lives, decay chain and time. If the leakage path is large, or if the local linear power is moderate (i.e. water can remain in a liquid phase), coolant can enter into the rod intraspace and dissolve soluble fission products. In this case, gases and soluble fission products are released to the coolant by a combination of diffusion and mass transfer. This progression is shown schematically in Figure 3-1.As the size of the leakage path increases, the rate at which coolant flows into and out of a leaking rod also increases. Similarly, the time needed for radionuclides to escape to the coolant decreases and the concentrations of soluble nuclides such as iodine and cesium in the primary coolant increase. The ratio of short half-life isotopes in the total escaped inventory increases also. Concurrent with the release of radionuclides from the rod intraspace to the primary system, additional fission products are being generated in the fuel and released to the rod intraspace.

Except for the "two phase regime" presented above, which can present high / very high release rates at constant power, the release rate of fission products is moderate at constant power for the most of the cases, because there is no pressure difference upstream and downstream the defected rod. However, in the long term, the water present in the rod should disappear if there are no power changes: radiolysis is probably the most effective mean for that and formed hydrogen provides the inner pressure increase for fluid release. Oxygen is then consumed, but at a lower rate, primarily by forming an oxide on the cladding surface and to a lesser extent, with the fuel (formation of oxide phases). Finally, without water ingress, the rod void should become dry.

However, under transient conditions that involve changes in power or coolant pressure, previous pressure equilibrium is completely changed. Fission products in the rod void can be released in a spike-like manner, by vapor ejection. Such release, frequently provide the first indication of failure in reactors that rely primarily on the activity of soluble radionuclides such as iodine isotopes in the primary coolant to indicate failure.

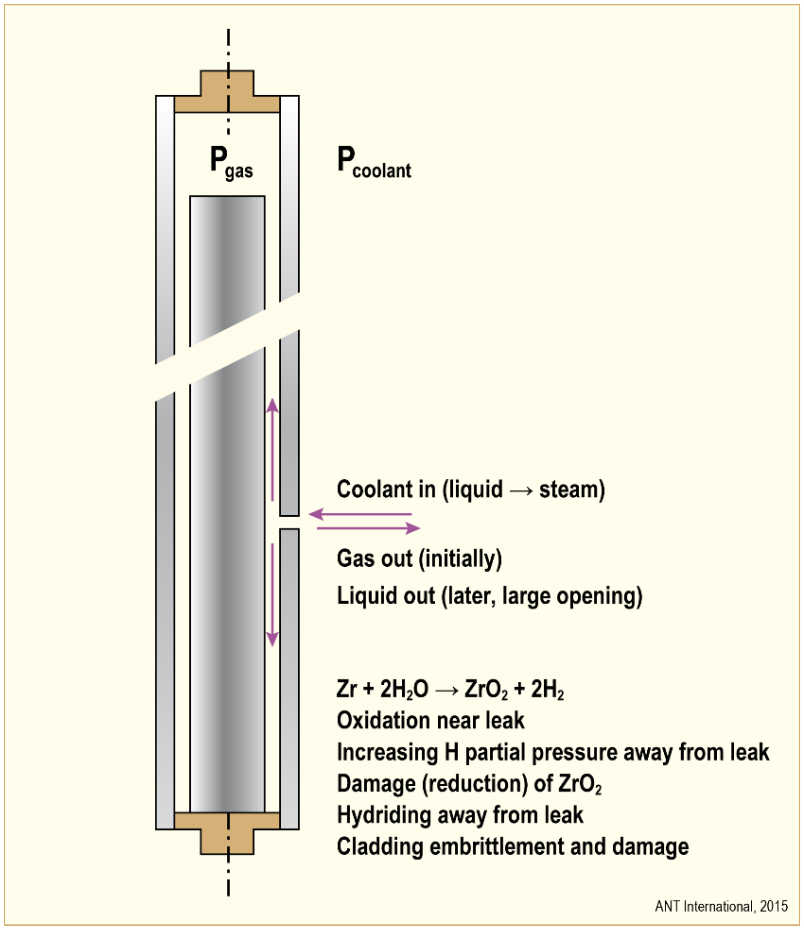


Figure 3-1: Schematic view of the exchange of primary coolant and radionuclides in a leaking fuel rod.

3.3.2 On-line monitoring of the primary coolant activity

Monitoring fuel reliability typically involves sequential, radiochemical measurements of the primary coolant and gas extracted from the primary coolant or the steam line (in BWRs). Measurements of the gross (total) beta or gamma activity of 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. In specific cases (as when fissile material is suspected of being released), alpha spectrometry can be carried out on a coolant sampling to detect presence of transuranic isotopes.

Detailed assessments are usually based on spectrometric measurements of coolant gamma activity to determine the composition and concentrations of radionuclides (fission products, but also activation products), with respect to their magnitudes and trends relative to time, burnup and plant maneuvers. In most reactors, the sources of such data are discrete samples of the primary coolant and offgas (BWR), or coolant and extracted-gas samples (PWR) that are processed in a gamma activity measurement laboratory (counting room). In some cases, and more recently, on-line measurements are being performed continuously with flowing samples routed through detectors, such as a γ-spectrometer, that are located near or connected to the coolant or offgas streams; e.g., [Parrat et al., 1991], [Sihver et al., 1999] and [Olsson et al., 2017]. Both monitoring methods are complementary: the on-line measurement deals with short half-life isotopes (approximately from a few tens of minutes to one day), because they dominate the gamma spectrum in terms of number of rays and background level. On the contrary, sampling is counted after different decay times (from few hours to few days) and middle- and long half-life isotopes are successively detected, as short half-life contributors progressively disappear from the gamma spectrum by radioactive decay. The radionuclides of interest include xenon and krypton isotopes in the coolant and gas samples, iodine and cesium isotopes in the primary coolant, and other nuclides that appear generally in the coolant with a large leak or after secondary degradation; e.g. Te, Ba, La, Rb, Sr, Ce, Np, etc.

An example of monitoring fuel reliability in a BWR is shown in Figure 3-2. This figure is a time-based plot of reactor power and the γ-activities of a long-lived fission gas, ^{133}Xe (5.25 d) and the sum of six isotopes of xenon and krypton with relatively long half-lives (called "Sum-of-Six" and 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 gamma spectrometry measurements of periodic samples. The initial spikes are due to the release of fission gases stored in the void volume of the leaking rod prior to failure.

In Figure 3-2, variations in the relative magnitudes of ^{133}Xe and the Sum-of-Six activities represent changes in the state of the leaking fuel rod subsequent to failure. The large initial increase in ^{133}Xe activity was due to its longer half-life and the larger inventory of this isotope in the rod inner free 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, see Figure 3-3.

Power suppression tests were conducted to locate the fuel assemblies with leaking rods. The leaking assemblies were confirmed by sipping and 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 subsequently deposited on core internals and fuel rods that were in the core when washout occurred and then continued operation in the next reactor cycle.

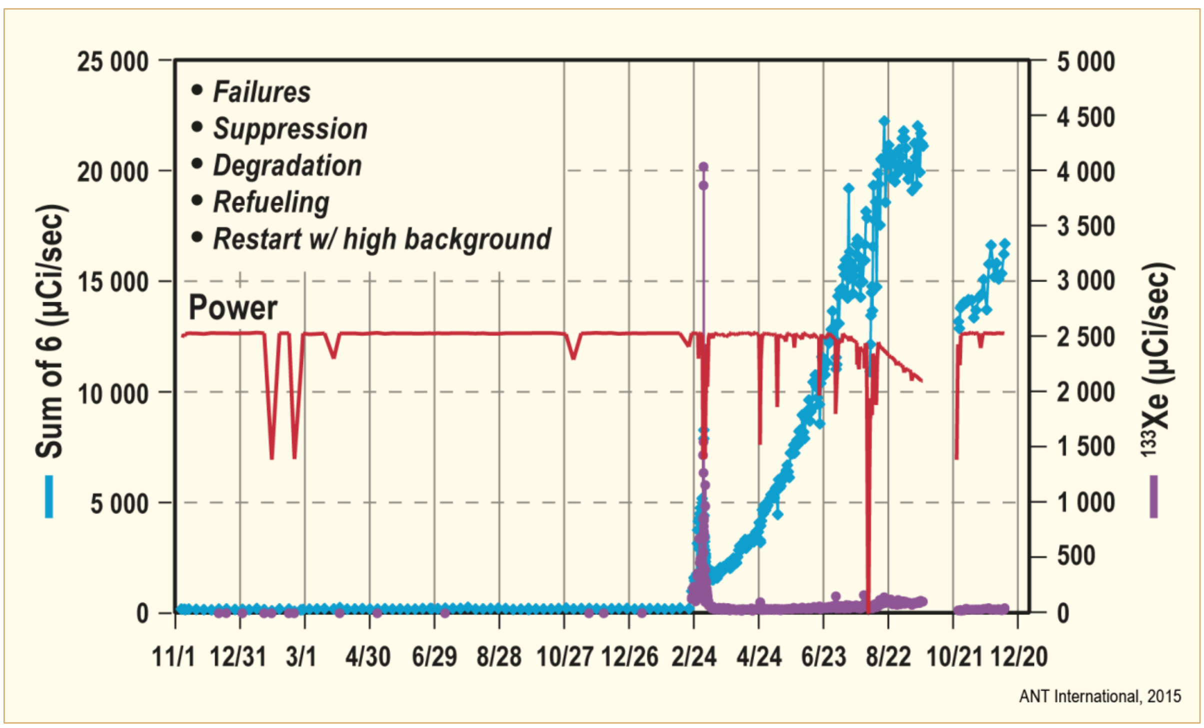


Figure 3-2: Example of a BWR fuel failure occurrence and resulting changes in offgas activity, modified according to [Yeager & Schneider, 2005].

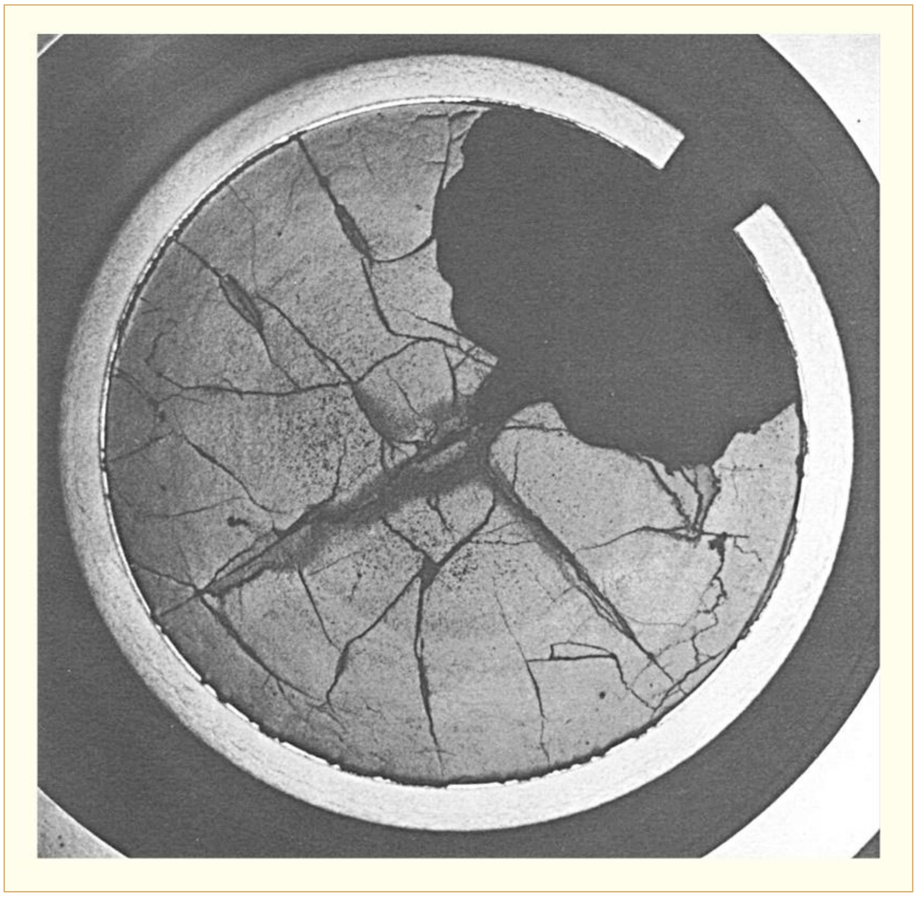


Figure 3-3: Macrography of a BWR rod, showing a secondary degradation of the Zr-lined, with a long axial cladding crack, modified according to [Yeager & Schneider, 2005].

3.3.3 Radionuclides used in reliability monitoring

3.3.3.1 Introduction

The fission products commonly used to monitor fuel performance are identified in Table 3-1. 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. In addition, the fission products ^{134}Cs, ^{136}Cs, ^{137}Cs, ^{91}Sr and ^{92}Sr, and the transuranics listed below are frequently used to monitor fuel performance during operation; i.e.,

- Neptunium ^{239}Np (PWRs and BWRs with normal water chemistry (no hydrogen));
- Plutonium ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu; (specifically if mixed oxide fuel is used in some assemblies)
- Americium ^{241}Am;
- Curium ^{242}Cm,^{243}Cm, ^{244}Cm.

The gamma activities of fission gases generally provide the first sustained indication of a new failure because they can escape through small leakage paths and do not deposit on surfaces. The fission gas isotopes ^{138}Xe (14.1 min) through ^{133}Xe (5.25 d) decay slowly enough to be collected and measured in counting rooms via “grab samples”. The sum of the activities of these isotopes, together with ^{135}Xe (9.10 h), ^{85m}Kr (4.48 h), ^{88}Kr (2.84 h) and ^{87}Kr (76 min) constitute the “Sum-of-Six” mentioned above. The activities of shorter-lived isotopes, ^{89}Kr (3.15 min) and ^{137}Xe (3.82 min), are sometimes measured in addition to those of the longer-lived nuclides by means of γ-spectrometers connected to flowing sample lines. They typically decay too rapidly for assessment by means of discrete samples since firstly the transport times from a leaking fuel rod to normal sample points are in the range of 3–6 minutes, and secondly the sample has to be transferred to the counting system, which also takes time.

Measurements of the activities of soluble fission products such as the isotopes of iodine listed in Table 3-1 are used alone and in conjunction with measurements of fission gas to identify and track the state of leaking fuel rods. The presence of iodine, cesium and other soluble nuclides (^{88}Rb, ^{89}Rb, ^{91}Sr, ^{92}Sr, ^{132}Te, …) during steady-state operation indicates mass transfer of coolant into and out of one or more leaking fuel rods. However, when the inner free volume of the leaking rod becomes empty of water), the release of these isotopes can be reduced dramatically compare to the gases release level, because they deposit on the inner surface of the clad; i.e., they are "trapped": see an example of Figure 3-4. Other soluble fission products or those released as metallic or oxide particles (^{91}Y, ^{92}Y, ^{99}Mo, ^{103}Ru, ^{106}Ru, ^{140}Ba, ^{140}La, ^{144}Ce…) are generally observed with larger leakage paths or following secondary degradation. The exception is the occurrence of spikes in the activities of soluble nuclides such as iodine, which can occur during shutdown, startup or other transients involving changes in pressure that promote the flow of coolant (in liquid phase or after intense vaporization) into and out of a leaking fuel rod.

When observed in measurements of coolant activity, the range of decay rates of the five iodine isotopes starting with ^{134}I (52.6 min) 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 intraspace and with increasing rates of release to the coolant. In some plants, a specification on the primary activity monitoring uses the ^{134}I isotope as a "signature" of fissile material release, and a quantitative assessment (relationship between the concentration of ^{134}I in coolant and the quantity of released fuel material) can be set up with some hypotheses on the type of fuel concerned and its burn-up.

Table 3-1: Tabulation of fission product isotopes useful in on-line monitoring with their respective decay constants, production rates and typical inventories, after [Rudling & Patterson, 2009].

