Hot Cell Post-Irradiation Examination Techniques for Light Water Reactor Fuels

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Contents

1 Introduction

1.1 Background and Introduction

In order to handle radioactive materials, special facilities are required to shield the radiation emitted and to prevent contamination of the environment by radioactive materials released during handling and processing. In such laboratories the radioactivity levels are high; therefore, there are very strict requirements for shielded facilities with well controlled ventilation and remote handling devices. Several versions of shielded hot cells[1] and remote handling devices have been developed and used over the years. The evolution of different designs has reflected the growing emphasis on optimization of protection and safety.

Post Irradiation Examination (PIE) is the study of used nuclear materials such as fuel rods and rest of the fuel bundle hardware (water rods/guide tubes, spacers/grids, and channel boxes). It has several purposes. It is known that by examination of used fuel the failure modes that occur during normal use (and the manner in which the fuel will behave during an accident) can be studied. In addition information is gained which enables the users of fuel to assure themselves of its quality and it also assists in the development of new fuels. When fuel rod(s) fail in a Light Water Reactor (LWR) in-core and pool-side examinations are performed to locate the failure and determine the root cause of the failure. If these examinations are inconclusive the failed rods, along with some sound sibling rods, are shipped to a hot cell[1] facility for further examinations and tests for root-cause analysis.

The growing operational demands on nuclear fuel, such as longer fuel cycles, higher burnups, and use of transient regimes, call for more robust fuel designs and more radiation-resistant materials. Implementation of new materials and fuel designs that are able to meet these more challenging conditions requires adequate operational feedback and practical verification of models for prediction of fuel behaviour. Post-Irradiation Examinations (PIE) provide fuel vendors and nuclear utilities with data on how newly developed or established materials withstand normal operating conditions in new environments. Post-irradiation examinations are largely carried out at a hot cell laboratory where irradiated fuel rods and other bundle hardware can be received, handled, examined, and tested. The investigation results provide information for fuel improvement and, thereby, can potentially enhance operating efficiency and reliability.

Regulatory authorities, when issuing a permit to a utility to operate the core under higher duties, must have sufficient information on nuclear fuel behaviour under normal, transient and accident conditions. This is necessary to prove that fuel safety margins have not been affected. The status of development of destructive and non-destructive fuel examination methods has always guaranteed the quality and volume of required information. Status and development trends in the examination techniques of LWR fuel in different countries have been discussed in the proceedings of the IAEA technical co-operation meetings [IAEA, 1994], [IAEA, 2001], and [IAEA, 2009].

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[1] Shielded containments are commonly referred to as Hot Cells. The word "hot" being used as a synonym for radioactive. Hot cells are used in both the Nuclear and the Nuclear Medicines Industry. They are required to protect individuals from radioactive isotopes by providing a safe containment box in which they can control and manipulate the equipment required. Hot cells are used to inspect spent nuclear fuel rods and to work with other items that are high-energy gamma ray emitters.

As mentioned above, the demands on nuclear fuel have increased over the years. Accordingly, the demands on PIE techniques and respective results have also increased and their tasks have been changed. New materials and designs, including mixed oxide fuel, burnable absorber fuel and other additive fuels, together with corrosion resistant claddings, have become more prominent. PIE of lead test assemblies is complemented with high burnup test reactor experiments including refabricated (from irradiated commercial rods) fuel rods. Changes in composition, structure and properties of fuel and structural materials are to be investigated and understood to calculate, validate and forecast fuel operational margins and safety limits. Further development of advanced PIE techniques is demanded by justification and licensing of the above mentioned advanced fuels. Common approaches of PIE techniques allow comparing results obtained in different countries and different laboratories thus improving the trustworthiness of the results to be used for fuel performance evaluation and licensing.

There are several reasons why poolside and/or hot cell examinations are performed on fuel assembly components. The reasons include:

1) Root cause investigations of failed fuel assembly components that have degraded, or not.

   a) A “failed” fuel assembly component has a wider meaning in this respect. It does not only mean that the component has physically failed but it could also mean that the component does not behave satisfactory, e.g. fuel assembly bowing that is so large that control rods cannot be inserted.

2) Maintenance of good fuel reliability by:

   a) Providing base line data before a change in operational environment of the fuel.

   b) Getting early warnings of potential issues.

3) Fuel vendor design and licensing studies such as:

   a) Providing data to material models and fuel performance codes.

   b) Verification of the good performance of a new fuel design.

   c) Assessment of the effects of changes in the operating environment; e.g., water chemistry improvements or higher exposures.

The objectives of the hot cell examination of the fuel bundle hardware vary with the component being examined, e. g, hydrogen pick up and embrittlement of spacers and water rods, and irradiation and hydrogen induced bow of channel boxes. The hot cell examinations/testing include a number of tasks selected to address these objectives using available hot cell capabilities.

The hot cell facilities in the US and other countries are being reviewed and revived to meet today’s demand for fuel reliability and tomorrow’s demands for even higher burnup fuel. Fuel performance data are required to license fuel for higher burnup such as MOX for wide scale use in commercial reactors; and, possibly, to license fuel for a new generation of fast reactors. Additionally, fuel isotopic analysis and recycling technologies are critical factors in the goal to eventually close the fuel cycle. This focus on fuel reliability coupled with the renewed interest in recycling puts a major spotlight on existing hot cell capabilities and their ability to provide the baseline analysis to achieve a closed fuel cycle [DeCooman & Spellman, 2007].

This Special Topic Report (STR) provides an overview about the status of post-irradiation examination and testing techniques for nuclear fuel and their applications for analysis of materials degradation during fuel operation in a reactor core. Emphasis is given to advanced non-destructive and destructive PIE techniques applied to LWR fuel rods and bundle hardware. Section 1 of the report briefly describes typical LWR fuel assemblies, effects of neutron irradiation and reactor environment on fuel assembly materials, and information on specialized equipment and trained manpower needed for hot cell examinations. Section 2 reviews the on-site examinations performed on the fuel assembly components before shipping those to a hot cell facility. Section 3 gives details of the most commonly used non-destructive examination techniques used on the failed and sound sibling LWR fuel rods. Section 4 describes destructive examination techniques and tests used to determine the failure root-cause. Section 5 gives details of the tests used to evaluate the mechanical properties of the irradiated material. Section 6 is dedicated to PIE techniques used for examination and testing of the fuel bundle hardware such as spacers/grids, water rods/guide tubes, and channel boxes. Section 7 provides a list of relevant references.

Each hot cell facility that performs post irradiation examinations maintains Standard Operating Procedures (SOP) for each examination/test, and set of instructions for operation of each piece of equipment. Since details of these procedures and instructions may vary from facility to facility this STR does not provide step-by-step instructions.

1.2 BWR and PWR fuel assemblies

Since this report concentrates on PIE of LWR fuel rods and bundle hardware, it is important to provide here a brief description of typical LWR fuel assemblies. Additional details on this topic are available in Fuel Material Technology Report (FMTR) Vol. IV [Rudling & Patterson, 2009]. There is a wide variety of different types of fuel assemblies for Light Water Reactors as listed in Table 1-1 [Rudling & Patterson, 2009].

Table 1-1: Design parameters in water cooled reactors

Parameter		Western type PWR	VVER[1] (440/1000) MW	BWR
1.	Coolant	Pressurised H_2O	Pressurised H_2O	Boiling H_2O
2.	FA materials (pressure tube materials)	Zry-4, ZIRLO[2], DUPLEX, M5, MDA[3], NDA[4], Inconel, SS[5]	E110, E635	Zry-2, Zry-4, Inconel, *SS*
3.	Average power rating, (kW/l)	80-125	83/108	40-57
4.	Fast neutron flux, average, n/cm^2.s	6-9E13	5E13/7E13	4-7E13
5.	Temperature, °C Average coolant inlet Average coolant outlet Max cladding Outer Diameter (OD) Steam mass content, %	 279-294 313-329 320-350 	 267/290 298/320 335/352 	 272-278 280-300 285-305 7-14
6.	System pressure, bar	155-158	125/165	70
7.	Coolant flow, m/s	3-6[6]	3.5/6	2-5[6]
8.	Coolant chemistry Oxygen, ppb Hydrogen (D_2), ppm cc/kg Boron (as boric acid), ppm Li (as LiOH), ppm K (as KOH), ppm NH_3, ppm NaOH, ppm	 <0.05 2-4 25-50 0-2200 0.5-3.5 - 	 <0.1 30-60 0-1400 0.05-0.6 5-20 6-30 0.03-0.35	 200-400 <1.8[7] - - -

[1] Voda Voda Energo Reactor (Russian type PWR)
[2] Zirconium Low Oxidation
[3] Mitsubishi Developed Alloy
[4] New Developed Alloy
[5] Stainless Steel
[6] Variation from lower to upper part of the core and from plant to plant
[7] Dependant on whether hydrogen is being added to the feed water or not.

