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Research

Study of degradation of spent fuel's structural material and core components under long-term wet storage

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SSM perspective

Background

Sweden is managing the spent nuclear fuel and core components from its commercial nuclear reactors with a national strategic plan. The fuel and core components are initially stored at the reactor site. After an on-site storage period to comply with transportation limits on decay heat and radiation, these materials are shipped to the Clab (Central Interim Storage Facility for Spent Nuclear Fuel). The Clab is located in Oskarshamn, Sweden. The Clab facility operates several linked underground storage pools which are expected to have a storage capacity of 11,000 tons. Figure 1 is a photo of one of the underground spent fuel storage pools at Clab. Clab is expected to be operational until 2070 to 2075.
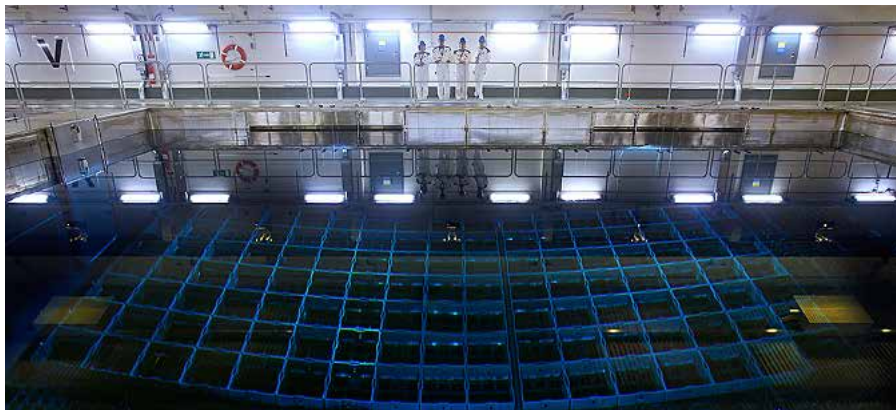

Figure 1: Spent fuel storage pool at Clab (Source: SKB – "Clab – Central Interim Storage Facility for Spent Nuclear Fuel", Photo Curt-Robert Lindqvist)

Spent fuel and core components may reside at Clab for up to fifty years prior to final disposition or disposal in Sweden's repository. Storing this volume of spent fuel and core components requires significant handling: moving fuel and core components into compact storage canisters and moving the canisters in the various pools. During this storage period, aging and structural degradation may compromise the integrity of the fuel and core components to be packaged in repository containers. Packaging or encapsulation in the copper disposal canisters requires handling and vacuum drying. Handling and drying of spent nuclear fuel at the end of an extended pool storage period is a structural integrity and fission/activation product containment concern. The release of fission and activation products into the Clab pool storage facilities due to this degradation would increase the risk and operational difficulties at Clab and later with encapsulation of spent fuel at Clink.

The design and operation of the Clab facility leverages a considerable body of data demonstrating safety and efficacy of pool storage of spent nuclear fuel and core components. Pool storage allows direct and indirect monitoring of the storage conditions and inspection of the items in storage up to the point of final encapsulation in disposal containers.

Results

The A.N.T. International team used public domain data regarding the nuclear and mechanical design of fuel and core components from Sweden's commercial nuclear power program. Clab conditions of storage were assessed regarding design basis fuel and core components as described in [EPRI, 2010] and using pool conditions described in [Gustafsson and Hagberth, 1978] and [Strasser et al., 2008]. Compliance with industry water

chemistry guidelines was assumed [Svensk Kärnbränslehantering A, 2011]. This assessment identified potential threats to fuel and core component structural integrity. The team reviewed public domain industry literature for comparison and benchmarking purposes regarding material stored, conditions of storage, and surveillance methods. The authors have developed surveillance plans to identify and track the rate of degradation to confirm performance expectations.

The key mechanism that would lead to degradation of spent fuel and core components stored at Clab was determined to be corrosion and corrosion-related phenomena. No additional engineering measures were identified during the course of the study that may further mitigate or eliminate identified structural degradation threats due to corrosion.

The authors concluded that spent fuel and core components may be safely stored at Clab for up to fifty years. The key assumptions leading to that conclusion include:

1. Water quality at the Clab storage pools is consistent with the [Svensk Kärnbränslehantering AB, 2011] guidelines in a surveillance program provided in this report.

2. Sufficient water circulation exists in the Clab storage pools to ensure adequate mixing and prevention of areas of stagnation.

3. The spent fuel and core components are as described in [EPRI, 2010].

4. Clab operators implement a water chemistry and material surveillance program similar to that provided in this report.

5. Spent fuel and core components are properly handled.

Relevance

The design and operation of Clab is an integral component of Sweden's management of high-level nuclear waste generated in the course of electricity production. The choice of Clab as an interim pool storage facility allows the ability to monitor and inspect the integrity of spent fuel and core components throughout the interim storage period. This ensures the ability to address public and regulatory issues and concerns up to the point of final disposition of the items in the repository. It is hoped that this report become an integral part of SSM's approach to the safe storage of spent nuclear fuel and core components from Sweden's commercial nuclear power industry.

Need for further research

This report may be considered complete. As Clab operates, data from water chemistry monitoring program, material surveillance program and periodic inspection of representative fuel and core components may identify areas of further research and analysis.

It is recommended that operating experience from LWR storage pools be reviewed periodically for relevance to Clab.