Nuclide	Half-life	Decay constant 1/sec	Production rate ^{235}U	Production rate ^{239}Pu	Production rate ^{241}Pu	Production ratio ^{235}U	Production ratio ^{239}Pu	Production ratio ^{241}Pu	Inventory	Inventory ratio
Fission gas isotopes			MBq/(gm sec)			$^{133}Xe/x$			MBq/rod	$^{133}Xe/x$
^{89}Kr	3.15 m	3.66E-03	1.31E+02	3.99E+01	3.29E+01	6.10E-04	2.05E-03	2.40E-03	3.84E+07	2.33
^{137}Xe	3.82 m	3.02E-03	1.43E+02	1.37E+02	1.53E+02	5.59E-04	5.96E-04	5.16E-04	7.68E+07	1.17
^{138}Xe	14.1 m	8.19E-04	3.91E+01	3.05E+01	3.77E+01	2.04E-03	2.69E-03	2.09E-03	7.68E+07	1.17
^{87}Kr	76 m	1.52E-04	2.96E+00	1.08E+00	8.90E-01	2.69E-02	7.59E-02	8.88E-02	2.30E+07	3.89
^{88}Kr	2.84 h	6.78E-05	1.88E+00	6.94E-01	5.12E-01	4.25E-02	1.18E-01	1.54E-01	3.20E+07	2.8
^{85m}Kr	4.48 h	4.30E-05	4.34E-01	1.88E-01	1.31E-01	1.84E-01	4.36E-01	6.05E-01	1.28E+07	7
^{135}Xe	9.10 h	2.12E-05	1.07E+00	1.22E+00	1.17E+00	7.42E-02	6.73E-02	6.77E-02	2.62E+07	3.41
^{133}Xe	5.25 d	1.53E-06	7.97E-02	8.19E-02	7.90E-02	1	1	1	8.96E+07	1
Iodine isotopes						$^{131}I/x$				$^{131}I/x$
^{134}I	52.6 m	2.20E-04	1.30E+01	1.22E+01	1.37E+01	1.70E-03	2.36E-03	1.80E-03	9.60E+07	0.43
^{132}I	2.28 h	8.45E-05	2.80E+00	3.43E+00	3.06E+00	7.87E-03	8.39E-03	8.03E-03	5.95E+07	0.69
^{135}I	6.61 h	2.91E-05	1.41E+00	1.42E+00	1.55E+00	1.56E+00	2.03E-02	1.58E-02	8.32E+07	0.49
^{133}I	20.9 h	9.21E-06	4.80E-01	4.91E-01	4.77E-01	4.59E-02	5.87E-02	5.15E-02	8.96E+07	0.46
^{131}I	8.04 d	9.98E-07	2.21E-02	2.88E-02	2.46E-02	1	1	1	4.10E+07	1

Notes:
10 × 10 rod
2% fissile material (^{235}U or ^{239}Pu)
24 W/(g HM)

ANT International, 2015

Note that measurements of fission product activity via coolant samples can be confounded both during and soon after the end of operation with significant fuel leakage due to the effects of fission products in the cleanup systems or on the surfaces of fuel rods. Such confounding occurs also in the sampled fluid itself, when it is counted after several days of decay. An example is the decay of tellurium to iodine, xenon and then cesium. Issues can arise when measurements are made after power or flow transients or soon after shutdown, as in sipping during fuel shuffling. On-line measurements tend not to be affected by such precursor effects when performed during steady-state operation – a common practice.

In addition to fission products, fuel can be transported from a leaking rod when fuel pellets are exposed to flowing coolant and its radiolytic decomposition products. In such conditions, $UO_2$and MOX fuel pellets oxidize first along their grain boundaries and then into the affected grains. The oxidation of UO_2 to U_4O_9 produces a slight volumetric contraction; i.e., $\Delta V/V \sim -2\%$. Continued oxidation leads to the formation of U_3O_8 with a large volumetric expansion, +36%, which causes stress on the clad material and can directly affect the leak size. Further exposure to coolant can produce small, hydrated crystals similar to the mineral schoepite, $(UO_2)_4O(OH)_6 \bullet 6(H_2O)$. The oxidation process and volumetric strains weaken the pellet matrix and form grain-size and smaller particles that can be transported from a leaking fuel rod by flowing coolant as shown in Figure 3-3. This degradation is even very efficient when submicronic grains are already present at the pellet surface, coming from the "high burn-up structure", HBS (it is a porous peripheral layer formed at intermediate and high burn-up and at low temperature).Fuel particles which are distributed throughout the primary system are called tramp uranium or, more generally, a tramp source as noted earlier. Tramp sources in the active core produce fission products as does the fuel in a leaking rod.

4 Poolside assessments

If a core is suspected or known to contain leaking fuel based on the methods discussed in the previous sections, the fuel is almost universally checked for leaks during the subsequent refueling (or sometimes forced) outage. In many cases, all fuel assemblies of the core are checked for leaks using failure detection systems that either do not affect or have only a minor impact on outage time. In a few cases, when failure diagnosis made during the cycle gives unambiguous information on the failed rod(s) (e.g. a specific set of assemblies based on a narrow range of burn-up or power suppression testing), it is possible that only a part of the fuel assemblies is checked.

As LWR fuel assemblies are now dismountable and "reparable"[6],identifying individually the leaking rod(s)enables continued irradiation of the affected assemblies after the leaker(s) have been replaced with suitable, non-leaking rods. This approach, based on technical and economic considerations in relation with plant operation specifications, is clearly valuable when the assembly has to be irradiated during at least two more cycles. When only one more cycle remains, other information is typically considered to make a decision on its reutilization; e.g., length of the next cycle, number of detected failed fuel assemblies[7], burn-up already reached, availability of equivalent fresh rods etc. Moreover, as an assembly repair campaign is expensive, time consuming and requires space in a fuel handling pool, repairs are often carried out not during the concerned refuelling outage, but later, when the plant is operating. Repair operations are sometimes performed when several leaking fuel assemblies are present on the plant site and are gathered in the same pool to take advantage of a "serial effect". However, the timing of such repairs is affected by constraints such as the need to resume plant operation with suitable fuel that is available on site and the ability to store the leaking fuel for extended intervals (one to a few years). Sometimes these situations are not easily manageable by the plant operator.

Independent of of the resource and scheduling issues, poolside assessments involve identifying the leaking fuel assemblies and then identifying the leaking fuel rods in the affected assemblies if they are to be repaired for continued irradiation or if a detailed failure investigation is needed. The methods for identifying the leaking fuel are reviewed in te sections which follow.

4.1 Identification of failed fuel assemblies by "sipping test" methods

4.1.1 Physical principles applied in a sipping test

Various techniques are used to identify leaking fuel assemblies and are collectively referred to as "sipping" methods. Although the used equipment can vary considerably, the underlying method is basically the same: it consists of placing the checked assembly in a closed, or partially closed, environment to create physical local conditions and provoking a release of radionuclides from out of the failed rod to this environment. Radioactive isotopes of fission products are more are used as a "signature" of the leak presence. These conditions correspond the most often to a relative pressure discrepancy between the inner free volume of the rod and the external pressure, at the defect level. This relative discrepancy can be made:

- Either by increasing the rod internal gas pressure (and maintaining the external pressure constant by other means). This change is generally achieved by increasing its temperature by a few tens of °C. Temperature increase can be due to the decay heat of the fuel assembly and/or to a supplementary external heating. There are two contributions to this pressure increase (see Figure 4-1):

[6] Reparable in this case means that each fuel rod can be extracted from the assembly and replaced with another fuel or dummy rod (e.g., solid Zr-alloy rod) for continued operation.

[7]The term « failed/leaking fuel assembly » is often used: it means that one or several untight fuel rod(s) is/are present inside the assembly

- as the inner free volume is filled by non-condensable gaseous products (mainly ^4He coming from rod pressurization at manufacturing, xenon and krypton isotopes coming either directly from fission, or after radioactive decay of a father "isotope", iodine or bromine respectively, and H_2 coming from water radiolysis), the Boyle-Mariotte law PV=nRT is active and is used in practice with its differential form : $\Delta P/P = \Delta T/T$.
- as liquid water is often present in a failed rod (see previous chapters), temperature increase provokes vaporization of a part of this liquid and formation of a vapor pressure following the saturation law. When the local temperature is higher than about 65°C, this effect becomes more active than the first one.

- Or by decreasing the external pressure. This can be achieved:
  - by a vertical ascent of the fuel assembly in the reactor pool or in the storage pool of the plant. The external pressure corresponds to the hydrostatic pressure at the defect elevation (i.e. the depth in the pool), so it decreases,
  - by applying vacuum to an upper gas blanket when the checked assembly is enclosed in a "sipping cell" (see details below).

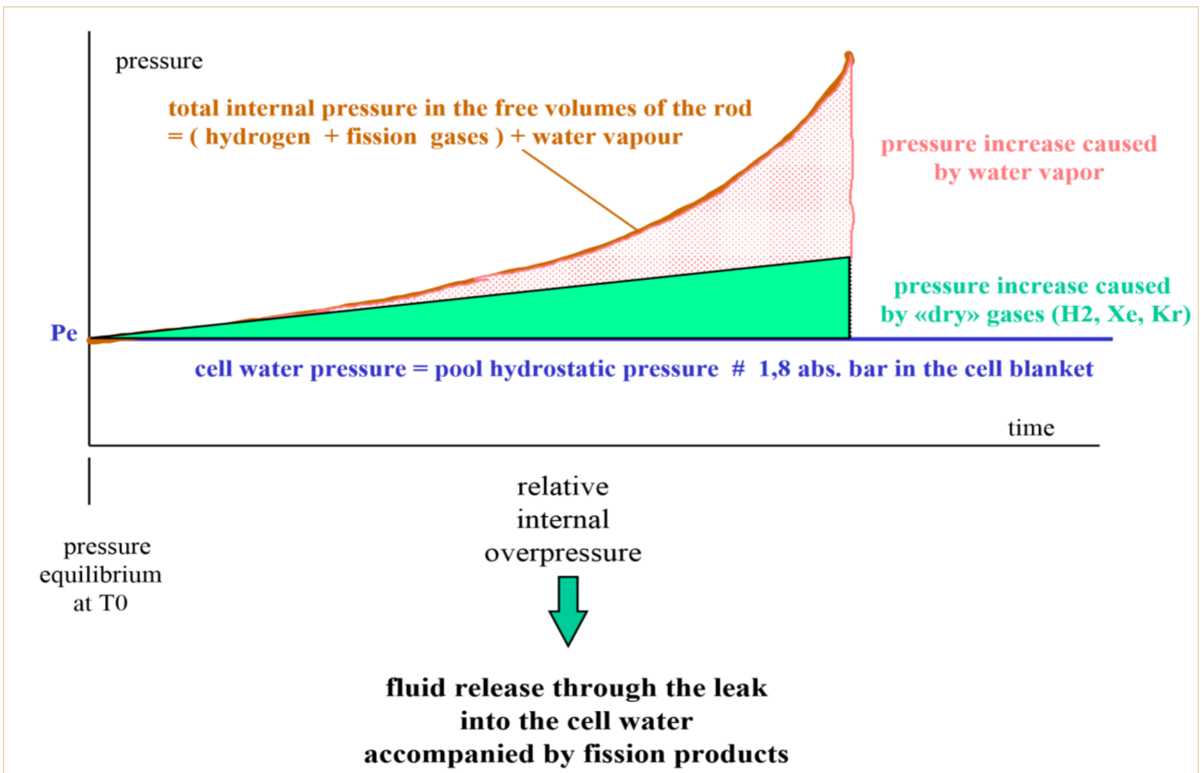


Figure 4-1: Effect of the heating of the free volumes of a failed fuel rod on the release of fission products (temperature increase is considered as proportional to the time).

For both physical phenomena, the achieved pressure difference with standard sipping methods is about 0.8 to 1 bar. The fluid transiting through the failure can be either gas or liquid water, depending on the composition of the inner free volume in the vicinity of the failure. Fission products are transported by the released fluid. If xenon and krypton isotopes are routed either by gas or by liquid in solubilized mono atomic form, a liquid phase is able to carry solubilized (or under the form of small particles) chemical compounds formed with other fission product elements: iodine, caesium, molybdenum, ruthenium, barium, lanthanum.

4.1.2 Sipping test assessment and analysis of the activity signal

The detection of a failure relies on changes in the concentration of fission products that are released from the fuel assembly or rod being tested; i.e., changes in gamma activity, beta activity, isotopic composition of fission products or combinations of such measurements. These changes can be monitored either by on-line measurement (e.g. by total beta or gamma activity, or by on-line gamma spectrometry), or by sampling with delayed analysis. In both cases, the fluid in contact with the fuel rods shall be routed to the measurement system or to a sampling (collection) point. The technique used to identify a leaking assembly can vary depending on the size of the leak, the background activity from tramp uranium and on the time of sipping relative to reactor shutdown. Changes in the gross (total) gamma activity of water or noble gas samples representative of the fuel being tested are sometimes used to identify a leaker, particularly in cases of low background activity. In general, however, changes in the gamma or beta activity of nuclides with moderate-to-long half-lives are typically used to minimize the effects of background activity and decay time. Isotopes such as^{133}Xe (5.25 d), ^{131}I (8.04 d),^{85}Kr (10.72 y), ^{134}Cs (2.06 y) and ^{137}Cs (30.2 y) are the most detected. Detection of less volatile isotopes, such as ^{99}Mo (66.0 h), ^{140}Ba (12.75 d), ^{140}La (1.68 d) etc. indicate generally presence of a relatively large defect. Finally, detection of non-volatile isotopes, such as ^{103}Ru (39.3 d), ^{106}Ru (373 d), ^{144}Ce (284.9 d) or heavy isotopes (^{239}Np, 2.36 d) indicates a probable release of fissile material through a large defect. In addition, many of the current sipping methods also involve the collection and measurement of noble gases to enable the detection of leaks that are too small, or when the inner free volume of the rod is filled only with gas, to allow the release of soluble fission products in quantities clearly detectable in the sipping process.

4.1.3 Main categories of sipping test methods

Historically, sipping methods have been classified as wet, dry or vacuum based on the manner in which fission products are collected for measurement [Lin, 2013]. In practice, however, sipping techniques can better be classified as ”open” or ”closed” methods based on the manner in which the fuel being tested is isolated from other assemblies or from its environment during the sipping process. This section gives an overview of each method. Additional information is provided on the methods in the sections that follow., which are described individually more in details in the following paragraphs: Sipping methods comprise:

- The wet, in-reactor building methods represented by the “TELESCOPE sipping”, “IN-MAST sipping”, “ON-LINE sipping system OLSS” and various hood systems are open methods, which take benefit either from the pressure differential produced by the positive altitude variation when the assembly is unloaded from the core or from temperature increases They have become the primary means of leak detection because of the small amount of supplementary time needed to inspect a BWR, PWR or VVER core; e.g. about 16 hours for a large BWR with an in-core sipping hood, versus close to a week for vacuum sipping [Knecht et al., 2001]. They are simple and passive systems that quickly provide information about the presence of absence of fuel rod leaks during the refueling operations. Although the open methods have become the standard for sipping, a disadvantage is that their resolution can be insufficient to detect small leaks, particularly when large leaks have occurred and background activity is high.

- Wet sipping is also used in storage pools to test individual assemblies or fuel rods. It makes use of a cannister (called “sipping cell”), anchored permanently on the bottom of the spent fuel storage pool or installed temporarily for a specific sipping campaign, to isolate the fuel being tested. Release of fission products is provoked by heating of the cell water. It is a closed test method similar to the vacuum sipping method described below which provides good resolution but is slow relative to the open methods.

- The dry sipping method uses also a cannister but is not used today in power reactors because of issues related to handling and test time, decay heat and cladding temperature.