ANT International, 2009

The fuel rod array for BWRs was initially 7x7 but there has been a trend over the years to increase the number of fuel assembly rods. The driving force for this trend was to reduce the Linear Heat Generation Rate (LHGR), which resulted in a number of fuel performance benefits such as lower Fission Gas Release (FGR), and increased Pellet Cladding Interaction (PCI) margins.

Also for PWRs there has been a trend to greater subdivision of fuel rods, e.g. from Westinghouse 15x15 to 17x17 design; however, to accomplish this one had to go to a new reactor design because PWRs do not have the same flexibility with core internals and control rods as is the case for BWRs.

In most PWRs, the assemblies are positioned in the core by bottom and top fittings, and the lateral clearances are restricted by the assembly-to-assembly contacts at the spacer-grid levels. Furthermore, the control rods consist of Rod Cluster Control Assemblies (RCCAs), the poison part of which moves into guide thimbles (or guide tubes). These guide thimbles are an integral part of the assembly structure.

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

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

- Maintain proper positioning of the fuel rods under normal operating conditions and in design basis accidents (e.g. seismic effects, Loss of Coolant Accident (LOCA), Reactivity Initiated Accident (RIA)).

- Permit handling capability before and after irradiation.

Figure 1-1 and Figure 1-2 show typical BWR and PWR fuel assemblies, respectively. Also, the different fuel assembly components are shown and the material selections for these components are provided. The reason for the difference in structural material selection is that in general the most inexpensive material is chosen for a specific component that yields the lowest cost to produce the component while ensuring adequate performance during normal operation and accidents. More information about fuel designs, functions of the different fuel design components and material being used are provided in Fuel Material Technology Report (FMTR) Vol. 1 [Cox et al, 2006].

The materials used for the fuel assembly components are Zr alloys, Inconel[2] (precipitation hardened Inconel X-750, Inconel 718 and solution treated Inconel 625) and stainless steel (SS 304L or similar austenitic stainless steels). A low cobalt content is desired to keep the radiological exposure of workers (man-rem) low in the stainless steel and nickel-base alloys. However, only Zr alloys have low thermal neutron cross section. Thus, for components in the core mostly Zr alloys are used since the high thermal neutron flux would otherwise result in large loss in reactivity if other materials such as stainless steels and/or nickel-base alloys were used. Spring materials need to be made of materials with low stress relaxation rates, such as Inconel X-750 or Inconel 718. These Ni base alloys are generally heat treated to reach an optimum precipitation hardening. To lower the parasitic neutron absorption for grids/spacers, the strips are made of Zry-2 and –4, while the spring itself is made of either Inconel X-750 or Inconel 718 to ensure adequate fuel rod support during its entire irradiation. In some fuel designs also the top and bottom PWR grid is entirely made of Inconel X-750 or Inconel 718. This is possible since the neutron flux is much lower at the top and bottom part of the core resulting in a very small loss of thermal neutrons due to parasitic material absorption. The low neutron flux at the top and bottom part of the core is also the reason why the much cheaper material SS 304 L can be used instead of Zr alloys for components at these elevations. In newer BWR designs the spacers are made entirely of Inconel X-750 or 718, using the minimum thicknesses and minimum planar cross-section possible.

[2] The name "Inconel" is a registered trademark of the International Nickel Company, which is now the Special Metals Corporation, and refers to their material. Although nickel-based alloys are now produced by a number of suppliers, the name "Inconel" is used instead of a more general designation, such as "alloy X750", for consistency with historic industry practices

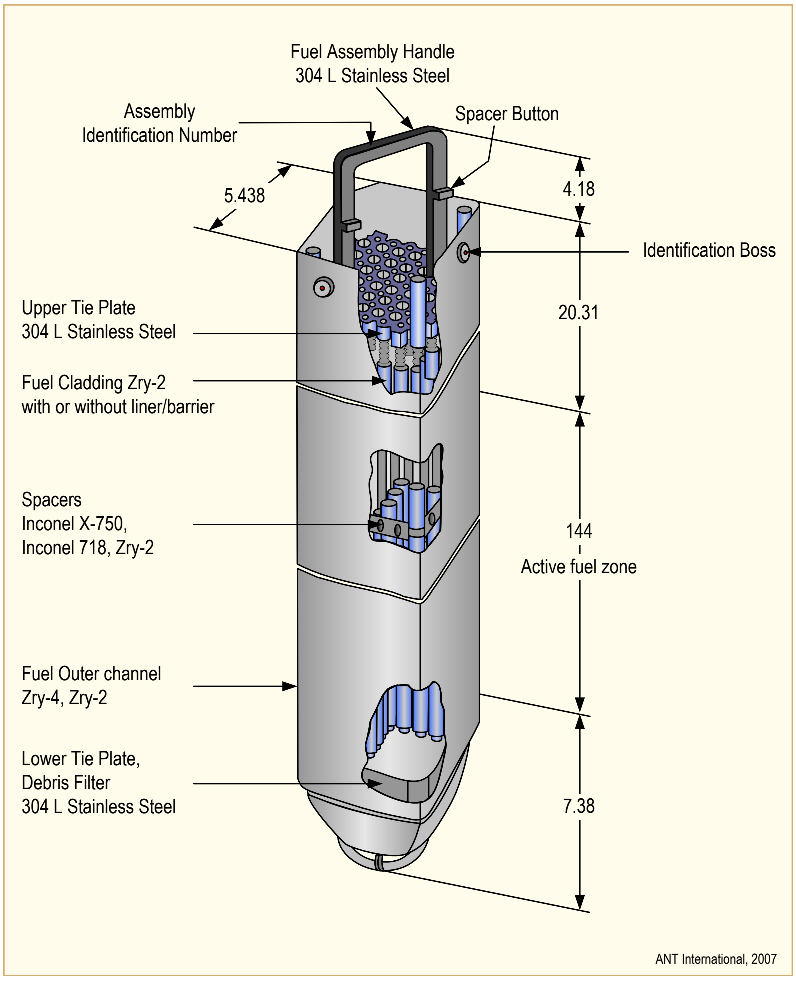

Figure 1-1: Typical BWR fuel assembly. All dimensions are in inches [Rudling & Patterson, 2009].

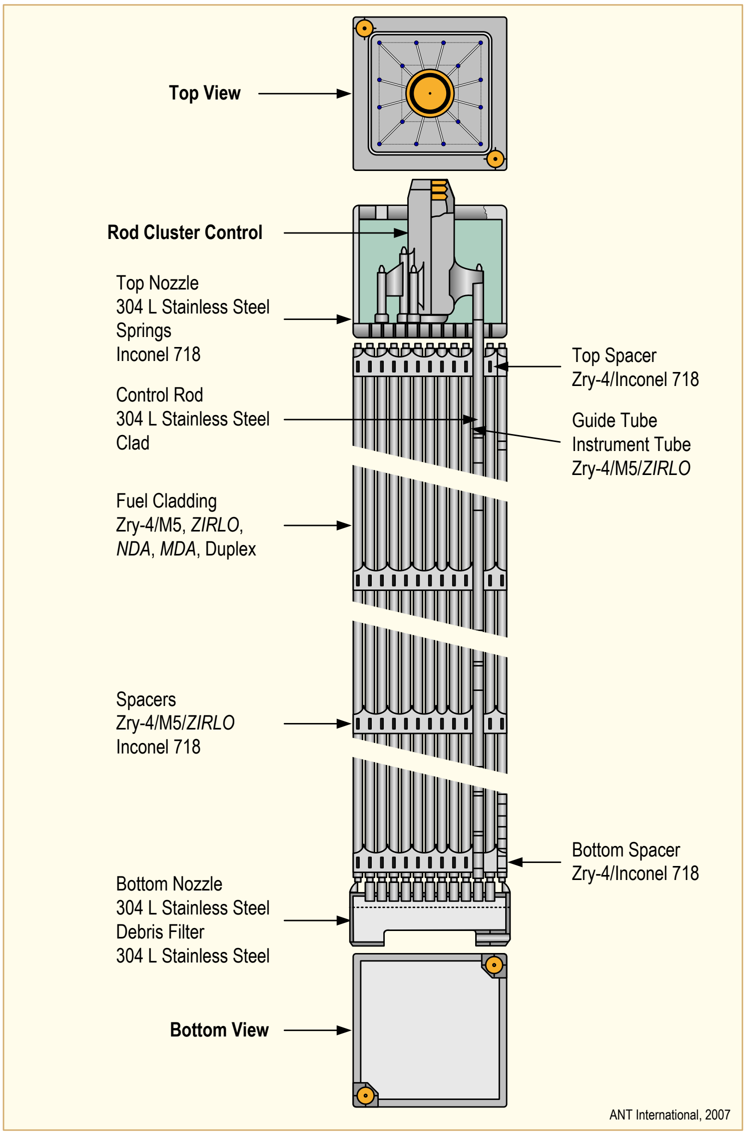

Figure 1-2: Typical PWR fuel assembly [Rudling & Patterson, 2009].

1.3 Effects of neutron irradiation and reactor environment on fuel assembly materials

This subsection gives a brief overview of how the neutron irradiation in the core impacts the properties of fuel and zirconium-based alloy materials and affects the performance of fuel assembly components. The interested reader of this topic is referred to FMTR Vol. 1 [Cox et al, 2006], FMTR Vol. 2 [Rudling et al, 2007] and FMTR Vol. IV [Rudling & Patterson, 2009] as well as the following ANT International Special Topic Reports (STRs):

- Corrosion & Hydriding
  - Corrosion of Zirconium Alloys, ZIRAT7/IZNA2 [Adamson et al, 2002/2003].
  - Corrosion of Zr-Nb Alloys in PWRs, ZIRAT9/IZNA4 [Cox et al, 2004/2005].
  - Corrosion mechanisms, ZIRAT12/IZNA7 [Adamson et al, 2007/2008].

  - Impact of Water Chemistry and Chalk River Unidentified Deposits (CRUD) on Fuel Performance, ZIRAT6/IZNA1 [Wikmark & Cox, 2001/2002].

  - The Effects of Zn Injection (PWRs and BWRs) and Noble Metal Chemistry (BWRs) on Fuel Performance, ZIRAT8/IZNA3 [Cox et al., 2003/2004].

- **Mechanical Properties**

  - Mechanical Properties of Zirconium Alloys, ZIRAT6/IZNA1 [Adamson & Rudling, 2001/2002].

  - Hydriding Mechanisms and Impact on Fuel Performance, ZIRAT5/IZNA1 [Cox & Rudling, 2000].

  - Pellet Cladding Interaction and Pellet Cladding Mechanical Interaction, ZIRAT11/IZNA6 [Adamson et al, 2006/2007].

- **Manufacturing**

  - Manufacturing of Zirconium Alloy Materials, ZIRAT5/IZNA1 [Rudling & Adamson, 2000].

  - Manufacturing of Zr-Nb alloys, ZIRAT11/IZNA6 [Nikulina et al, 2006/2007].

  - Welding of Zirconium Alloys, ZIRAT12/IZNA7 [Rudling et al, 2007/2008].

- **Dimensional Changes**

  - Dimensional Instability, ZIRAT7/IZNA2 [Adamson & Rudling, 2002/2003].

  - Structural Behaviour of Fuel Components, ZIRAT10/IZNA5 [Cox et al, 2005/2006].

- **Other Topics**

  - High Burnup Fuel Issues – Their Most Recent Status, ZIRAT8/IZNA3 [Adamson et al, 2003/2004].

  - Impact of Irradiation on Material Performance, ZIRAT10/IZNA5 [Adamson & Cox, 2005/2006].

During irradiation, the thermal, mechanical, and chemical conditions in a fuel rod vary with power, exposure or time and operating history. The temperature of the fuel follows a nearly parabolic distribution across the pellet radius and varies strongly with power. It also varies with the thermal conductance of the pellet-to-cladding gap, the cladding itself, the oxide, CRUD and water films on the outside surface of the cladding and with the bulk coolant temperature. Inside the fuel rod, several processes occur during irradiation. Early life is characterized by transfer of heat across the gas-filled gap between pellets and cladding, which almost universally contains helium. Early life also involves a number of processes that affect thermal and structural conditions in the rod; e.g.:

- Fracture of the fuel pellets due to thermal stresses starting on the first rise to power and continuing during the initial 5-10 GWd/MTU according to power and power history.

- Relocation of the resulting fuel fragments toward the inner cladding surface.

- Densification of pellets, which is typically small in modern fuel.

- Depending on the pellet manufacturing process, the evolution of a small amount of volatiles.

Continued irradiation of fuel rods leads to:

- Fission gas release with increasing gas pressure inside the fuel rod and decreasing thermal conductivity of the helium filler gas due to the addition of krypton and xenon.
- Solid and gaseous swelling due to the accumulation of fission products and the formation of bubbles and other gas filled voids.
- Inward cladding creep due to the temperature, fast neutron flux and the net difference between internal gas and external coolant pressures.
- Increasing amounts of pellet-cladding mechanical interaction due to gap closure by relocation, swelling, cladding creep down and differential thermal expansion between pellets and cladding.
- Healing of the cracks among pellet fragments by in situ sintering to a degree that depends on operating conditions and history.
- Hot pressing and the formation of dishes on the ends of flat pellets or filling of dishes, which were formed during pellet fabrication.
- The release of fission products with the potential for contributing to cladding failure by processes such as Stress Corrosion Cracking (SCC) or liquid metal embrittlement; i.e., iodine or cadmium with caesium.
- Formation of a unique "high burn up structure" at the pellet periphery.

At moderate to high exposures, the chemistry of the fuel changes in a manner that can affect its thermal and mechanical properties. An example is an increase in the pellet oxygen-to-metal ratio leading to a decrease in thermal conductivity, an increasing pellet temperature and an increase in pellet creep rate. With sufficient contact at the pellet-cladding interface, temperature and time, oxygen can also be transferred from the fuel pellets to the cladding (fuel-clad bonding).

On the outer surface of a fuel rod, exposure, time, temperature or combinations of these factors lead to oxidation of the cladding and deposition of corrosion products (CRUD) from reactor internals and piping. Surface corrosion reduces the thickness of the cladding wall and leads to the pickup of hydrogen in the cladding metal. Wall thinning is typically small in modern zirconium-based cladding, but is addressed in the design process and is identified as a key factor in postulated LOCAs. Hydrogen pickup leads to concentrations in the cladding that increase with exposure or time and normally exceed the solubility limit at operating temperatures by low-to-moderate burnups. Hydrogen concentrations in excess of the solubility limit lead the precipitation of hydrides. Such hydrides are brittle relative to zirconium-based cladding. At sufficiently large concentrations and sufficiently low temperatures (<200°C), hydrides can degrade the strength and ductility of the cladding. The density of zirconium hydride is lower than that of zirconium, so that their precipitation generates local strains in the cladding. With increasing concentrations, the combination of low ductility and local strains can cause structural failure at the location of hydride blisters, lenses or sunbursts. Based on experience in Japanese power and test reactors, hydrides have also been implicated in an outside-in, hydride-assisted fuel cladding cracking process.

The irradiation of neutrons, with energies in excess of about 40 eV, results in damage of the zirconium alloy lattice. The higher the energy of the neutrons and the larger the fluence, the larger lattice damage. This damage changes the zirconium alloy properties dramatically. Such changes include:

- An increase of yield and ultimate tensile strength.
- A decrease of ductility.

- An increase in corrosion rate (NB: The decomposition of water by neutron interactions and other irradiation effects will also increase the corrosion rate by itself).

- An increase in rate of dimensional changes due to increases in:

- Creep rate (also including residual stress relaxation).

  - Irradiation growth and growth rate.

In addition, zirconium alloys undergo water-side corrosion. Corrosion of zirconium alloys is a thermodynamic and electrochemically based process affected by the following parameters:

- Protectiveness of the zirconium oxide film formed which depends on:

  - The microstructure of the metal surface.

  - The water chemistry and the hydraulic conditions.

- The Zr alloy temperature (at the metal/oxide interface).

Pellet-cladding interaction (PCI) is a failure mechanism associated with local power increases (ramps) during reactor start-up or manoeuvring; e.g., rod adjustments, sequence exchanges or power increases after an extended interval of low power operation. The process, which is shown schematically in Figure 1-3 involves the joint occurrence of tensile stress in the cladding, the presence of damaging fission products and material that is susceptible to SCC or Liquid Metal Embrittlement (LME). The PCI process is generally attributed to iodine released as a fission product from irradiated fuel pellets and SCC. The sensitivity of zirconium alloys (to SCC) comes from the alloying elements in the material and from irradiation damage, which increases YS above the threshold for crack initiation. PCI cracks start at the inner cladding surface and progresses towards the outer surface in the minute scale.

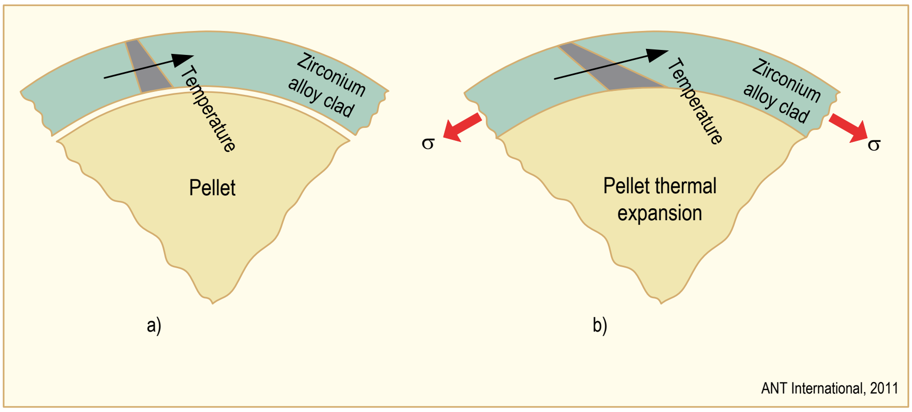

Figure 1-3: Schematics showing the fuel rod condition, (a) before the ramp and, (b) during the ramp.