- The vacuum sipping method also makes use of a cannister, located in the storage pool, to isolate individual fuel assemblies. As for the wet in-cell sipping, vacuum systems include a gas circuit to facilitate the fission product measurement or collection. It is frequently used to supplement the in-reactor methods, because of its higher resolution and detection capabilities. Moreover, a detectable signature is obtained in favourable cases after a few minutes, and it is not necessary, compare to wet in-cell sipping, to wait for the effect of the heating, which starts to be detectable after about 10 or 15 minutes. This method utilizes decay heat and does not require an external heat source, which is an advantage regarding the technology of the circuits, the management of the steam in the gas circuit and the risk of associated contamination.

Note also that the results of power suppression tests are sometimes used in conjunction with the wet, in-reactor methods to decrease the number of bundles that must be sipped and to decrease the chance of missing a leaking assembly. This is the case in some BWRs with an operation called flux tilting or power suppression testing as described earlier.

4.1.4 Wet in-core sipping systems

The configuration of wet, in-core sipping systems varies according to reactor type. In BWRs, where each fuel bundle is surrounded by a flow channel, sipping can be performed with a sampling hood at the top of a fuel assembly or a group of assemblies. An example of a multi-assembly sipping hood and measurement system is shown in Figure 4-2. In this system, the hood is open at the bottom and divided into four quadrants which match the core cells. The hood is placed on the upper core support grid over a 4 x 4 array of fuel assemblies by means of the refueling mast. Air is injected into the top of the hood to displace water and expose the upper ends of the fuel channels, which extend slightly above the support grid. The air bubbles isolate the assemblies covered by the hood and interrupt the convective flow of cooling water through the bundles. As a result, the temperature of the fuel increases and fission products are driven out of any leaking rods that might be present. The fuel channels prevent cross mixing of released fission products among the 16 assemblies. Suction probes, which also include thermocouples to monitor water temperature, extend down into the fuel channels. After fuel heatup, water is drawn from the fuel channels and passed through gas separation and scanning systems. Leaking fuel is identified by continuous measurement of the extracted gas to detect changes in the activity of ^{133}Xe or ^{85}Kr.

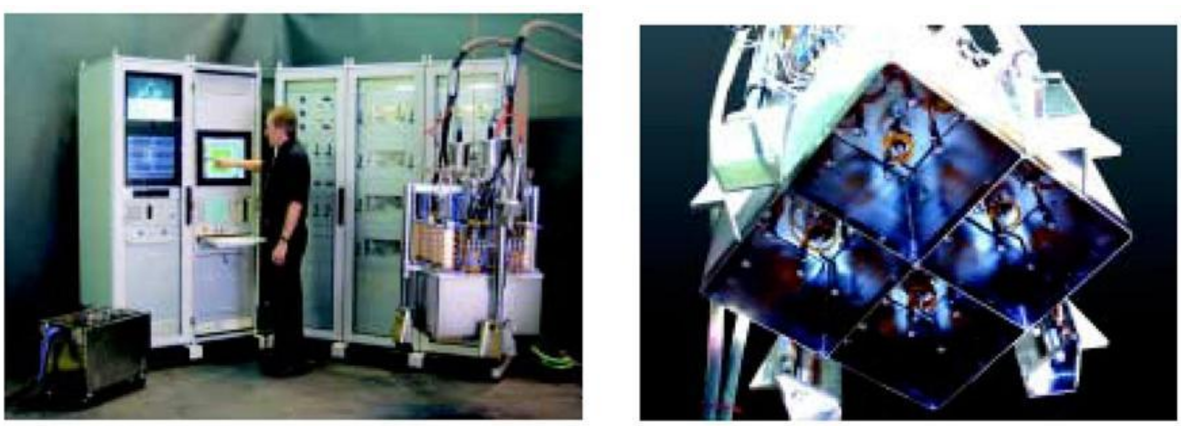


Figure 4-2: Wet in-core sipping system with control and data acquisition modules, detector and housing and 16 bundle sampling hood, modified according to [Knecht et al., 2001].

This concept has also been applied to individual fuel rods by replacing the in-core hood with a tubular sleeve. The single-rod technique is used in fuel handling or storage pools, where rods can be extracted from fuel assemblies. As shown in Figure 4-3, a fuel rod is removed from its bundle and drawn vertically into the sleeve. The sleeve isolates the rod from the pool and interrupts the convective flow of cooling water, so the fuel temperature increases due to decay heat. Fission products diffuse or flow out of a leaking rod at a rapid rate because of the increase in temperature inside the fuel rod and because of the decrease in ambient pressure caused by raising the rod into the sleeve. The concentration of released fission products in the water within the sleeve is enhanced by the small annular volume between the sleeve and fuel rod. Samples are drawn from the top of the sleeve and routed through separation and measurement systems in the same manner as the in-reactor, hood or mast sipping systems. Water samples can also be analyzed for the presence of soluble fission products such as ^{131}I or ^{137}Cs.

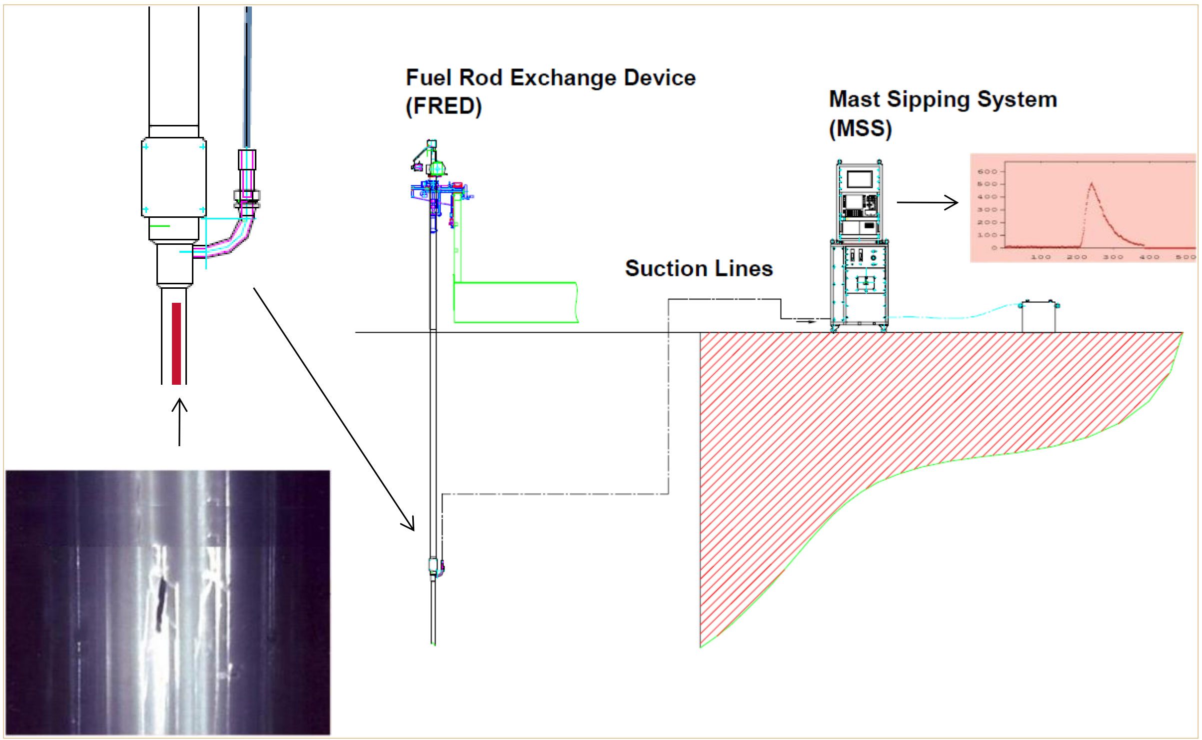
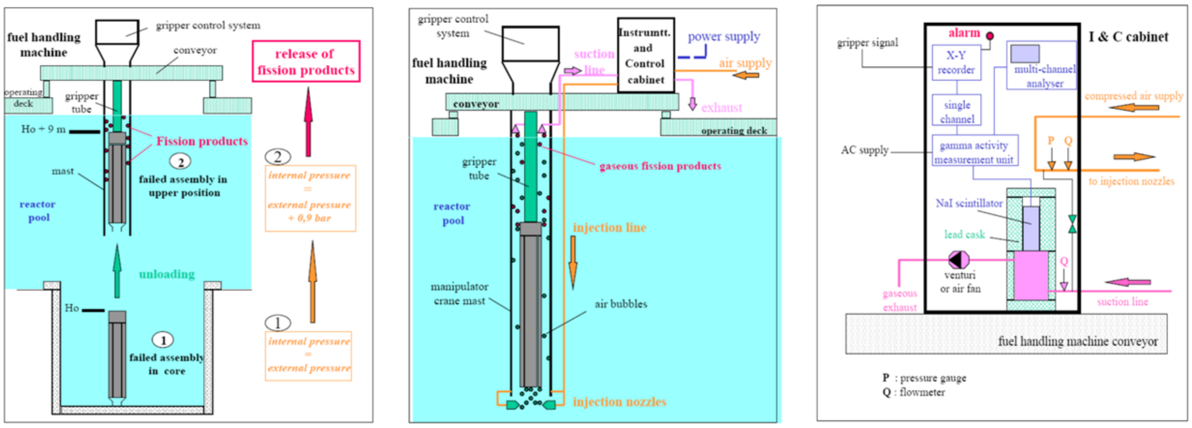


Figure 4-3: Wet, in-pool system for sipping individual fuel rods, modified according to [Dust et al., 2008].

In-core wet systems for sipping PWR fuel differ from those used with BWR fuel because of the absence of a flow channel around PWR fuel bundles. In a PWR, sipping is performed by lifting a bundle into a tube that surrounds the refueling grapple. The tube isolates the fission products released from the bundle being tested from the balance of the core. The release of fission products from a leaking rod is enhanced by the change in elevation as the bundle is lifted into the tube (see Figure 4-4, "phase 1"). As shown in Figure 4-4 "phase 2"', air is also sparged upward through the tube and fuel assembly to enhance the transport of gaseous fission products from the fuel to the collection port located at a higher elevation in the tube. The collection port is connected to a suction line which routes the noble gases to an on-line gamma activity analyser, e.g. with a window centred on the 81 keV gamma ray of the ^{133}Xe (see Figure 4-4, "phase 3"). Figure 4-5 details the air circuit used for sweeping the fission gas dissolved in water or present in the form of small bubbles. Leaking fuel is identified by changes in the gross or ^{133}Xe gamma or beta activity as shown in Figure 4-6.

Testing is typically performed during refueling outages when fuel is moved among core positions. Detection is at best when the fuel assembly has reached the upper position in the mast of the handling machine and when this machine is stationary. The process is reported to add only a few minutes to each move so that the incremental time to test an entire core is less than a day; e.g., incremental time of 2 minutes per move and about 5 hours for a 900 MW core, [Beuneche et al., 1988].



Phase 1:
Fuel assembly lifting
Release of fission products

Phase 2:
Fission product recovering
Routing to measurement system

Phase 3:
Activity signal assessment
I&C unit

Figure 4-4: On-line sipping system in the tube of the handling machine in a PWR reactor - Successive phases of a fuel assembly tightness monitoring.

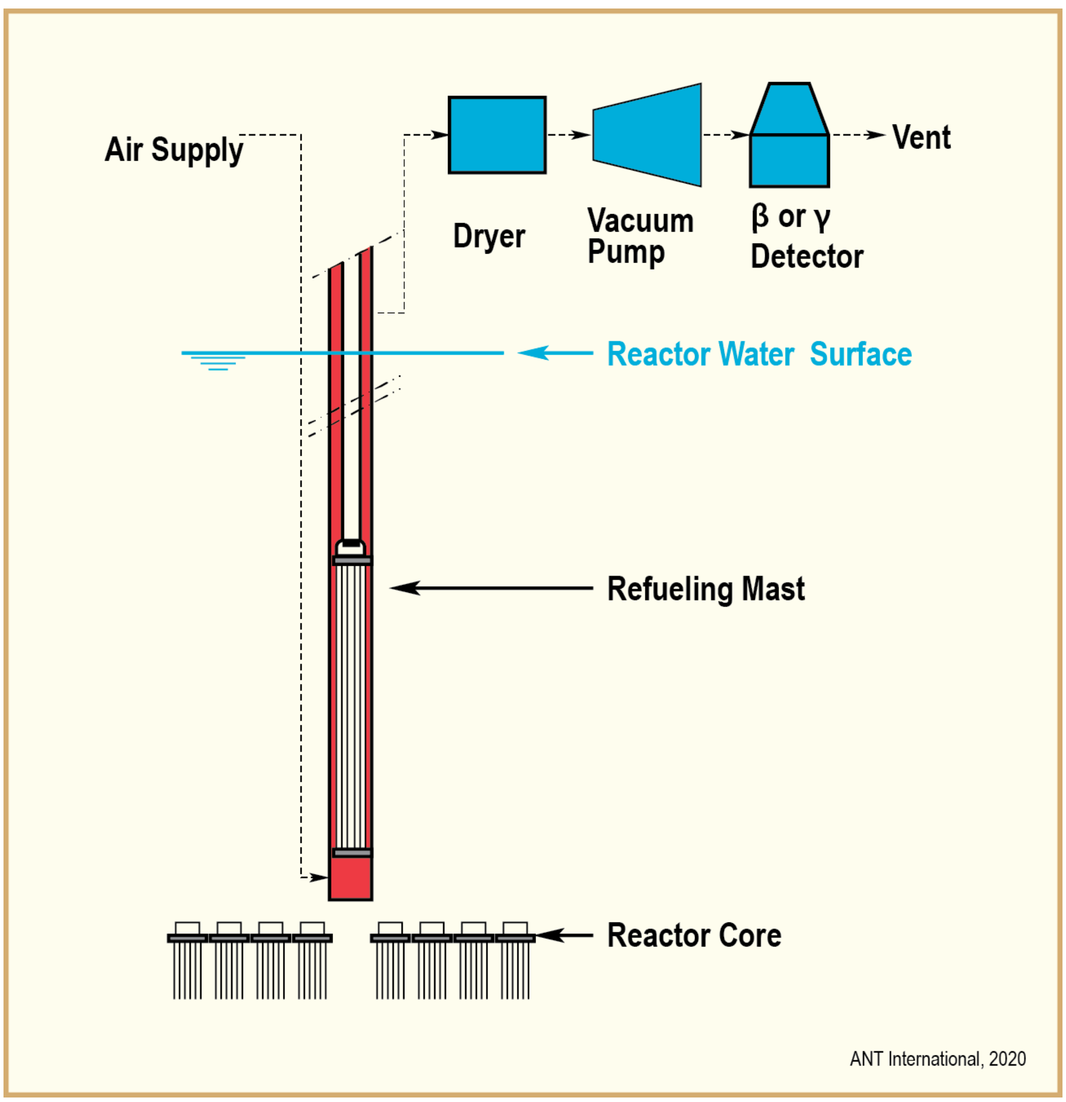


Figure 4-5: Schematic diagram of in-mast sipping of PWR fuel assembly, modified according to [Beuneche et al., 1988].

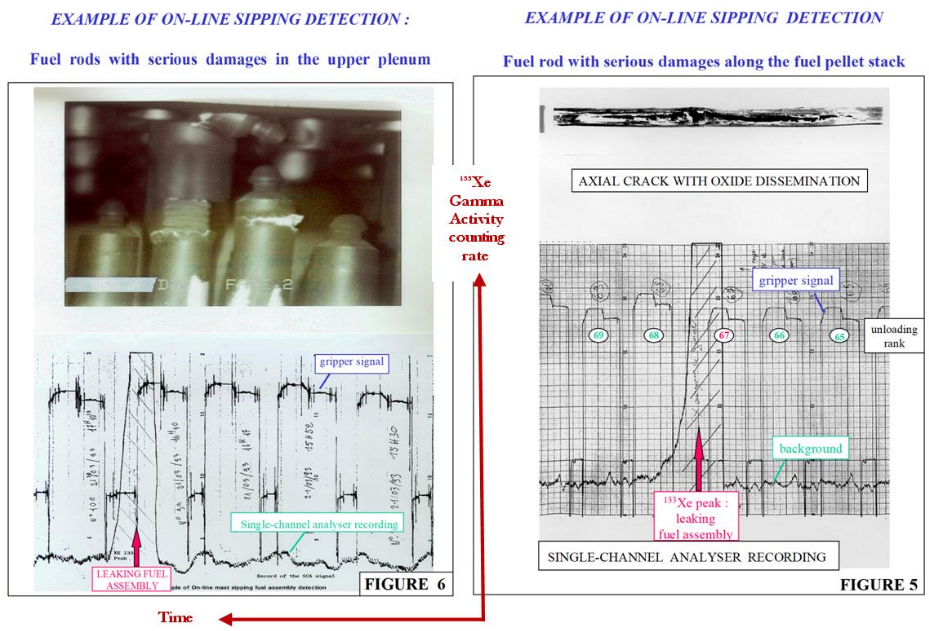


Figure 4-6: Examples of change in gamma activity due to a leaking fuel assembly during in-mast sipping, modified according to [Parrat et al., 1997].