PCI failures have been observed in PWRs, VVERs, CANDUs and BWRs. The failure mechanism is much more prevalent in BWRs since reactor operation is frequently done by control rod movements at moderate to high power. In PWRs/VVERs, reactor power is normally not controlled by insertion and extraction of the control rods in the core; instead, reactor power is controlled by the boron concentration that is continuously decreased during operation to compensate for the decrease in reactivity. This type of reactor power control is much slower and more uniform over the length of the fuel than in the BWR case and consequently PCI failures are less common in PWRs/VVERs. However, during reactor power increases, and specifically during a class II transient (Anticipated Operational Occurrence or AOO), PCI failures may occur in PWRs/VVERs.

Pellet-cladding mechanical interaction (PCMI) refers to contact and mechanical interaction between pellets and cladding. With sufficient tensile deformation of the cladding, PCMI can produce cracking that initiates on the outside surface in a purely mechanical way; e.g., crack initiation in a hydride-rich brittle zone at the outer surface with crack propagation by stress-induced hydrogen diffusion to the crack tip. The ductility of irradiated zirconium alloys is generally adequate at the operating temperatures and hydrogen concentrations typical of modern fuel. So, there have not been any recent fuel failures in BWRs or PWRs reported due to PCMI and hydrogen assisted cracking. However, PCMI has been observed in test-reactor power-ramp experiments with high-burnup rods and is an area of active investigation for high burnup fuel [Hayashi et al, 2005].

Irradiation of the fuel assembly leads to several dimensional changes, which may have a significant impact on its performance. Below are given some examples of potential issues related to dimensional changes:

- Excessive fuel rod length increase may lead to fuel rod bowing, Figure 1-4,
- Excessive spacer strap length increase (BWR),
- Excessive guide tube growth leading to fuel assembly bowing and twisting (PWRs), Figure 1-5,
- Excessive fuel outer channel bow, twist, and bulging (BWRs), Figure 1-6 and Figure 1-7.

Figure 1-4: White arrow indicates large fuel rod bowing due to excessive fuel rod growth [Franklin & Adamson, 1988].

2 On-site fuel examinations

Fuel examination programs at reactor sites and in hot cells address the design and licensing criteria. Such programs are typically structured both to generate data on the performance of materials and fuel assemblies and to verify conformance with licensing requirements. A large amount of these data have been acquired in the past and constitute the technical basis for current materials, designs and operating strategies. New data are being acquired in ongoing programs with the objective(s) of reducing the uncertainty associated with existing models, developing improved models, assessing the performance of new materials, design and operating changes and resolving fuel performance issues. The methods used on-site and in hot cell examinations are discussed in this and the following sections, respectively.

2.1 On-site fuel assembly characterization

2.1.1 BWR fuel channel characterization

Various pieces of poolside equipment can be used to determine the dimensions of the outer flow channel of BWR fuel assemblies. Figure 2-1shows a schematic of a system, which utilizes LVDTs (Linear Voltage Differential Transducers), to measure the position of a channel relative to a reference surface. In this case, each of the four sides is measured by three transducers and consequently 12 axial traces are obtained simultaneously over the circumference of the channel. From these measurements, the bulge, bow, and twist may be calculated over the total length of the channel. The dimensions of the outer flow channels are of interest in the performance of individual BWR fuel assemblies and in the interaction of the bundles comprising a control cell with each other and with their control blade.

Bulge is a measurement of the out-of-plane deformation of the channel faces. Bulging results from the difference in pressure between the inside and the outside of the flow channel and from the effects of fast neutrons on channel creep. Bulging can also result from the force of mechanical seals (or finger springs) that are used in some designs to reduce the bypass leakage between the channel and lower support plate. The pressure differential and seal force is greatest at the bottom of the flow channel and vanishes at the top of a fuel assembly. The fast neutron flux varies with operating conditions, but is relatively small at the bottom of a fuel assembly, increases with elevation and void fraction and then decreases due to flux leakage near the upper end of an assembly. The net result is an outward deformation that is greatest at the transverse centre of each face near the lower end of a fuel channel and varies with axial position along the length of a channel. Such outward deformation of a channel is of interest because of its potential effects on the leakage of coolant into the bypass region outside of the channel and on the interference with the movement of adjacent control blades.

Channel bow, as shown in Figure 2-1, is a lateral deformation relative to an idealized centreline of the fuel assembly. Twist is an angular deformation over the length of an assembly. A BWR fuel assembly without internal flow channels is sufficiently flexible that it follows the bow and twist of its outer channel. Part of both types of deformation typically results from the effects of temperature and fast neutrons on stress relaxation and irradiation growth. Residual stress from manufacturing operations can cause a channel to deform due to thermal and fluence induced relaxation. The texture and residual, as-built microstructure of zirconium-based alloys can contribute to significant amounts of lateral deformation due to the effects of fast neutron fluence and fluence gradients on the differential irradiation growth among channel faces. Another part of channel bow results from shadow corrosion that has been observed to lead to differences in the pickup of hydrogen and differential growth among channel faces. For both irradiation and corrosion induced bow, a channel bows outward away from its centreline on the face with the largest growth.

Lateral growth is of interest due to its potential effects on the motion of adjacent control rods, on local increases in thermal neutrons and, in extreme cases, on the width of flow passages between the channel and fuel rods.

Length of the fuel channel is also of interest and can be measured with systems similar to that shown in Figure 2-1 when equipped with the appropriate sensors. Length can also be determined by means of in-pool devices that range from tape measures to calliper-like gages equipped with LVDTs or similar sensors. Length data are used to assess the effects of differential growth in the axial direction on the fit and remaining growth margin of the channel relative to its fuel bundle. Length data can also be used to estimate lateral bow in cases where time or equipment availability prevents more accurate, explicit measurements.

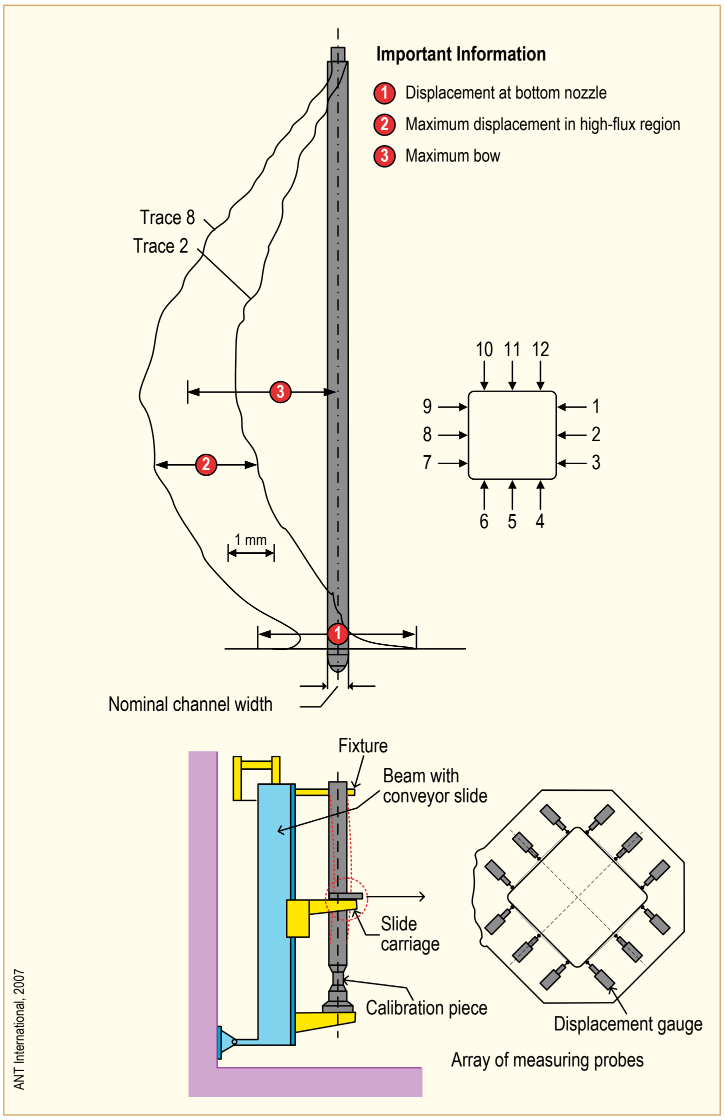

Figure 2-1: Schematic view of the channel measurement device and typical measurement results [Knaab & Knecht, 1978].

Oxide characteristics and oxide thickness measurements of the fuel channel can be obtained by visual examination and EC (eddy current) measurements, respectively. Figure 2-2 shows normal appearance of oxide on a fuel channel. Figure 2-3 shows fuel channel bow for three different materials, while Figure 2-4 provides oxide thickness data for Zry-2 and Zry-4 fuel channels.

Figure 2-2: Photos showing normal oxidation features of fuel channels. The figure to the right shows some Shadow Corrosion as a result of the core grid, provided by the courtesy of OKG-EON.