4.1.5 Wet sipping methods using a canister

The canister used for wet sipping is located in a fuel handling/storage pool and is initially full of pool water. When an assembly to be tested is placed in the canister and the lid has been closed, the water is heated either by the thermal power (decay heat) of the enclosed assembly or by electrical heaters installed either at the cell bottom or along a closed circulating water circuit going through a cubicle located on the pool deck. Fission products are released into the circulating water due to the temperature increase and can be routed either by a closed gas circuit to an on-line activity detector or a sampling point (for gaseous fission products) or by a closed water circuit for all radionuclides. In the "gas loop", gaseous fission products are conveyed to the activity detector or sample port with either air or nitrogen as the carrier gas. In the "water loop", due to the large number of middle and long half-life radionuclides routed (fission products and activation products), it is preferable to perform only a water sampling and then to optimize the delayed analysis (decay time, counting duration, removal of activation products if necessary etc.) to detect isotopes indicating without ambiguity the presence of a leaker. Figure 4-7 and Figure 4-8 show a design of a wetsipping cell.

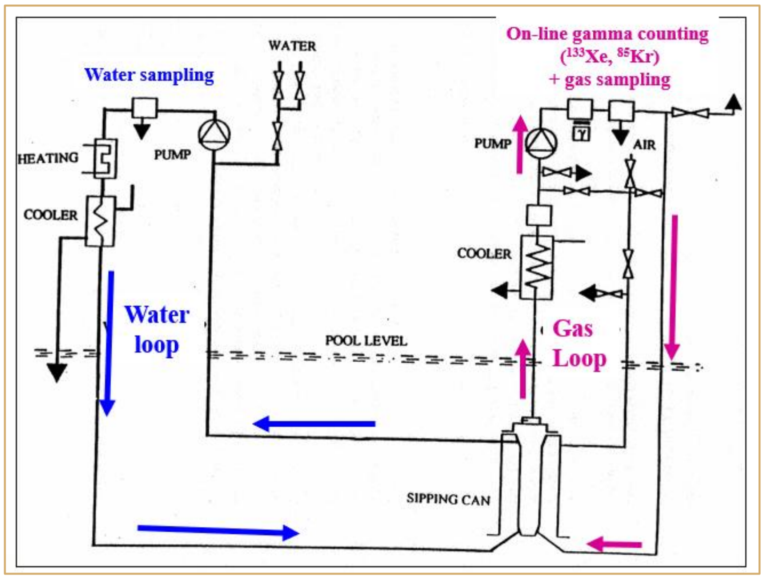


Figure 4-7: Schematic of a sipping cell equipped with a closed gas circuit and a closed water circuit (Framatome design).

5 Hot Cell Examination assessment

The methods utilized in hot cell examination vary with the objective of the campaign, but usually begin with a visual inspection. Visual inspections are used to characterize the external surfaces of a fuel rod, including the *CRUD* and oxide layer, and to identify features of interest for subsequent examinations. High resolution diametral profilometry and *EC* lift-off measurements are also used to characterise the thickness of the *CRUD* and oxide layer relative to axial and azimuthal position, particularly if such measurements were not performed before transporting the rod to the hot cell.

EC measurement with encircling, pancake or pencil coils can detect the axial and tangential location of cracks and other damage that might not be readily discernible in visual inspections; e.g., small or incipient cracks due to *PCI*.

Neutron radiography - If a test reactor is adjacent to a hot cell, neutron radiography can be performed of the fuel rod to non-destructively locate regions with different hydrogen levels in the cladding, Figure 5-1 and Figure 5-2. Depending on the method for capturing and displaying the neutrographic image, accumulations of hydrogen are indicated by light or dark areas. As shown in Figure 6-7 neutron radiography can also reveal fuel washout. In addition, neutron radiography provides geometrical information of the fuel column for the selection of cutting positions of samples for ceramographic examinations.

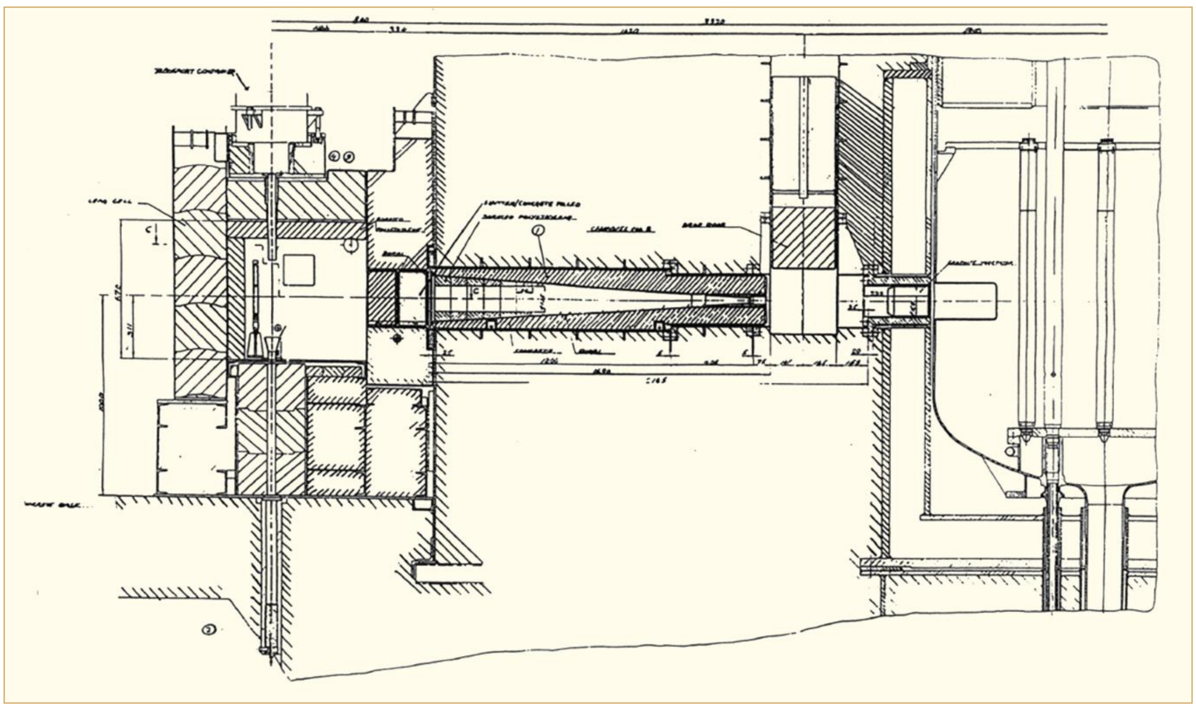


Figure 5-1: Schematic view of hot cell with a neutron radiography capability. The hot cell is to the left while the reactor is to the right, modified according to [Jenssen & Oberländer, 2001].

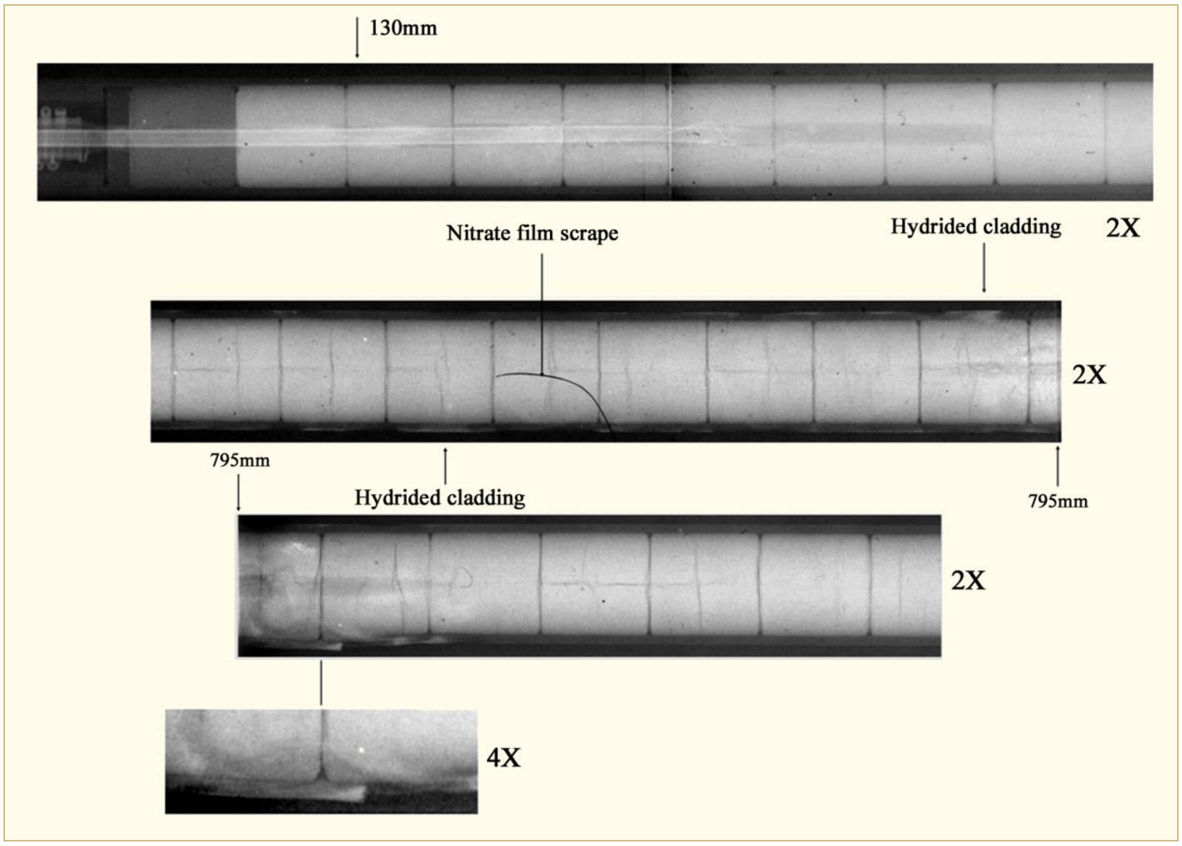


Figure 5-2: Neutron radiographs of a fuel rod subjected to a fuel degradation test, modified according to [Jenssen & Oberländer, 2001].

Figure 5-3: Neutron radiography showing hydrides in fuel claddings (the whitish area to the right).

Destructive examinations – The fuel and cladding can be examined by means of a number of destructive methods; e.g.:

- Fission gas collection and analysis.
- Cladding oxide thickness (*ID* and *OD*) measurement.
- Hydride concentration and morphology.
- *SPP* size and distribution.
- Microstructure characterization.
- Ceramographic characterization of fuel pellets.
- Micro-gamma scanning across the diameter of fuel pellets.
- Electron microprobe analysis of the fuel pellets and the pellet-cladding interface.

Metallographic mounts of the cladding and fuel may be prepared, polished, and photographed. The mounts may be examined for clad oxide layer (external and internal), fuel/clad interactions, fuel restructuring, rim effect, and agglomerate behaviour. The mounts may also be etched to reveal grain boundaries for an analysis of the grain size.

For more information about Hot Cell examinations, see: ZIRAT21 STR on Hot Cell Post-Irradiation Examination Techniques Vol. II [Mahmood et al., 2016] and [Mahmood et al, 2014] ZIRAT19STR on “Hot Cell Post-Irradiation Examination Techniques for Light Water Reactor Fuels”.

6 Fuel reliability improvement

6.1 Introduction

For political and economical reasons, there are objectives to reduce the primary fuel failure frequency as much as possible and to minimize the consequences of fuel failures when they occur.

Fuel failures can have adverse impact on:

- Power generation.
- Outage time.
- Chemistry and radiation monitoring costs.
- Personnel exposure.
- Handling, transportation and storage or reprocessing.

The cost per failure was recently estimated by [Lemons, 2008] to range from $1 000 000 to more than $20 000 000 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 *FA*.

While *PWR*s 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 leads 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 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.

The Executive Board of the Institute of Nuclear Power Operations (*INPO*) set the goal of eliminating fuel failures in all U.S. plants by 2010 [*INPO*, 2007]. The Chief Nuclear Officers of U.S. utilities have since pledged their support. While the feasibility of “zero leakers” as opposed to “as low as reasonably achievable” remains to be established, an integrated approach by suppliers, operators and regulators to achieving ultra-high reliability fuel is expected to benefit operational, cost and exposure issues. A key factor in such an approach is the examination and analysis of irradiated fuel at reactor sites, 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 failed and non-failed 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 Handbook is to provide such an understanding.

Failures due to the first two causes, manufacturing defects and cladding collapse, are relatively infrequent. The exception is pellets with chips or missing pellet surface. Such pellets have been involved in recent *PCI* failures and are the object of ongoing manufacturing development. Another potential exception is primary hydriding. Failures due to primary hydriding were effectively eliminated by the introduction of fuel pellets with little or no open porosity and by the exclusion of moisture and other hydrogen-bearing material from the inside of fuel rods during the final assembly process. However, the underlying risk still exists and could lead to fuel failures in the event of an undetected excursion in moisture contamination. Fuel failures due to excessive cladding corrosion are also relatively infrequent. Corrosion failures typically result from a common cause or a set of causal factors and can affect a large number of rods in a given reload or core. The rate of *PCI* failures was greatly reduced by introduction of operating guidelines (on ramp rates) but appears to be increasing with more demanding core loadings and operating cycles; i.e., longer cycles, more rods at the upper end of the design range, larger power ramps to high powers after longer times at low power. Some of the recent duty-related failures have also involved pellets with missing cylindrical surface. Cladding perforation due to fretting remains as the leading recurrent failure mechanism. Fretting due to the trapping of foreign material next to fuel rods is a source of failures in *PWR*s. The frequency of failures due to debris fretting varies among reactors, ranging from none in some plants to multiple rods in multiple cycles in other plants. The frequency of such failures is decreasing, however, due to debris filters in fuel bundles, strainers in feedwater lines and debris exclusion practices by the fuel suppliers and reactor operators. Fretting due to fuel rod vibration relative to spacer grids is a problem unique to some *PWR* fuel designs and plants.

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 6-1:

- 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 6-1: Fuel failure causes.

Failure cause	Short description	Reason
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
Excessive Zr alloy corrosion	An accelerated corrosion process results in cladding perforation. This corrosion acceleration can be generated by e.g., CRUD deposition, dry-out due to excessive fuel rod/assembly bowing.	Poor design or fabrication process or non-optimum coolant chemistry
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 fuel causing fretting failures. Chipped pellets may result in PCI failures.	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 PWR. Grid-rod fretting – Excessive vibrations in the PWR 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.

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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.

6.2 Assessment of fuel failure causes

6.2.1 Classification of primary failure causes

The primary fuel failures can be separated into duty-related and non-duty related failures. The former type of failures is a result of a power transient while the latter may occur irrespective of a power transient. The failures often occur at different burnups dependent on failure mechanism.

The following subsections describes the failure mechanisms, shows examples and identifies the principal characteristics of each type of failure; e.g. the burnup ranges when they occur. Therefore, three different burnup ranges are defined as follows:

- Low burnup – up to about 15 MWd/kgU.
- Intermediate burnup – 15 to 30 MWd/kgU.
- High Burnup – Exceeding 30 MWd/kgU.