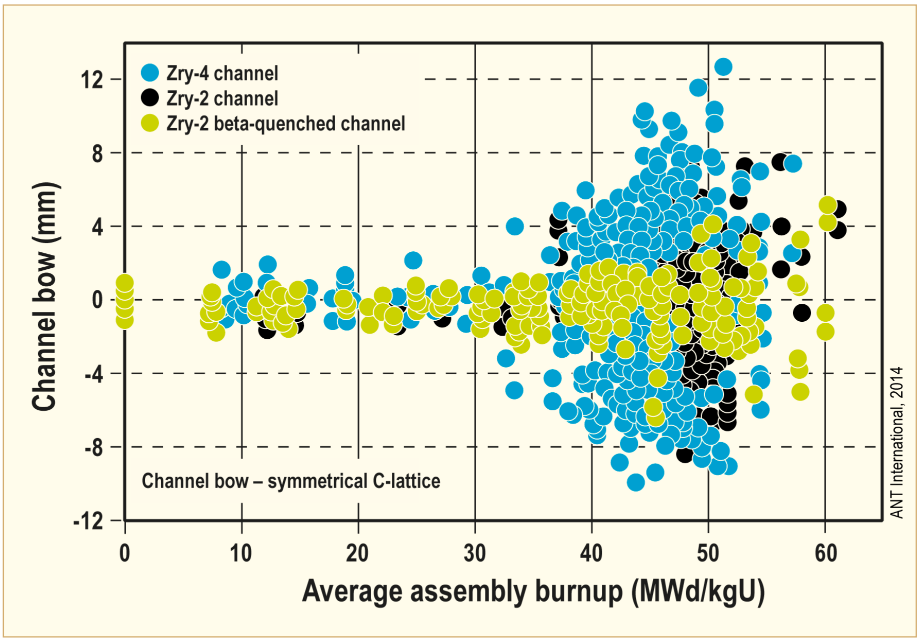

Figure 2-3: Results of measurements of channel bow on SVEA-96 channels of reload batch e7 to e20 by KKL (n = 1284), after [Ledergerber et al, 2007].

3 Non-destructive hot cell examinations

The shipment of failed and sound sibling fuel rods to a hot cell facility is accompanied by paperwork that provides needed information about the rods. This usually includes:

- Power plant name
- Reload number
- Bundle ID(s)
- Rod IDs
- Rod serial numbers
- Beginning of life (BoL) date
- End of life (EoL) date
- Irradiation time (days)
- Bundle operation history
- Rod average exposure (GWd/MTU)
- Rod condition (leaking or sound)
- Objectives of PIE

In addition, rod fabrication information is also needed, which includes:

- Type of rod (10x10, 12x12 etc., Tie rod, Poison, Additive)
- U-235 enrichment zone (%)
- Total U (grams)
- U-235 (grams)
- Initial fuel column length
- Initial He pressure

Also needed is the information about the spacer/grid design and azimuthal orientation of their contact points with the rods. These sets of information are needed for:

- Verification of the rod identities when rods are unload from the shipping cask into a hot cell
- determination of the azimuthal orientation of a fuel rod in its bundle,
- proper tracking of the rods and their segments (after sectioning),
- selection of examination/test equipment with correct dimensions,
- accurate inventory of radioactive and fissile materials, and
- calculations such as fission gas release and void fraction.

3.1 Visual examination

Hot cell examinations of fuel rods vary with the objective of the campaign, but usually begin with a visual inspection. The visual inspection method is used to verify the rod serial numbers, document the condition of the failed fuel rods, and to ensure the integrity of any sound sibling rods. Visual inspections are also 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. Another piece of information collected/documented during visual examination is the azimuthal orientations of the spacer-rod contact points. This information along with the bundle fabrication records is used to determine the azimuthal orientation of the rod in the bundle.

The equipment used for visual examination may be a digital camera connected to the outer end of the hot cell Kollmorgan[3] through-wall shielded periscope (Figure 3-1) for full-surface inspection and photo-documentation of irradiated fuel rods or a computer-controlled shielded digital video camera that can also be used to take still photographs. The cell is usually equipped with commercial photographic strobe lights that are used exclusively for photography. It is important that the Kollmorgan and the video camera are equipped with special planar optics that maintain the entire surface of a flat object (oriented normal to the optical axis of the system) in focus at the film plane. The fuel rod is supported by the in-cell roller assembly fitted with a calibrated and certified tape measure. The fuel rod is placed next to the tape measure with zero inch mark coinciding with the tip of the bottom end plug. All other elevations are measured with reference to this point. Both ends of the rod are secured in a system that allows azimuthal rotation of the rod. The visual examination is usually performed at 0, 90, 180, and 270 degree orientations. For reference purposes the first letter of the rod serial number engraved at the bottom end plug is considered as zero degrees with positive angular position defined as clockwise rotation when viewed from the top of the rod.

When a fuel rod segment is visually examined, the cumulative length of all lower segments and cutting swarf is added to the scale reading to determine the true axial position. The azimuthal orientation and top/bottom orientation of each rod segment is maintained through the use of small paint dots that are usually placed at the zero degree orientation of the top end of each rod segment and sub-segment along with the segment number. Record of each and every sectioning of the rod is maintained in the corresponding sectioning diagrams.

Examples of some features of LWR fuel rod photographed during visual examination are shown in Figure 3-2 through Figure 3-4.

[3] In 1900s, Frederick Kollmorgan, skilled in optics, decided to move from Germany to United States. He settled in New York and started Kollmorgan Corporation to develop and manufacture optical periscopes for use on US submarines. Later modifications were made to the Kollmorgan design for use as sealed hot-cell periscope to permit installation from the outside of the cell, simplify movement, and prevent motion of the optics during use. The modifications included a carefully prepared penetration of the cell wall, an O-ring sealed dome and sleeve assembly, and replacement of a counterweight with a simple brake shoe [Olsen et al, 1963].

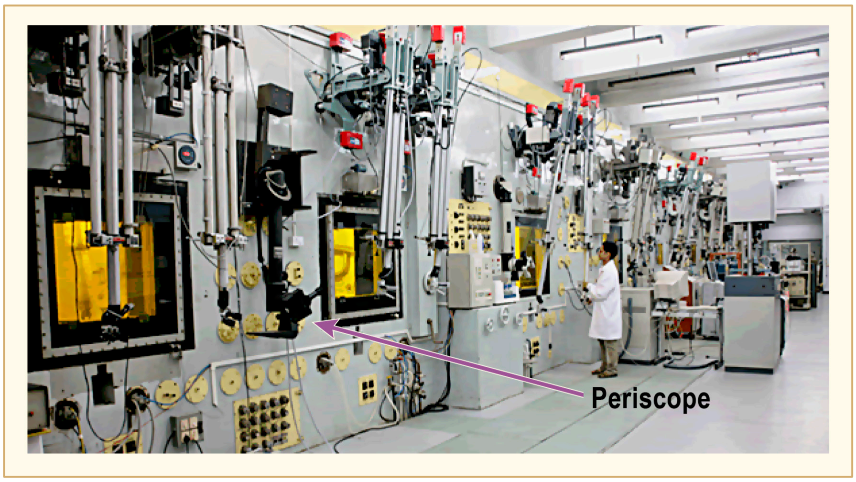

Figure 3-1: Hot cell periscope may be used for visual examination of a fuel rod (from www.barc.gov.in).

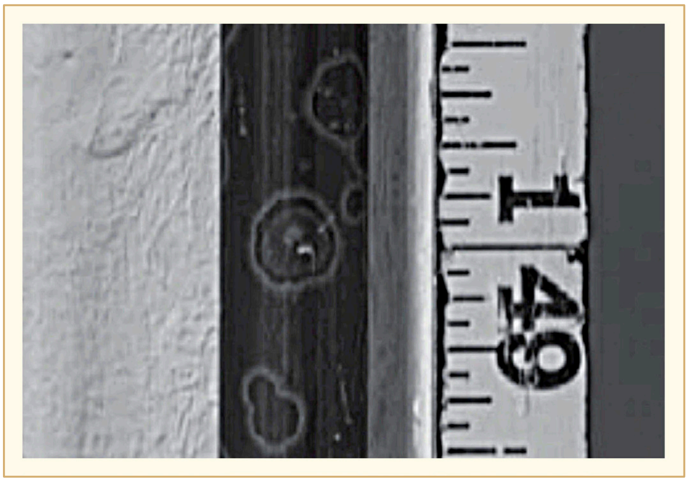

Figure 3-2: Visual image of features on a light water reactor fuel rod.

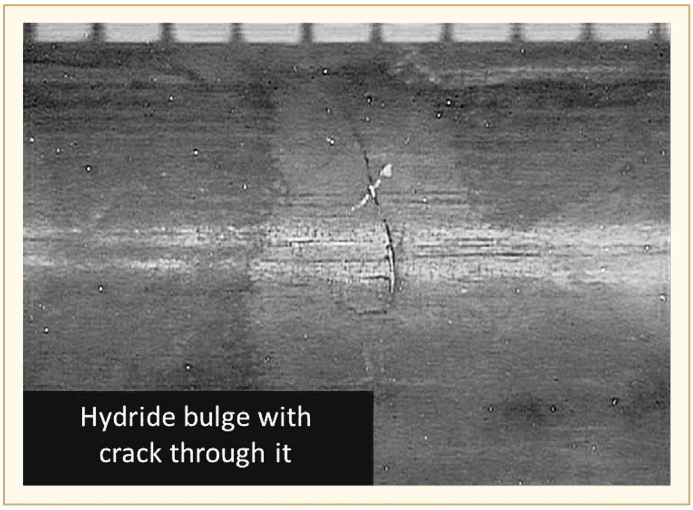

Figure 3-3: A cladding hydride bulge with crack through it.

4 Destructive hot cell examinations

A number of destructive techniques are used for hot cell examination of the fuel assembly components, such as

- Optical metallography to measure cladding ID, OD, and oxide thickness
- Optical metallography to characterize hydride distribution in the component under investigation
- Hot extraction technique to measure hydrogen concentration
- Measurement of second phase particles (SPPs) size and distribution
- Microstructure characterization
- Ceramography of fuel pellets
- Microgamma scanning across the *diameter* of the fuel pellets

4.1 Cladding metallography

Metallographic samples of irradiated material are prepared in hot cells. Various optical devices are used for process control and inspection. Periscopes may be used to view and photograph a wide region inside a hot cell within an area of 180^{o} centred on the objective end.

For inspection of a metallographic sample, optical Macro Viewing System is deployed because the samples need to be checked for proper polishing and etching. There is a sufficient distance between the object and the instrument, so that the sample may be easily kept under the macro-scope with the help of manipulators. The macro viewing system has binocular vision for ease of use. The length of the system depends on the wall thickness of the hot cell. The optical path is periscopic in nature and the macro-scope has radiation shielding blocks built into the optical system to avoid direct exposure of radiation.

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 or distribution of hydrides in the sample wall thickness.

Several samples along the length of a fuel rod are selected to establish axial oxide thickness and cold gap profiles, to establish hydride distribution characteristics, and to characterize primary and secondary failures (Figure 4-1). Metallographic samples are obtained by sectioning the fuel rod segments normal to their long axis with a slow speed diamond saw. A well-established system is used to track the axial location and axial and azimuthal orientation of the metallographic samples. Following sectioning, the metallographic samples are photographed (Figure 4-2), mounted and then ground through 4000 grit SiC paper followed by polishing with a silica-based polishing compound and polishing cloth. In many cases, multiple planes are examined either to locate a defect or to establish axial profile of a primary failure.

Hot cell facilities usually have a metallography cell connected to a main cell through a transfer tube. Metallographic images of irradiated samples can be acquired using either an optical metallograph or optical system of a micro-hardness tester. A polished cross-sectional view of a metallographic sample is shown in Figure 4-3. Figure 4-4 and Figure 4-5 show examples of PCI cracks in the polished and etched conditions, respectively, discovered by successive polishing and examination of the metallography sample obtained from a region of interest based on non-destructive examinations.

The use of cladding metallography for various purposes is discussed in the following sections.

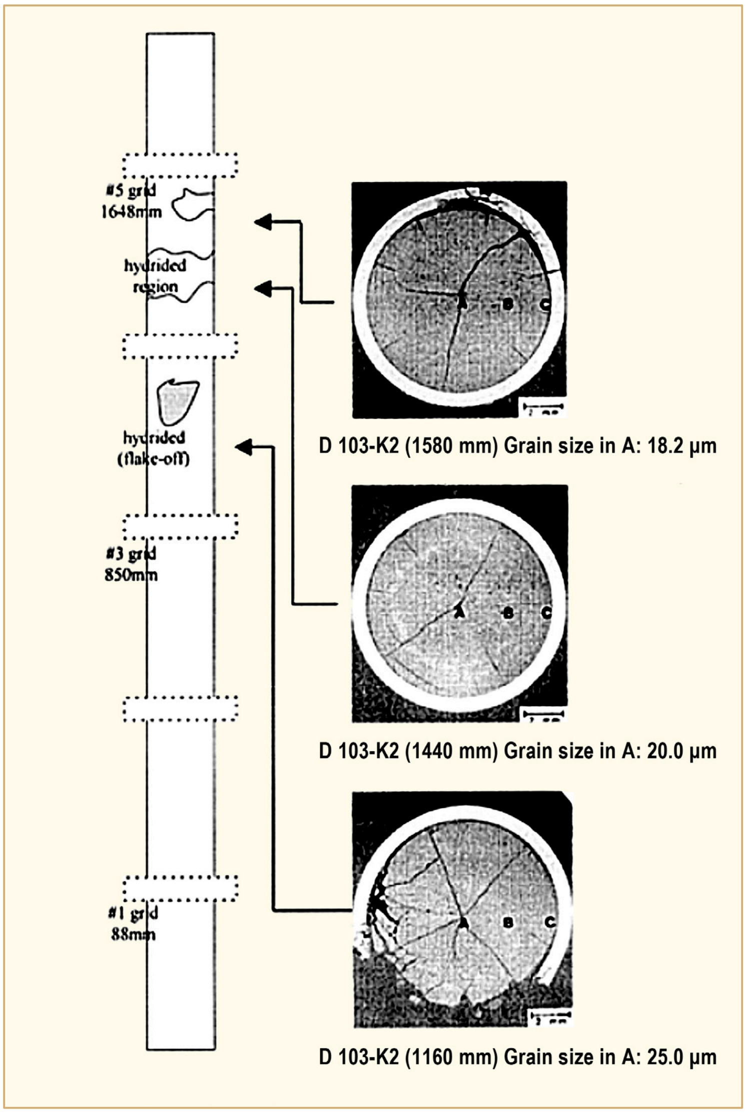

Figure 4-1: Samples taken from different elevations of a PWR fuel rod to investigate the microstructural changes of UO_2 [Kim, 2002].

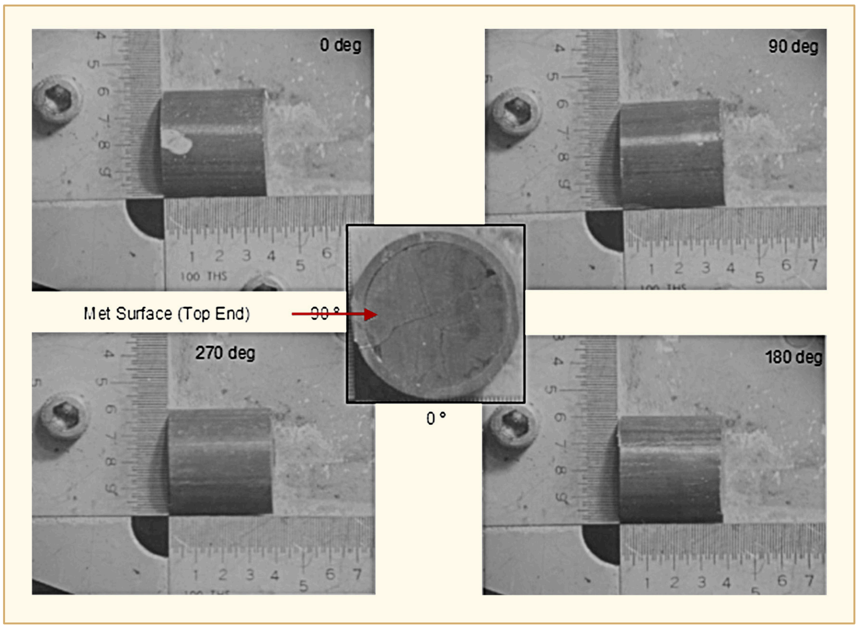

Figure 4-2: Optical photographs of a metallography sample sectioned out of an irradiated LWR fuel rod.

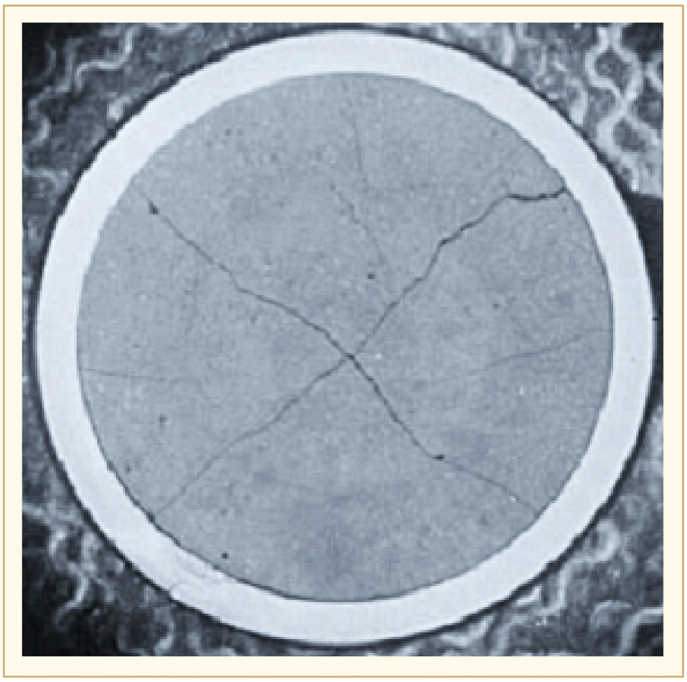

Figure 4-3: Optical photograph of polished cross-sectional view of a metallographic sample taken from an LWR irradiated fuel rod.