These subsections also provide information on what additional data may be retrieved before the poolside/hot cell campaign to facilitate assessment of the failure cause.

6.2.2 Duty related failures – PCI/PCMI

6.2.2.1 Introduction

To prevent *PCI* failure, it is necessary to remove at least one of the fundamental conditions (tensile stress, sensitive material, aggressive environment), Section A.1.3 which cause *SCC*. There are two principal types of remedies.

One remedy is to develop reactor operation restrictions that will ensure that the cladding stresses will always be below the *PCI* threshold stress during power increases. This is the main measure to avoid *PCI* defects and the only measure used in *PWR*s. Operating rules (also called management recommendations, or Preconditioning Interim Operating Management Recommendations (*PCIOMR*s) to limit local power increases and “condition” fuel to the effects of power ramping were implemented during the late 70s to resolve the *PCI* issue. The rules are usually a function of exposure or exposure and power change, and were developed by the different fuel vendors, so they differ between various fuel types. To establish and validate these rules, extensive power ramp tests were performed by the fuel vendors in experimental reactors.

The second remedy — design improvement — consists of several approaches.

a) Cladding design:

   - Development of radial cladding texture and small grain size may increase cladding *PCI* resistance [Garzarolli, 2001].

b) Pellet design:

   - In the early 1970s, it was postulated that *PCI* performance could be improved by reducing the cladding local strains by shortening the pellet, chamfering the corners and eliminating the dishing. Such changes were known to reduce pellet-cladding mechanical interaction by both experimental data and analytic results and were implemented to varying extents by *BWR* fuel suppliers. Given that *PCI* failures were also observed in rods loaded with compacted powder, however, the pellet design changes were a means for increasing margin to such failures.

- Pellet dopants are being developed for *PWR*s that are designed to reduce the mechanical loading imposed by fuel pellets on their cladding during power ramps and thereby reduced the risk of *PCI*/*PCMI* failures.

Due to more aggressive core loading patterns in *US PWR*s combined with pellet manufacturing defects, *PCI* failures have reoccurred. Although *PCI* tends to occur in *BWR*s, it is noteworthy that some *PCI*-suspect failures have been experienced in three Babcock & Willcox (*B&W*)-designed *US PWR* plants following the movement of Axial Power Shaping Rods (*APSR*s) even though the calculated stress levels remained within the permissible range. Figure 6-1 gives an example of *PCI* failures that occurred in a *PWR* during cycle 5 and 6 of the KraftWerkUnion (*KWU*) designed Obrigheim *PWR* plant. The *PCI* failures occurred at the lower rod elevation that experienced the largest power increase while secondary defect (secondary hydride blisters) were formed in the upper part of the rods.

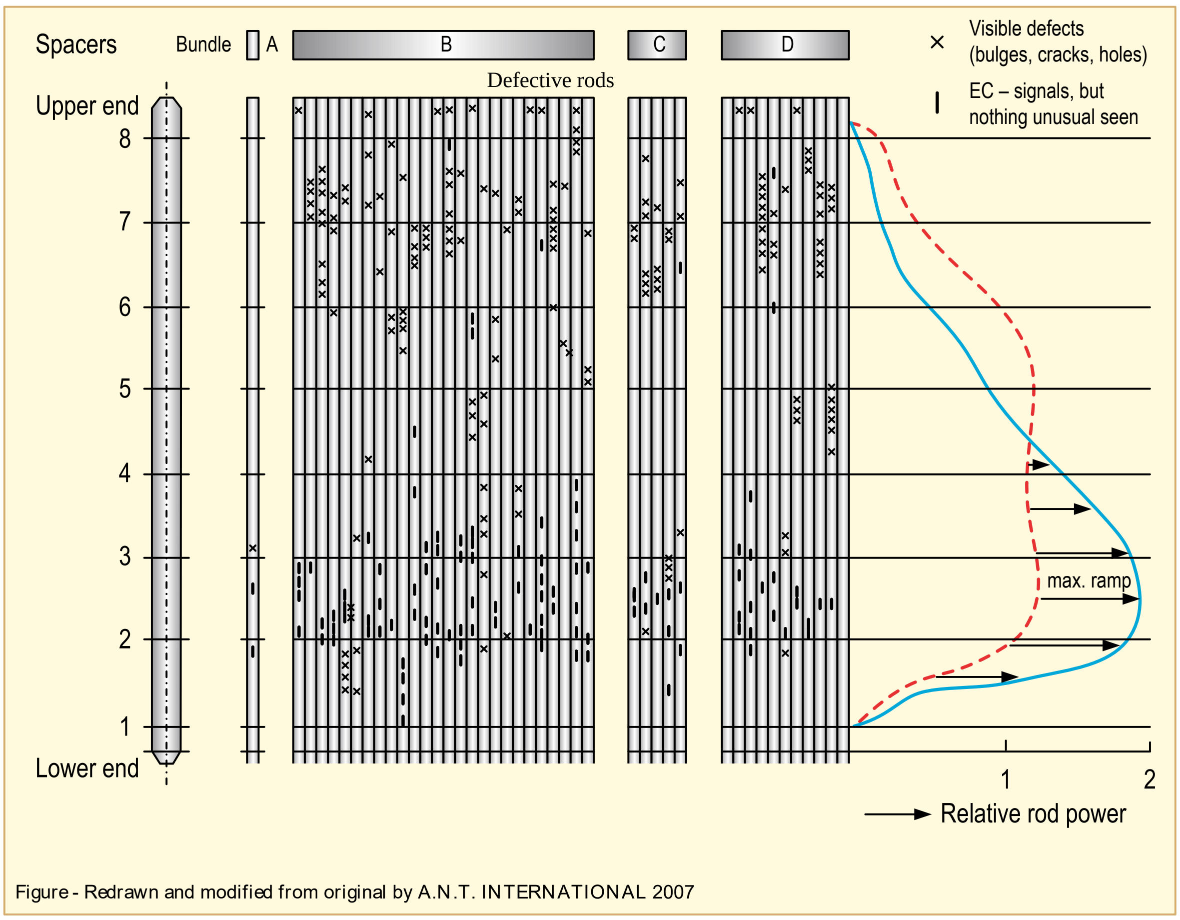

Figure 6-1: Correlation between power ramp and the location of defects (KernKraftwerk Obrigheim (*KWO*) cycle 6), modified according to [Garzarolli et al., 1978].

Manufacturing defects may facilitate *PCI* cracking. Quite some time ago, small defects or in homogeneities were produced during a non-optimised manufacturing process of the zirconium material tubing resulted in locally weaker cladding areas in a fuel rod. It has been reported for one fuel vendor that fuel failures have occurred due to such weak points, reportedly leading to axial cracks, [Potts & Proebstle, 1994]. Appropriate inspection techniques during manufacturing, especially by the use of ultra-sonic investigation should make the probability for incipient cladding cracks low.

More recently, pellet defects such as "pellet missing surface" have resulted in *PCI* cracking both in *BWR*s and *PWR*s, Figure 6-2 and, Figure 6-3. Pellet "chipping" has also contributed to *PCI* failures, Figure 6-4. In both cases the defects raise the cladding stresses significantly and thereby increase the tendency for *PCI* failure. Appropriate manufacturing Quality Assurance (*QA*)/ Quality Control (*QC*) methods for the pellets should mitigate this type of failures.

Figure 6-2: Missing pellet surface with fully penetrated crack in the cladding, modified according to [Groeschel et al., 2002].

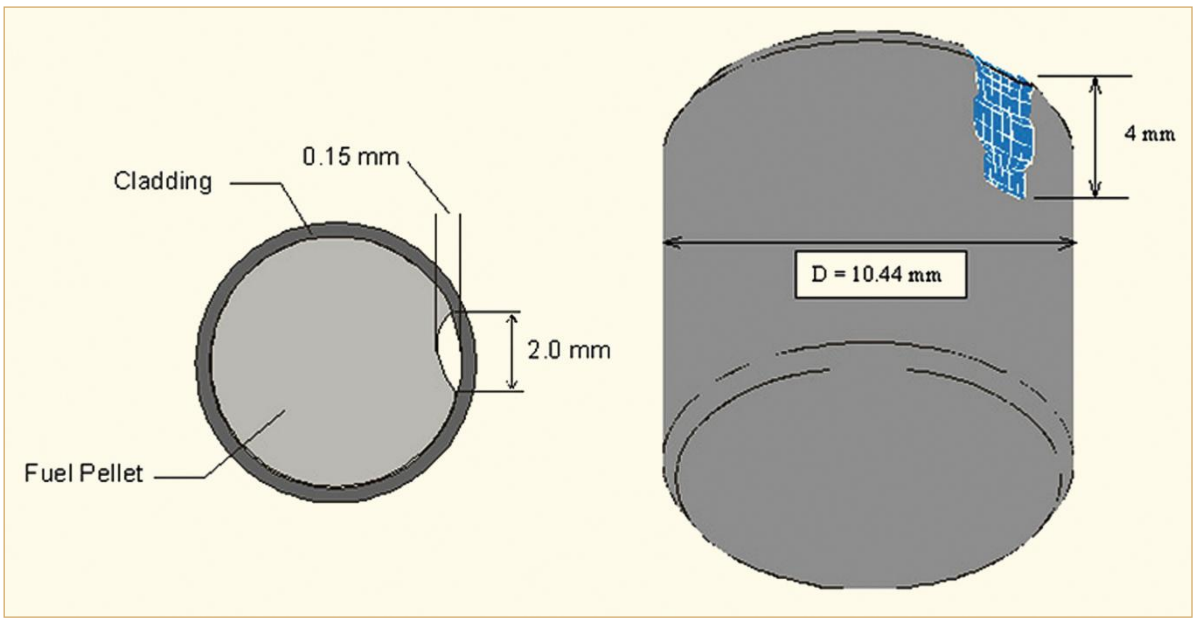

Figure 6-3: Assumed pellet geometry for the FREY code calculations, modified according to [Groeschel & Yagnik, 2001].

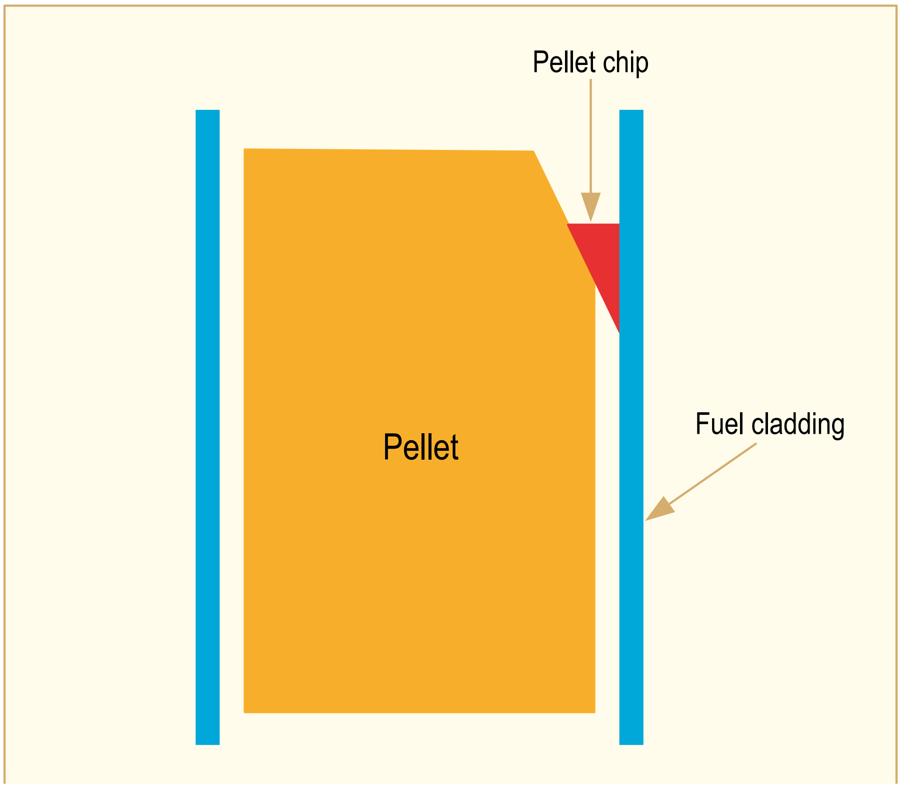

Figure 6-4: Schematics showing how a pellet chip may get stuck at the pellet-cladding gap.

More detailed information about *PCI* and *PCMI* is covered in "The Pellet-Cladding Interaction (*PCI* and *PCMI*)" *ZIRAT*11/*IZNA*6 Reports [Adamson et al., 2006/2007].

6.2.2.2 Information collected before outage

During operation - The timing of a failure and the nature of its activity release rate give a hint as to whether the primary failure is due to *PCI*. Such failures are always preceded by an increase in local power in a region that has operated for an extended interval at a lower power. Since the *PCI* defect is very small - it is only a hairline crack– very little activity will be released by the failed rod unless it degrades. The released fission products typically consist of the longer-lived noble gases with essentially no iodine or caesium during steady state operation. In some cases a *PCI* crack may reseal during operation due to the oxidation of the crack surfaces. Such a rod should still be considered as failed and should therefore be removed from the core.

Location of failures in core – *PCI* failures result from increases in local power and can be recognized by increases in the activity release rate of gaseous and sometimes soluble fission products within minutes to a few days of the power increase. As a result, the possible locations of the leaking assemblies can generally be determined from operating and chemistry records. Depending on the timing of the failures relative to the operating cycle, power suppression testing can also be used to identify the cell or *FA* with the leaking rod(s) and develop a plan for managing the leaking fuel.

6.2.2.3 Information collected during poolside fuel examinations

Sipping/UT during outage - If the failed assembly or rod(-s) identified during the sipping/*UT* campaign were:

- Located in a core position subjected to a significant power ramp at the time the assembly or rod(-s) failed and,

- in the intermediate to high burnup range

The failure mechanism is likely to be *PCI*. But, additional characterization is needed to make such a determination.

Visual examination - The *PCI* defect is initially only a hairline crack that is very difficult to detect by visual examination of the fuel rod outer surface. Although the ends of *PCI* cracks are sometimes decorated with zones of plastic deformation (X-marks), locating the actual fracture site frequently requires a combination of visual and eddy-current examinations. Another indication of a *PCI* failure is the formation of secondary hydrides at locations away from the primary failure site. As discussed in Section 3 failed rods always develop secondary hydrides as a result of water entering an operating rod after failure. Since the hydrides have a larger specific volume compared to that of the zirconium alloys, most blisters can be observed during visual examinations as cladding bulges, see Figure 6-5. The usefulness of hydride bulges in the assessment of failure mechanisms depends, however, on the length of time a rod operates after failure, the operating duty the rod experienced after failure and the extent of secondary damage. Features of potential interest can be obscured by post-failure oxidation, hydriding and related secondary damage.

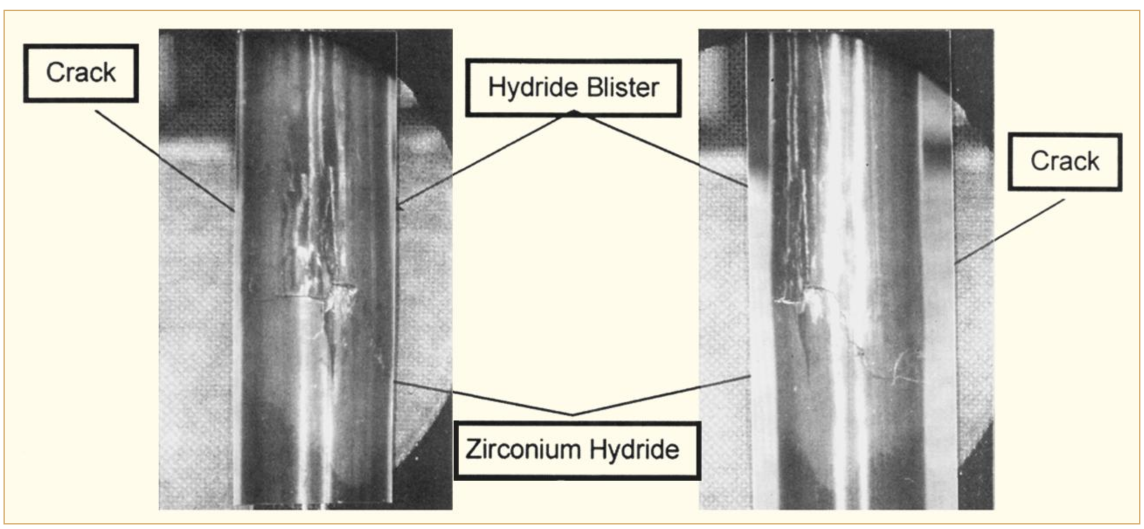

Figure 6-5: Failed rod with a secondary hydride blister, modified according to [Kim et al., 2000].