5 Mechanical properties measurements

Irradiation increases the tensile Yield Strength (YS) while decreasing the ductility/fracture toughness of Zr-based materials relatively early in the life of a fuel element; i.e., saturation values are approached in the initial 5-10 MWd/kgU of operation. Also as noted earlier, the uptake of hydrogen from the corrosion reaction during continued operation and the formation of hydrides in the matrix further reduces the ductility/fracture toughness of Zr-based alloys at temperatures less than 200°C. While in solution, hydrogen is distributed uniformly according to the local temperature and the corresponding solubility limit. The soluble hydrogen diffuses to cool spots that include the cladding OD, spaces under spalled oxides, pellet-to-pellet gaps, and to the ends of the fuel rods. The formation of hydrides can also be affected by the local stress field. The areas of high hydride concentration exhibit particularly low ductility/fracture toughness, even compared to the bulk cladding with homogeneously distributed hydrogen, and can be very sensitive to crack initiation and propagation. The margins to ductility limits can be reduced significantly by local, high hydrogen concentrations. In addition, the orientation of hydrides relative to the stress field can have a dramatic impact on ductility/fracture toughness. For example, circumferential hydrides tend not to affect cladding ductility/fracture toughness under the tensile loading due to PCMI while radial hydrides do, Figure 5-1. It turns out that the embrittlement effect of the hydrides is very much dependant on temperature so that fuel cladding at operating temperature is ductile even with very large hydrogen concentrations. At room temperature, however, ductility/fracture toughness is reduced significantly by low-to-moderate concentrations of hydrogen.

The tendency to form radial hydrides is much stronger in RXA material than in SRA material due to a preferential precipitation of hydrides on grain boundaries. RXA material has a larger fraction of grain boundaries in the radial direction. SRA material contains a larger fraction of grains with the shape and orientation created by the fabrication (tube reduction) process.

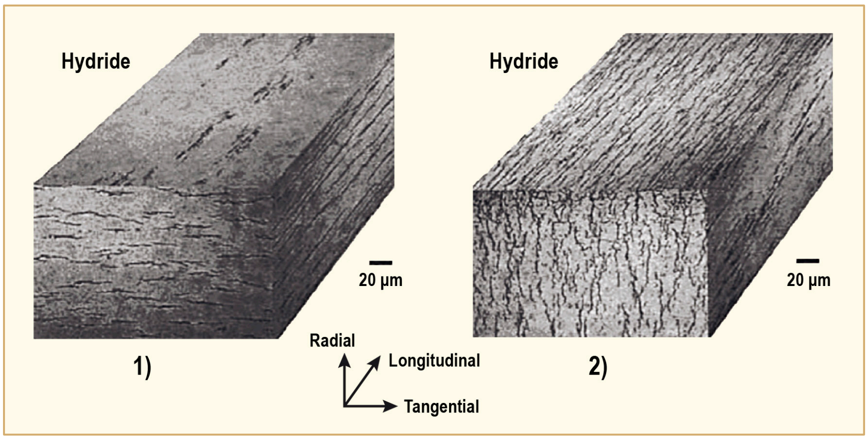

Figure 5-1: Hydride morphology in stress relieved Zircaloy. (1) Circumferential (tangential hydrides). (2) Radial hydrides.

When preparing to do mechanical testing of a material or component of interest, several factors must be considered:

- Careful analysis of the intended application of the data
- Material availability:
    - Irradiated or not
    - Actual component or not
- Means of fabricating specimen

- Conditions to be simulated (or “mocked-up”)
  - Temperature
  - Strain rate
  - Stress state
  - Stress or displacement-controlled deformation
  - Environment
- Specimen geometry
  - Plane strain
  - Plane stress
  - Gauge length and gauge shape
- Metallurgical condition of material
  - Irradiated or not
  - Texture
  - Heat treatment (RX, PRX, CW, CWSR)
  - Hydrided or not
- Test facility
  - Allow radioactivity or not (exposure level from the test sample)
  - Apply and measure force and displacement
  - Measure or calculate strain
  - Post-test examinations

The strength, ductility, and other mechanical properties of irradiated Zircaloy cladding can be measured by performing mechanical tests on differently shaped samples machined out of irradiated cladding. Machining of test samples out of irradiated Zirconium alloy components is performed remotely in a hot cell using either a milling machine or an electric discharge machine (EDM). This makes sample preparation more complicated and time consuming than unirradiated material. Depending on the contamination and exposure levels of the samples the testing can be performed in a hot cell or in a radioactive materials laboratory. Some of the tests that can be performed on irradiated cladding (and un-irradiated archive material) include

- Hardness testing
- Room and elevated temperature uni-axial tensile tests
- Ring tensile tests
- Ring compression tests
- Plane strain tests
- Fatigue tests
- Creep tests

5.1 Hardness testing

Hardness of a metal is a measure of how resistant the metal is to various kinds of permanent shape change when a force is applied. Hardness of a material is related to strength and ductility, resistance to plastic deformation, and wear resistance; however, it is most commonly defined as resistance to indentation. All of the hardness tests evaluate indentations in one way or another. Hardness measurement can be defined as macro-, micro- or nano-scale according to the forces applied and displacements obtained.

Where materials, like zirconium alloys, have a fine microstructure, are multi-phase, non-homogeneous, or prone to cracking, macro-hardness measurements will be highly variable and will not identify individual surface features. It is here that micro-hardness measurements are appropriate. Micro-hardness is the hardness of a material as determined by forcing an indenter such as a Vickers or Knoop indenter into the surface of the material under 10 to 1000 gf load; usually, the indentations are so small that they must be measured with a microscope or a closed circuit digital (CCD) camera. The technique is capable of determining hardness of different micro-constituents within a structure, or measuring steep hardness gradients such as those along the wall thickness of unirradiated and irradiated Zircaloy cladding. For metallurgical issues that require hardness determination over small areas or on components too thin to tolerate large loads or indentations, micro-hardness techniques are used. Such techniques are regularly used for zirconium alloy components such as fuel cladding or spacer/grid bands.

Special considerations for hardness testing of irradiated zirconium alloy components in a hot cell include the following.

- Although elevated-temperature (hot) hardness techniques and equipment are available, they are not often used and these tests are carried out at an ambient temperature within the limits of 10 to 35°C [ASTM E110, 2013], [ASTM E18, 2013] and [ASTM E384, 2013]. It is, therefore, important to make sure that the cell has proper air ventilation so that the temperature inside the cell stays within the limits and is steady during testing.

- Due to a number of factors, there are more vibrations inside a hot cell compared to a regular laboratory. Attempts should be made to keep these vibrations to a minimum so that they don't interfere with the process of indentation and measurement of the indent dimensions.

- If the test sample is machined out of an irradiated Zircaloy component, the sample must be machined in such a way that the indenter moves exactly perpendicular to the surface of interest (radial, hoop, and axial in the case of a cladding, pressure tube, water rod etc.; and axial, transverse, and thickness faces in the case of plate or sheet such as channels, spacers etc.)

- When performing hardness test on an unirradiated or irradiated Zircaloy sample/component, make sure the test surface is mechanically polished without any significant mechanically damaged layer. Never measure hardness on a chemically etched surface, etching changes the hardness of the surface.

- If the hydrogen concentration of the irradiated Zircaloy component is very high, the component and the machined sample need to be handled very carefully because the material might be very brittle and can crack or even break during handling.

- The load applied during hardness testing depends on the thickness of the sample/component. If it is too thin like cladding, channel, and spacer components, smaller loads will be more appropriate. The general rule is that the sample thickness should be about 10 times the indentation depth.

6 PIE of fuel hardware

6.1 PIE of spacers/grids

6.1.1 Visual examination

Before any destructive examinations, visual examinations are carried out. The equipment used for visual examination may be a digital camera conned to the outer end of the hot cell Kollmorgan through-wall shielded periscope for full-surface inspection and photo-documentation or a computer-controlled shielded digital video camera that can also be used to take still photographs. The cell is usually equipped with commercial photographic strobe lights that are used exclusively for photography. It is important that the Kollmorgan and the video camera are equipped with special planar optics that maintains the entire surface of a flat object (oriented normal to the optical axis of the system) in focus at the film plane. An example photograph is shown in Figure 6-1.

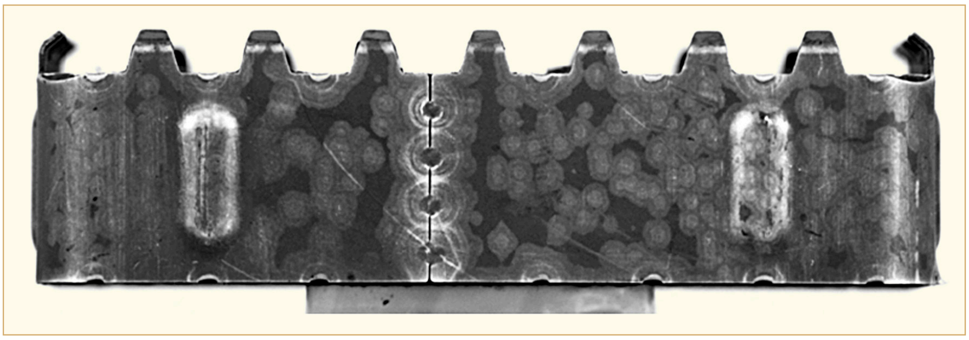

Figure 6-1: Visual Appearance of a Zr BWR spacer. The lighter areas indicate thicker oxides [Shimada et al, 2005].

6.1.2 Oxide thickness measurement

The oxide thickness on the surface of fuel bundle hardware can be measured either by in-cell EC technique or by destructive optical metallography. Figure 6-2 gives oxide thickness data of spacer components measured by EC technique. The collected EC data require correction if ferromagnetic CRUD is present on the spacers.

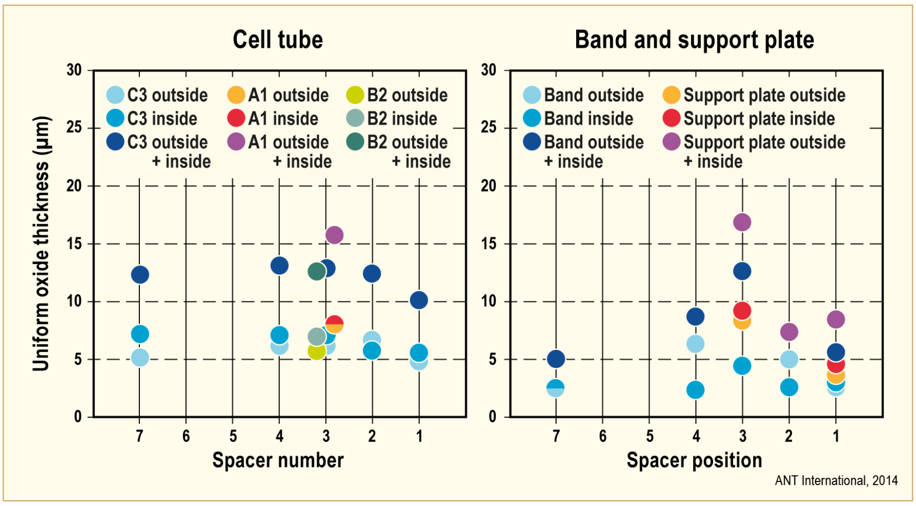

Figure 6-2: Uniform oxide thicknesses of spacer's components [Shimada et al, 2005].

6.1.3 Hydrogen measurement

The hydrogen concentration data, corresponding to the corrosion data in Figure 6-2, measured by hot vacuum extraction technique (described earlier) is provided in Figure 6-3.

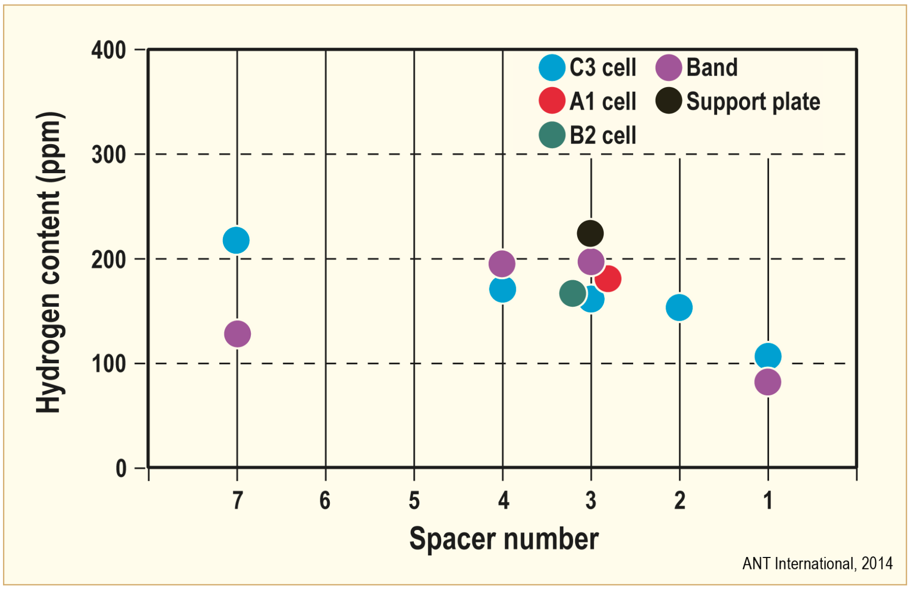

Figure 6-3: Hydrogen content in spacer straps, after [Shimada et al, 2005].

The hydride morphology can be determined by etching the Zr alloy materials to reveal the hydrides using optical metallography, as shown in Figure 6-4.

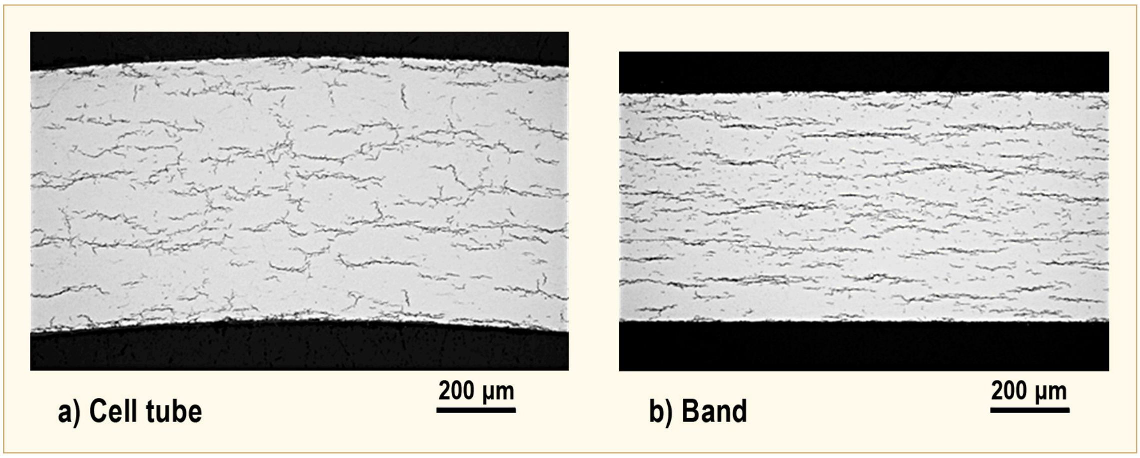

Figure 6-4: Hydrides in spacer components [Shimada et al, 2005].

6.2 PIE of BWR channel boxes

Shadow corrosion is an important area of investigation. Although agreement does not exist regarding the underlying mechanism, shadow corrosion is frequently postulated to be a galvanic type of corrosion. It normally occurs only in BWRs because of the large oxidation potential in the coolant. Shadow corrosion occurs in areas where a Zr alloy is in contact with or in close proximity to a dissimilar material such as nickel-based alloys (Inconel) or stainless steel. Shadow corrosion seems to be a form of nodular corrosion, which grows relatively quickly and then remains at a constant thickness through higher burnups. An example of shadow corrosion caused by the neighbouring control blade handle on the channel surface is shown in Figure 6-5.

Figure 6-5: Shadow corrosion due to control blade handle at ~8 inch elevation, which corresponds to the fully-withdrawn blade position [Mahmood et al, 2007].

Due to the low oxidation potential in a PWR/VVER only uniform corrosion normally exists. However, if for some reason there would be an increase in the coolant oxidation potential, nodular corrosion may also occur in PWRs/VVERs. In PWRs, there may be an increase in the oxide growth rate at about 20-30 microns because solid hydrides may form in the Zr alloy at this point, thus increasing the oxide growth rate. At oxide thicknesses of about 100 microns and a large surface heat flux, there may also be a thermal feedback effect, which then leads to a second oxidation acceleration step. In VVERs, there is normally no acceleration of the uniform corrosion process due to the very low hydrogen pickup. The reason for the low hydrogen pickup is due both to the use of the Zr1Nb alloy E110 and different coolant chemistry [Rudling & Patterson, 2009].

Prior to year 2000, channel distortion was attributed only to bow caused by differential irradiation growth due to a fluence gradient across the channel (i.e. the reactor core's flux gradient leads to higher accumulation of fluence on one side of the channel than the opposite side) and to bulge arising from irradiation creep induced by the channel inside-out pressure difference. These distortion mechanisms were effectively managed and accommodated during core design (e.g. by limiting the number of cycles in peripheral locations). Thus, when channel-control blade interference was noted starting in 2000, the observations were unexpected. A BWR control cell with four bundles and a control blade is shown in Figure 6-6. The bow and bulge conditions are exaggerated for illustration purposes [Mahmood et al, 2010].