Eddy Current (EC) measurements – After identifying the rod(-s) that is/are believed to have failed by *PCI*, the next step is to use *EC* techniques to hopefully detect incipient defects in non-failed symmetry rod(-s) (rods with design characteristics and power ramp histories similar to those of the failed rod(-s)). Note that *PCI* cracks originate from the inside surface of fuel cladding, where the ability of conventional *EC* equipment to detect small, tight cracks is limited. So, *EC* examinations of rods suspected of *PCI* failures are useful in locating the actual failure sites and in identifying large cladding defects that might have contributed to the failures.

Fuel rod profilometry – Measurements of fuel rod diameter as well as the thickness of *CRUD* and oxide are an important part of determining the presence of *PCI* failures. Diameter and thickness profiles in the form of helical scans or axial scans at multiple azimuthal orientations are frequently used in such investigations. Measurements of *CRUD* and oxide thickness combined with visual observations provide a basis for determining the potential involvement of a corrosion or *CRUD*-related failure process. The

7 Summary

Fuel reliability has improved in all types of water-cooled reactors so that fewer fuel rods fail by developing leaks during operation, failures are being caused by fewer damage mechanisms and failures are taking place in fewer reactor operating cycles. Nevertheless, incentives exist to detect the occurrence of fuel failure and to mitigate both the near and long-term effects of fuel failure. On-line measurements of the radioactivity of nuclides and fissile materials released from leaking fuel rods to the reactor coolant and offgas system are used for both purposes. These data are supplemented by poolside examinations and, when necessary, by hot cell examinations.

The detection of a new failure is typically indicated by spike increases in a long-lived gaseous fission product such as ^{133}Xe (5.25 d) in plants that monitor gas activities or in spike increases of a long-lived soluble fission product such as ^{131}I (8.04 d) in plants that monitor only coolant activity. Success in detecting fuel failure is also reported with the use of an on-line spectrometer set to detect the presence of helium from the rod intraspace in the offgas stream. Changes in the ratio of long and short-lived nuclides are also useful in the detection of failure and in assessments of the progression of failure, particularly of small leaks; e.g., the ratio of $^{133}Xe/^{138}Xe$ or $^{131}I/^{134}I$ activities. The magnitude of ratios such as $^{133}Xe/^{138}Xe$ relative to the activity of the long-lived term (^{138}Xe) is used in some analyses to define the presence or absence of leaking fuel.

Fission gases are typically released first after fuel failure, with the release of soluble fission products following after enlargement of the leakage path. However, the presence of a failed rod with a small leakage path is sometimes indicated by spike increases in the activity of ^{133}Xe or ^{131}I during or after a reactor reactor transient involving changes in power or coolant pressure.

The onset of failure, the condition of the leaking rods and, to a limited extent, the number of leaking rods can also be inferred from changes in the regression coefficients of phenomenological models. In one type of model, fission product release is evaluated in terms of the relative activity release rates attributed to the assumed, underlying mechanisms; i.e., recoil, equilibrium or diffusion. In a second type, the release of fission products is characterized by means of empirical coefficients attributed to release from the fuel material to the rod intraspace, to escape from rod intraspace to the reactor coolant and to tramp uranium. The resulting values and applications tend to be specific to the respective models.

The consequences of cladding degradation after the initial failure (secondary degradation) are usually more severe than the failure itself because of the dispersal of fuel material into the primary coolant system due to the corrosion and erosion of fuel pellets exposed directly to the coolant. Radioactivity increases in the primary coolant system due to fuel failure and, to a greater extent, due to enlargement of the leakage path as a result of secondary degradation, which allows dispersal and deposition of fuel material in the core and related systems. The dispersed material contributes to system activity while the leaking fuel is operating and after the fuel has been discharged due to the residual material (tramp uranium) that remains in the core or that returns to the core from components of the primary system; e.g., steam generators in PWRs.

When leaking fuel is detected in BWRs, power suppression (or increase) testing is frequently performed to identify the location of the failed fuel. This testing involves the sequential movement of control blades to vary power in the respective control cells while measuring offgas or offgas and coolant activities. Increases in activity following a blade move indicates the leaking fuel is in a region where power is affected by the control blade. In many instances, the specific leaking fuel assembly can be determined by activity gradients around a control cell. When the leaking cell or fuel assembly is located, one or more control blades can be inserted to protect the leaking fuel from secondary degradation due to power changes during subsequent operation. Note that the need for such action has been reduced with the introduction of corrosion-resistant (lightly alloyed) Zr-liners in BWR fuel cladding. Note also that the risk of secondary degradation is lower in a PWR due to differences in coolant chemistry and power control, so the proactive management of leaking fuel practiced in BWRs is not required in PWRs or VVERs.

A common issue in the analyses of coolant or offgas activity is the effect of tramp uranium. Activity released by distributed fissile material can be a problem with respect to radiation exposure but is also a problem because it obscures activity released directly from failed rods. Methods have been developed to identify the relative contributions of tramp sources and leaking fuel rods and are summarized in Table 3-3.

Additional information such as the burnup of a leaking fuel rod can be approximated by the ratio of short and long-lived isotopes of cesium; e.g., ^{134}Cs(2.07 Y)/^{137}Cs(30.17 y). An example is shown in Figure 3-8. However, measurement uncertainty combined with decreasing changes in this ratio with increasing burnup frequently restrict the use of such ratios to the identification of fuel in its first or second cycle (i.e., at low-to-moderate burnup) versus its third or later cycle (high burnup).

Fuel reliability specialists tend to have preferred radiochemical methods for detecting and assessing fuel performance. However, no single method gives unambiguous results on a consistent basis. Uncertainty associated with the interpretation of measurement data is generally reduced by applying a combination of empirical and phenomenological methods.

If one or more leaks are indicated by on-line measurements, then sipping is generally used at end of cycle to identify the failed fuel assemblies and to verify that only non-leaking fuel is loaded back into the core. The sipping process involves the sampling and analysis of coolant activity of samples that can be associated with specific fuel assemblies. The methods used for sipping vary among utilities and reactors, but include sampling hoods placed over groups of bundles in a reactor core, sampling probes on fuel-handling masts and sipping chambers both in a reactor core and in fuel handling pools. Similar techniques can also be used to assess the hermeticity of spent fuel prior to placement in dry storage containers.

The identification of leaking fuel rods in an assembly that has been identified as failed by in-core monitoring and post-operation sipping requires a number of poolside inspection techniques. The attenuation of ultrasonic waves in fuel rod cladding by water in the rod intraspace is a common and effective method for identifying leaking fuel rods. Such ultrasonic tests can be performed while the fuel rods are in their respective assemblies. Sipping of individual fuel rods or groups of rods is also used to identify failed fuel rods. The location of leakage paths through fuel cladding and other damage (secondary hydriding) is identified by eddy current inspections and visual examinations. Visual examinations with image capture by radiation-resistant, high-resolution closed circuit video cameras is typically required to characterize cladding perforations and either establish the most likely cause of failure or contribute to decisions regarding hot cell examinations.

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Appendix A - Primary failures and degradation

As indicated in the preceding section, recent fuel rod failures have been caused by grid-to-rod fretting, debris fretting, fabrication defects and sporadic instances of crud or corrosion and PCI. The initial leakage paths resulting from these failure mechanisms provide varying degrees of resistance to the release of radionuclides from the affected rod to the reactor coolant system. These leakage paths can change (almost always increase in effective size) during post-failure operation, particularly due to the potential effects of secondary degradation. However, the characteristics of these leakage paths and their evolution during operation enable the use of radiochemical analyses for the detection of fuel failure(s) and the estimation of conditions such as the number of leaking fuel rods, their approximate power and burnup, and the state of the leakage path. The methods for assessing fuel performance by means of radiochemical analyses are reviewed in Section 3. The characteristics of fuel failures are discussed below as background for the review of radiochemical analyses.

A.1 Primary Failure Causes

A.1.1 Fretting

A.1.1.1 Introduction

The perforation of fuel cladding by fretting wear is the leading recurrent cause of failures in water reactor fuel rods. This section discusses the mechanism of fretting, provides examples of fretting failures and reviews remedies that are being developed to mitigate fuel failures due to this process.

Both GTR and debris fretting involve the removal of material from surface(s) undergoing relative motion while in contact. The relative motion is a system response which usually involves a sliding contact. Fretting comes from wear processes that depend on a large number of factors related to the contacting materials, the local environmental conditions, the nature and frequency of the relative motion, interfacial forces and contact stresses. Fretting wear in water reactors typically involves relative displacements of small amplitude (few μm to a few mm) that are oscillatory and generally include a tangential component of motion in an oxidizing environment; i.e., an environment conducive to the corrosion of Zr-alloys. The actual metal loss that leads to fuel failure results from a combination of corrosion and wear. Failure takes place when the cumulative loss of metal due to repetitive cycles of corrosion and wear penetrates the cladding wall.

A.1.1.2 Grid to Rod (GTR) Fretting

The principal cause of GTR fretting failure is fuel rod motion relative to its grid supports or springs. Such sliding can result from grid design features, fabrication defects, handling damage, stress relaxation and creep of grid supports or springs or from combinations of these factors in conjunction with flow-induced assembly or rod vibration. That is, the manufacturing defects discussed above can contribute to GTR fretting, but failures involve conditions that begin with design and extend through in-reactor operation.

The grid structure shown in Figure A-10 involves discrete points of contact between each fuel rod and the spring and fixed supports (dimples) in its respective supporting grid. These springs and supports produce contact stress in both the fuel rod and grid components due to the behaviour of the fuel rods, grids and assembly as an integral structure. The contact conditions, which are shown schematically in Figure A-1 and Figure A-2, vary with grid design, but nominally involve geometries such as crossed cylinders, a cylinder against a sphere or a cylinder against flat plate. For a given cladding diameter, contact stresses increase as the radius of curvature of the contacting part decreases. Contact conditions also involve the effects of material dissimilarities, surface roughness and asperity contact, in-reactor corrosion, stress relaxation, creep and physical excitation with random and periodic components. Contact conditions vary spatially within fuel assemblies and can also vary with burnup and power over the course of in-reactor operation.

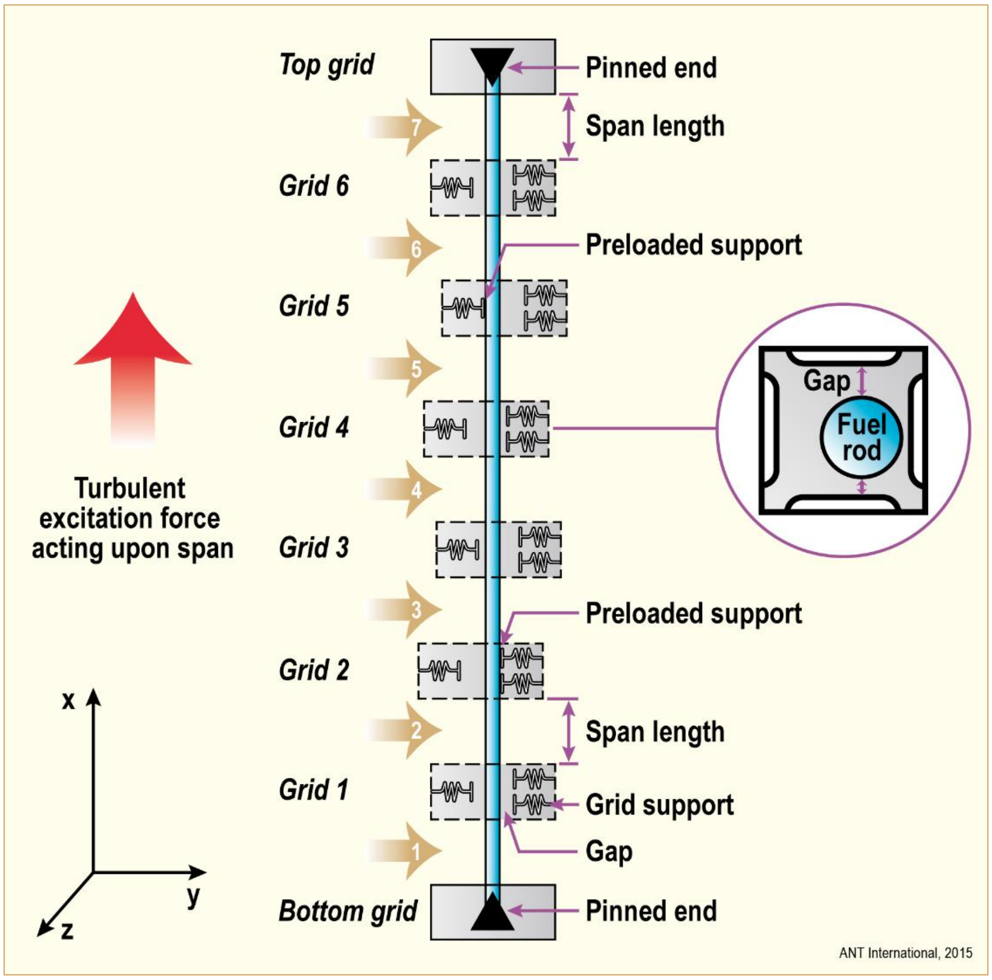

Figure A-1: Fuel rod and grid schematic with contacts at spring and fixed supports, after [Lu et al., 2013].

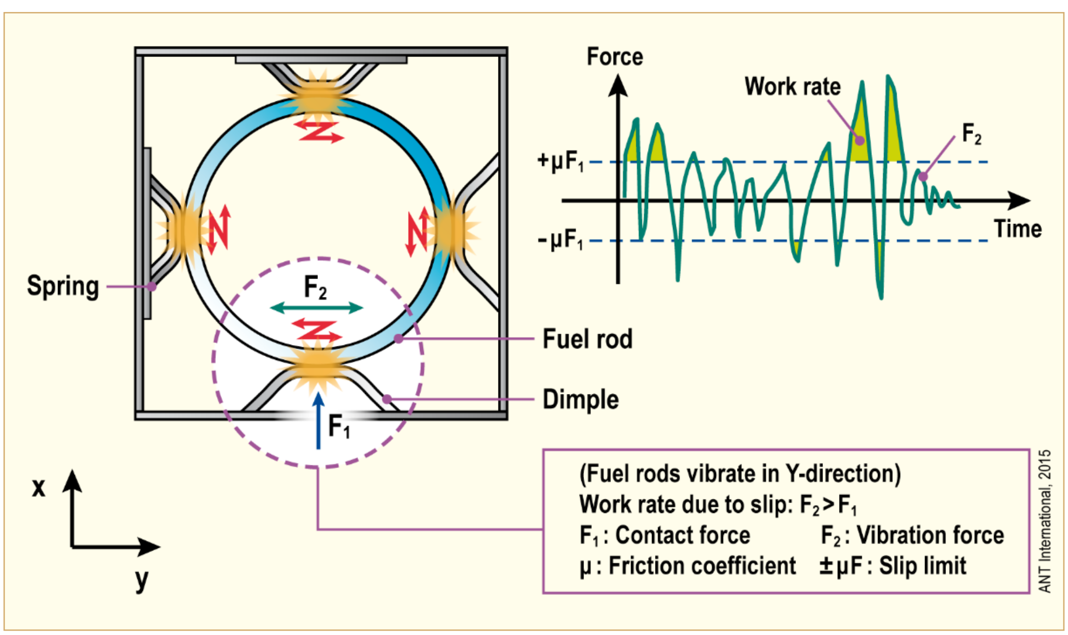

Figure A-2: Schematic of cladding wear points in spacer grid, modified according to [Yasuno & Kitashiba, 2014].

Grid design involves a number of conflicting performance requirements. The contact area between fuel rods and springs and fixed supports needs to be small to minimize the effects on coolant flow and heat transfer. Conversely, large areas are needed to minimize contact stresses.

The vibration of fuel rods relative to their supporting grids is induced by coolant forces and by vibration of the fuel assemblies. Such vibration is being investigated experimentally in instrumented flow loops and analytically by means of coupled fluid-structure models; e.g., [Lu et al., 2013]. Fuel rod vibration is generally treated as a random process, with displacements, forces or accelerations expressed in terms of average value over a frequency range (window) relative to frequency; i.e., the power spectral density (PSD) of hydraulic forces. As shown schematically in Figure A-3, this conversion is based on the Fourier transform of the random flow forces and leads to the forces and power spectral density distributions typical of Figure A-4 and Figure A-5.

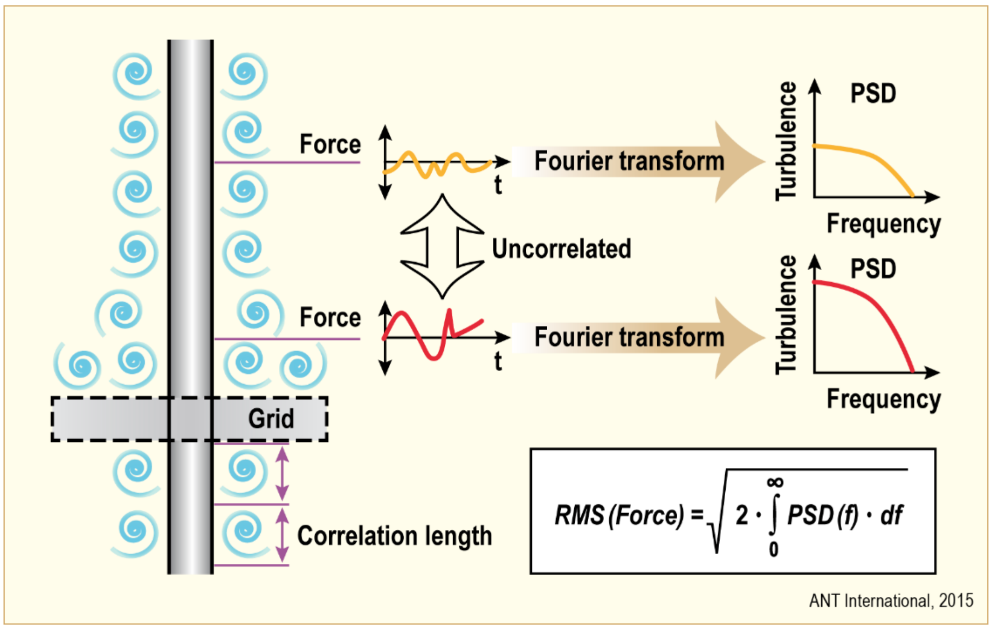

Figure A-3: Turbulence forces due to coolant flow and mixing vanes, modified according to [Karoutas et al., 2012].

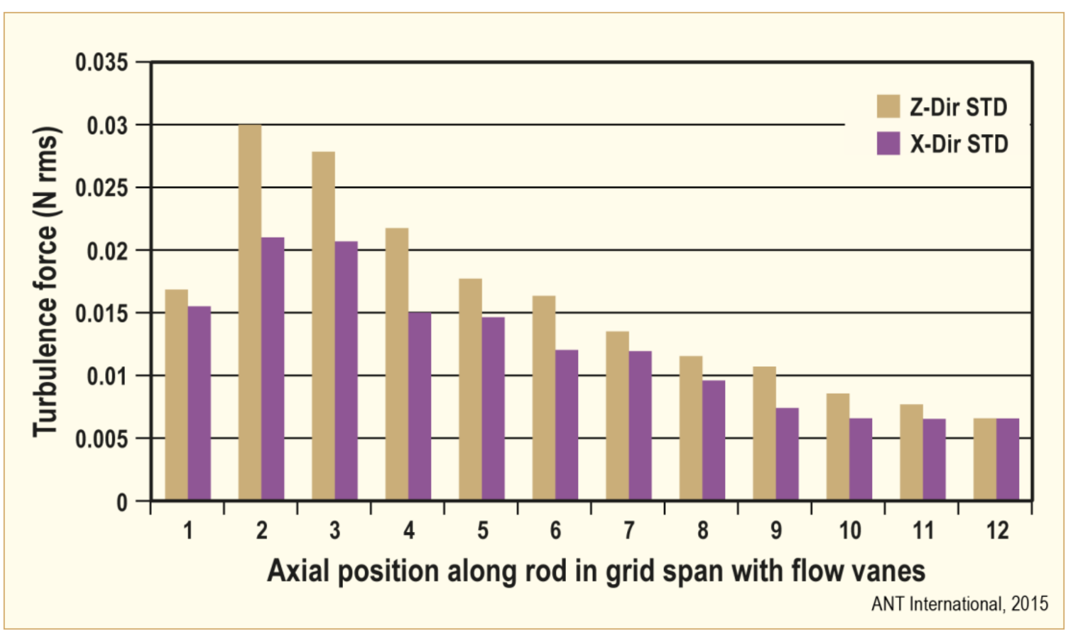

Figure A-4: Distribution of excitation forces along grid span with flow mixing vane, modified according to [Lu et al., 2013].

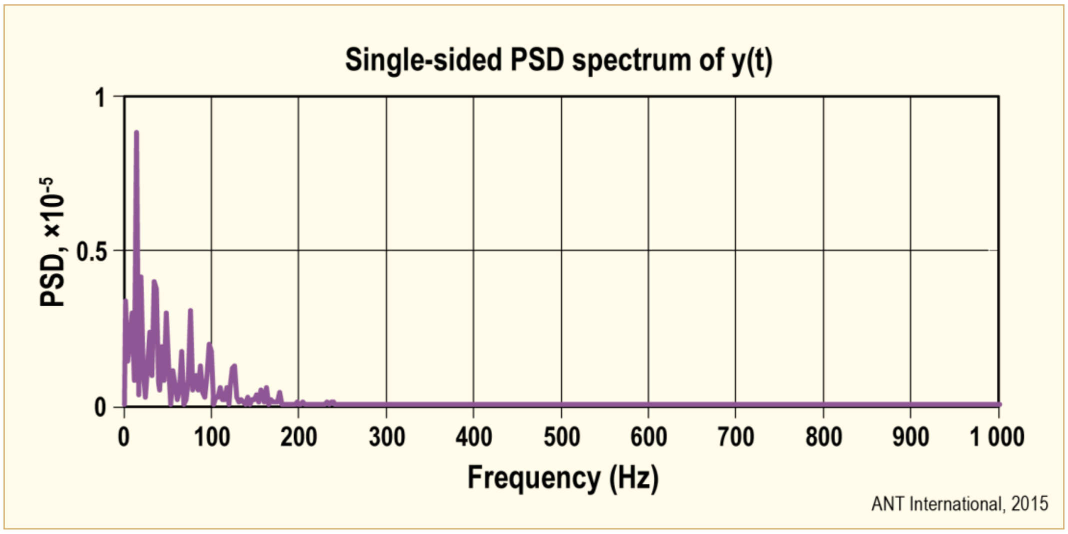

Figure A-5: Power spectral density (PSD) of flow forces on a fuel rod within a single span of support grids, modified according to [Lu et al., 2013].

The random excitation of fuel rods and assemblies produces repetitive, small amplitude motion between fuel cladding and the spacer contacts. This motion depends on the restraining force of the grid spring when contact exists with the fuel cladding and on gaps that develop during operation due to irradiation-induced, spring relaxation and inward cladding creep. An example of gaps that are predicted to develop during operation is given in Figure A-6. Gaps begin to form during the first cycle of operation and increase with time. With a core loading pattern involving interior (high power) positions during the first two cycles and an exterior (low power position) in the third cycle, gaps are predicted to reach 70–75 μm (2.8–3.0 mils) during the last operating cycle.

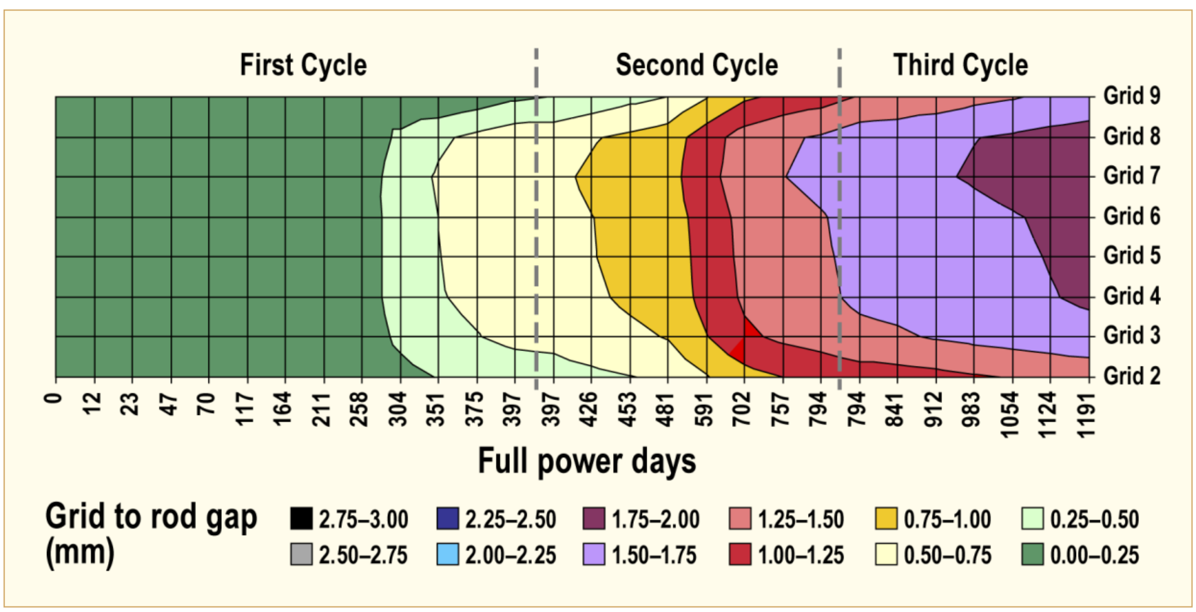

Figure A-6: Calculated grid-to-rod gaps relative to operating time and core elevation, modified according to [Karoutas et al., 2012].

As noted earlier, fretting removes cladding metal by the processes erosion and corrosion. The interfacial forces appear to be too small for metal loss by the normal process of wear. Instead, contact and relative motion between the grids and fuel rod is observed to remove the oxide film that forms on the surface of Zr-alloy cladding. The fresh, exposed cladding surface then re-oxidizes rapidly as shown in Figure A-7. The surface oxide reforms and is removed in a repetitive process which breaches the cladding wall.

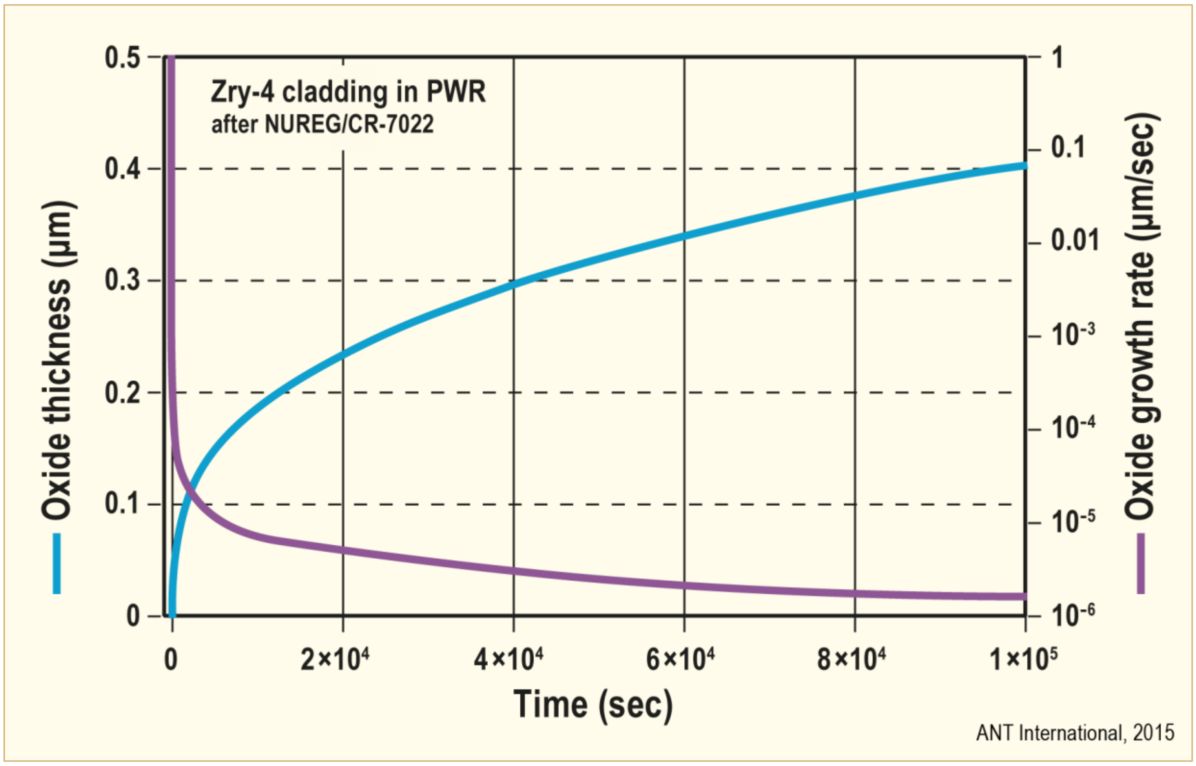

Figure A-7: Zircaloy cladding oxide thickness and corrosion rate relative to time; after models of [Geelhood et al., 2011].

Fretting wear depends on a large number of factors involving the contacting materials, the local environmental conditions, the nature and frequency of the contact motion, interfacial forces and contact stresses. Fretting wear in water reactors results from oscillatory, sliding or sliding and impinging displacements at frequencies less than about 200 Hz [Lu et al., 2013] of small-amplitude amplitude displacements; i.e., fewµm to a few mm. As noted above, an oxidizing environment is required for such contact to produce the metal loss that leads to fuel failure due to fretting. Failure takes place when the cumulative metal loss due to repetitive cycles of abrasion and corrosion penetrates the cladding wall.

The same fundamental mechanism is involved in grid-to-rod fretting (GTRF) and debris-induced fretting. The scar produced by fretting depends on the shape of the contacting material and the direction or directions of relative motion. An example of GTRF is shown in Figure A-8 in which the metal loss corresponds to sliding between the fuel cladding and the grid spring and supporting projections (dimples). Note that the wear in this example was produced in an ex-reactor test rig which induced vibrations normal to an assembly consisting of Zry-4 grids and Zry-4 clad fuel rod. The example illustrates that fretting can result from contact between similar metals. Examples of GTRF between dissimilar metals (Zr-alloy cladding and stainless steel grids) during reactor operation are shown in Figure A-9.

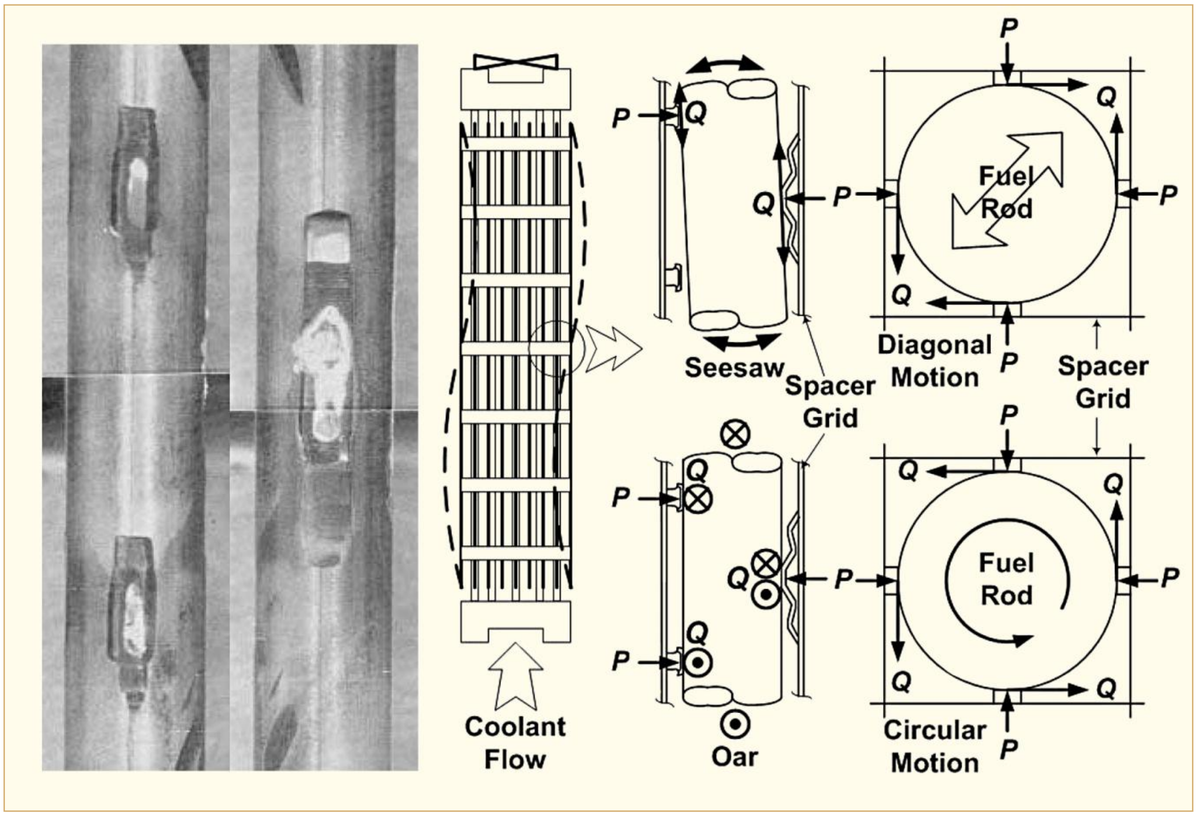

Figure A-8: Fretting of fuel rods with schematic diagrams of the grid-to-rod interaction that contributed to the wear, modified according to [Kim & Lee., 2007].

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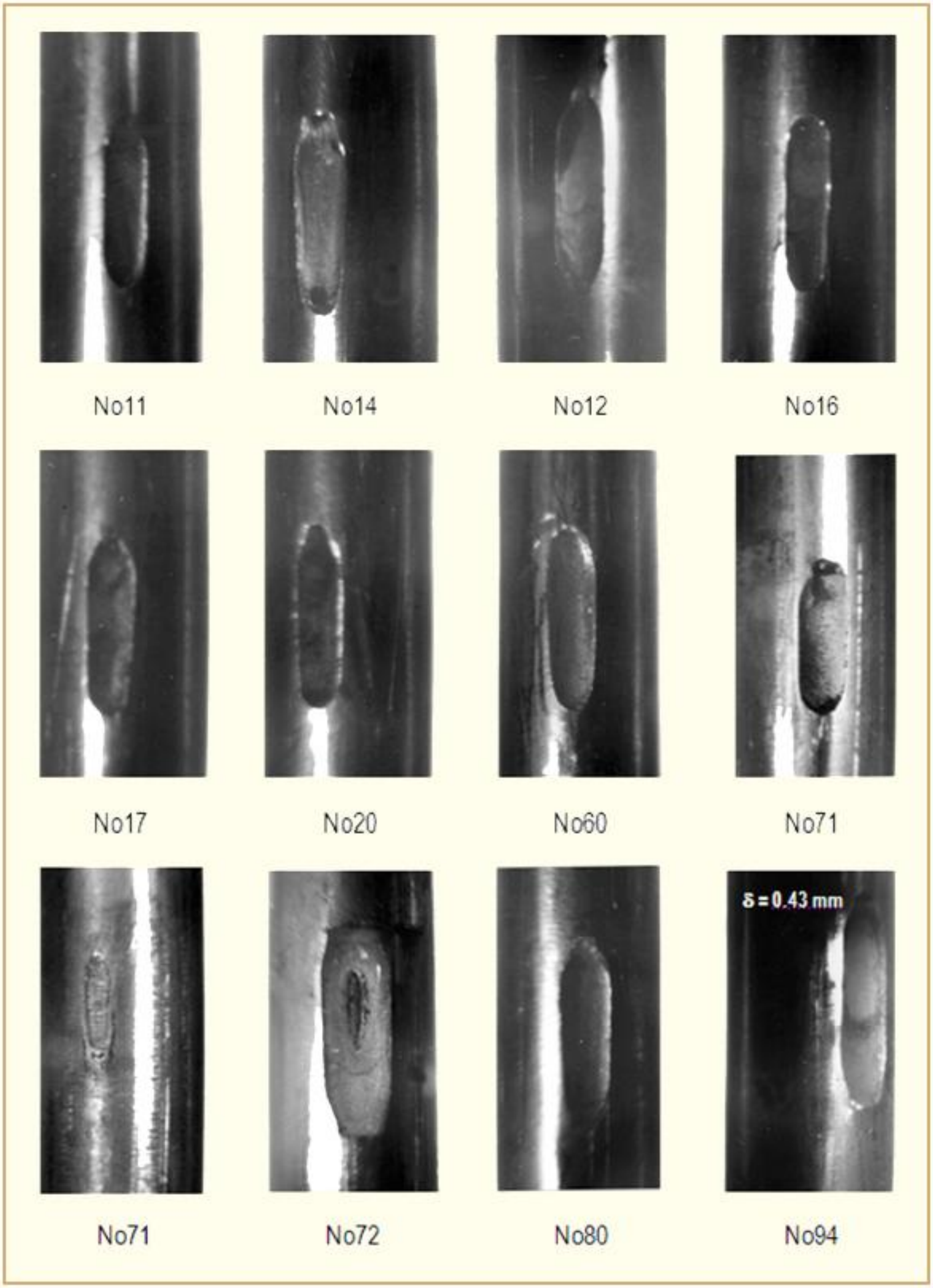

Figure A-9: Visual appearance of cladding wear scars due to grid-rod fretting at the places of contact with 1st and 2nd spacer grids in VVER 440 fuel, modified according to [Smirnov et al., 2002].

Efforts to reduce or eliminate incidence of GTR fretting vary among fuel suppliers and have varied over time for each of the suppliers. Current work includes understanding the flow forces in PWR cores and in the flow channels among fuel rods, particularly with regard to the effects of the grid flow vanes now being used to improve thermal-hydraulic capabilities; e.g., modelling of fluid-structure interactions by the Consortium for Advanced Simulation of LWRs (CASL)[14]. Work also includes improved spacer designs, materials and fabrication practices; e.g.,

- Relaxation resistant springs (age-hardening Ni-based alloys),
- Improved spring and contact designs with larger bearing area, lower contact stress and increased resistance to rod motion,
- Spacer band and vane designs to minimize handling damage,
- Reduced turbulence and cross flow among and within fuel assemblies.

A.1.1.2.1 Grid to rod (GTR) fretting resulting from poor spacer manufacturing

Grid-rod fretting is one of the major failure causes in PWRs. It is due to rod/assembly vibrations induced by turbulence and flow inhomogeneities, which are always present, particularly in the bottom nozzle inlet range up through to the bottom grid. Failure occurs by a repetitive process of erosion and corrosion if either the turbulence is higher than anticipated in design, or the fuel rod support in the spacer grid is not according to specification. Normally, grid-rod fretting failures occur by fretting of the rods against the spring and dimple of the spacer grid, Figure A-10. In severe cases when the springs are strongly deformed or damaged, fretting can also be observed against spacer strip and vanes. Grid-rod fretting failures can also occur occasionally as a result of damaged spacers - the result of interference between the adjacent PWR fuel assemblies during fuel shuffling.

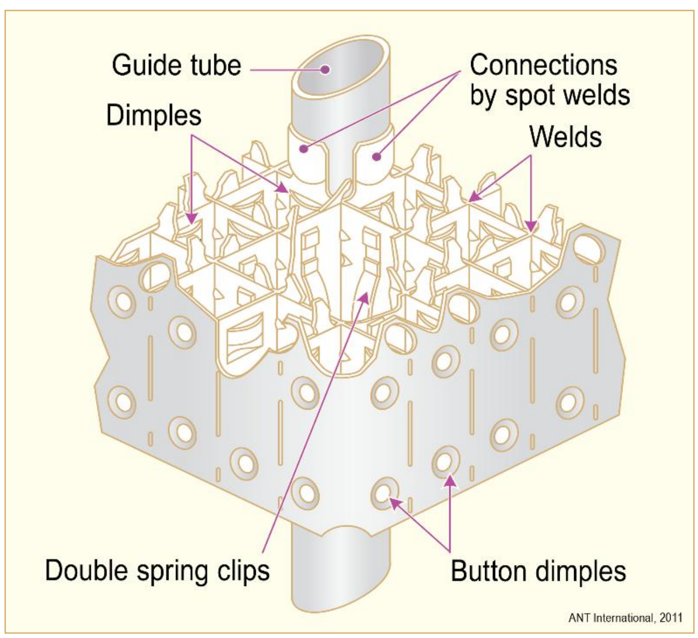

Figure A-10: Example of a spacer grid (this is the EDF AFA 2 G design), showing the location of the “dimples” and the springs (named “double spring clips” in the sketch above. In this design the sheet material, including the “dimples” is Zry-4, while the spring consists of 718 Inconel materials.

[14] Information on CASL can be accessed by means of the Internet link http://www.casl.gov/.

The problem in design and fabrication is to make sure that all spring forces are high enough to provide adequate rod support, not only after grid fabrication but also after rod assembling and transportation to the plant, and leave adequate margin for irradiation induced relaxation[15].

Effects of deviations

- Fretting of the cladding (GTR) by the spacer spring or hard stop due to inadequate spring force or off-specification cell size that can result in eventual clad penetration.
- For springs of nickel base alloys, the heat treatment is important to provide resistance to IASCC.

A.1.1.3 Debris Fretting

Debris fretting is occurring when there is a mechanical interaction between debris and the cladding. Loose metallic debris can circulate in the reactor system and become trapped in the fuel assembly. The debris causing the fretting could either be already present in the assembly or borne to the assembly by the coolant from other sources. Typically, the debris are trapped between fuel rods and spacer, usually the bottom one in PWRs since the largest turbulence is generated at this elevation due to the impact of the bottom nozzle to the coolant inlet flow. The debris particle will vibrate under the influence of the coolant flow and cause fretting holes within a few months in some cases. Unlike other defect types, fretting defects are not power related.

The observed debris have typically included turnings and shavings as those from metal working operations performed during primary system maintenance. In a few isolated cases, large objects such as tools, screws, bolts, nuts, metal clips, electrical connectors, pieces of lock wire and other wires, shavings from defective pumps or valves, parts of gaskets and saw blades have been found in damaged fuel assemblies. Since Zircaloy, the cladding material used in all BWRs, is a relative soft material, threads, shavings etc. of harder metals, such as steel and stainless steel, are often suspected to produce the debris fretting.

The actual fretting debris is generally not found in, or adjacent to, the fretting hole produced. The whole shape is normally instead used to deduce the debris fretting as the primary failure cause. There should, nevertheless, always be a debris-fretting hole found to deduce debris fretting as a primary failure cause by definition. This is because the fretting hole should have an equally large, but generally larger, aspect at the cladding outer wall in relation to the penetration at the inner wall. There are other primary failure mechanisms that may produce small primary failures (mainly because they grow from the inside), sometimes impossible to find by visual inspection of the cladding surface. If the crud layer is thick, or the cladding oxide layer thick and spalling, debris-fretting holes can be hard to detect.

The debris-fretting hole is difficult to detect for all rods remaining in an assembly. Even for the peripheral rods, there is theoretically slightly more than 50% chance to find the primary debris fretting failure, unless the rods can be rotated, even by careful visual observation. In reality, the detection probability by visual inspection methods is much less, since the debris is preferentially caught, and causing debris fretting failures, adjacent to the spacers, due to the higher probability for retaining the debris in the more confined spacer structure. For inner rods, depending on the fuel design, the probability to visually detect a debris primary failure is even much lower, due to the surrounding rods hindering optical observations.

Experience has shown that the probability for debris fretting primary failures varies considerably between plants and also between time periods for a specific plant. Fretting takes place with repetitive contact of the cladding by debris. The frequency of contact varies approximately with flow velocity and is estimated to be in the order of several kHz; i.e., higher vibration and contact frequencies with the higher flow velocities, [Ryttersson et al., 2007] and [Baranowski, 2014]. The contact areas and forces due to debris are ill-defined, but are believed to be small base on the size of debris that has been recovered from some affected fuel assemblies and, more commonly, from the size of the resulting wear

[15] That always will reduce the spring force

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List of Abbreviations

ABB	ASEA Brown Bovari
ABWR	Advanced Boiling Water Reactor
ALARA	As Low As Reasonably Achievable
ALCYONE	Name of computer code for fuel rod behavior used in France
ANF	Advanced Nuclear Fuels GmgH (now part of Framatome)
ANSI	American National Standards Institute
AOOs	Anticipated Operating Occurrences
BF2	Browns Ferry 2 BWR
BWRs	Boiling Water Reactors
CANDU	Canadian Deuterium Uranium
CILC	Crud Induced Localized Corrosion
DCP	Distinctive Crud Pattern
DOE	Department of Energy
EC	Eddy Current
ESSC	Enhanced Spacer Shadow Corrosion
FGR	Fission Gas Release
FPs	Fission Products
FRI	Fuel Reliability Indicators
GE	General Electric
GNF	Global Nuclear Fuel
GTRF	Grid-To-Rod Fretting
HBS	High Burnup Structure
HWC	Hydrogen Water Chemistry
IAEA	International Atomic Energy Agency
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
ISG	Interim Staff Guidance
LEP	Lower End Plug
LHGR	Linear Heat Generation Rate
LME	Liquid Metal Embrittlement
MPS	Missing Pellet Surfaces

MTRs	Material Testing Reactors
NMCA	Noble Metal Chemistry
NPPs	Nuclear Power Plants
NRC	Nuclear Regulatory Commission
NWC	Normal Water Chemistry
OLNC	On-Line Noble Metal Chemistry
OLSS	ON-LINE Sipping System
PCI	Pellet Cladding Interaction
PHWR	Pressurised Heavy Water Reactor
RBMK	Reaktor BolshoiMozhnostiKanalov
RCS	Reactor Coolant System
RPZ	Reactor Pressurizer
SCC	Stress-Corrosion Cracking
SNF	Spent Nuclear Fuel
SRP	Standard Review Plan
SS	Stainless Steel
STP	Standard Temperature and Pressure
TIG	Tungsten-Inert-Gas
TMI-1	Three Mile Island 1
UEP	Upper End Plug
US NRC	United States Nuclear Regulatory Commission
UT	Ultrasonic Testing
VCT	Volume Control Tank
Visual_DETECT	Defective Element/Tramp Estimate Characterization Tool
VVER	VodaVodaEnergo Reactor
WANO	World Association of Nuclear Operators
WS	Water Samples
ZIRLO	Zirconium Low Oxidation

Unit conversion

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

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

MASS	
kg	lbs
0.454	**1**
1	2.20

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

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

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