Project information

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Project reference: SSM2020-7419

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Summary

The A.N.T. International team used publicly available data regarding the spent fuel and core components to be stored at the Clab facility. The key reference regarding fuel and core components to be stored at Clab was "Spent Nuclear Fuel for Disposal in the KBS-3 Repository", Svensk Kärnbränslehantering AB, SKB Technical Report TR-10-13, December 2010.

For the purposes of alignment with fuel and core component data associated with the planned repository, the representative fuel assessed in this report were the SVEA 96 Optima 2 (BWR) and AREVA 17x17 (PWR). Public information regarding the materials and Clab pool storage conditions was also used. The key references regarding Clab pool materials, configuration and operation were "A Central Spent Fuel Storage in Sweden", B. Gustafsson, R. Hagberth, Swedish Nuclear Fuel Supply Co, SKBF, AB Kärnkraft, AKK, 1978, Clab general description, Swedish Nuclear Fuel and Waste Management Co, February 2010.

From this data, fuel and core component degradation mechanisms were identified. The key degradation mechanisms associated with Clab storage were determined to be associated with corrosion. Corrosion-related degradation mechanisms were identified to be: hydriding of zirconium and stainless alloys, general oxidation of zirconium and stainless alloys, localized oxidation of zirconium and stainless alloys, stress corrosion cracking of stainless alloys, galvanic corrosion of zirconium and stainless alloys, crevice corrosion of stainless steel alloys and CRUD-assisted corrosion of zirconium alloys. Of the identified degradation mechanisms, most were ruled out leaving galvanic and localized corrosion as potential long-term Clab storage structural degradation mechanisms. The full report documents the rationale used by the authors.

The degradation of the spent nuclear fuel is of largest concern since release of fission and activation products into the Clab pool storage facilities due to this degradation would increase the risk and operational difficulties at Clab.

These degradation mechanisms were assessed and compared with industry experience regarding pool storage. The corrosion assessment concluded that long-term storage of spent nuclear fuel and core components in the Clab facility may be considered safe with insignificant potential for degradation that would further impact fission and activation product release or adversely affect structural integrity.

These conclusions are based upon review of Clab design basis fuel and core components, their associated materials, the arrangement of specific elements of the fuel and core components and the assumption that the Clab pool chemistry is routinely monitored for compliance with industry standards and recommendations. From the assessment and industry operating experience review, a fuel material and core component surveillance program was developed.

The surveillance program provided in this report consisted of four key

elements which were: pool water chemistry monitoring, pool liner (pool inner stainless steel barrier) inspection and cleaning, material surveillance coupon examination and fuel and core component inspection.

1. Pool chemistry monitoring should be in compliance with "Good Practices for Water Quality Management in Research Reactors and Spent Fuel Storage Facilities", IAEA Nuclear Energy Series, No. NP-T-5.2, July 2011 augmented to include lead and fluorine.

2. Pool liner inspections should be performed to identify accumulation of dirt and/or sludge. If identified, this debris should be removed.

3. Corrosion monitoring should be a combination of periodic inspection of material surveillance coupons and;

4. Periodic inspections of designated representative BWR and PWR fuel assemblies and PWR core components.

Sammanfattning

A.N.T. International har granskat den mest relevanta litteraturen relaterad till mellanlagring av det använda kärnbränslet och härdkomponenter vid CLAB i väntan på att slutförvaret ska bli klart att tas i drift. I vår rapport har vi utgått från att SVEA 96 Optima 2 (BWR) och AREVA 17x17 (PWR) är det mest representativa bränslet. Den viktigaste referensrapporten för vår rapport har varit "Spent Nuclear Fuel for Disposal in the KBS-3 Repository", Svensk Kärnbränslehantering AB, SKB Technical Report TR-10-13, december 2010. En annan viktig referenslitteratur har varit "A Central Spent Fuel Storage in Sweden", B. Gustafsson, R. Hagberth, Swedish Nuclear Fuel Supply Co, SKBF, AB Kärnkraft, AKK, 1978 och Clab general description, Swedish Nuclear Fuel and Waste Management Co, februari 2010.

Fokus på vår studie har varit att identifiera potentiella skademekanismer för kärnbränslet eftersom frigörelse av radioaktiva klyvnings- och aktiveringsprodukter får störst konsekvens.

De viktigaste potentiella degraderingsmekanismerna för det använda kärnbränslet och övriga reaktorkomponenter i Clab bedömdes vara relaterade till korrosion som följer:

- Väteförsprödning av Zr legeringar och rostfritt stål,

- Allmän och lokaliserad korrosion av Zr legeringar och rostfritt stål,

- Spänningskorrosion av rostfritt stål,

- Galvanisk korrosion av Zr legeringar och rostfritt stål,

- Spaltkorrosion av rostfritt stål och

- CRUD relaterad corrosion av Zr legeringar

Degradering av det använda kärnbränslet skulle få störst konsekvens för driften av Clab eftersom frigörelse av klynings- och aktiveringsprodukter från det degraderade bränslet skulle kunna leda till driftstörningar i Clab och senare vid inkapsling av bränsle i Clink.

Dessa degraderingsmekanismer utvärderades och jämfördes med industrierfarenhet av lagring i vattenbasäng. Författarna har dragit slutsatsen att långtidslagring av använt kärnbränsle och härdkomponenter i Clab-anläggningen kan anses vara säker med obetydlig risk för degradering till den grad att klyvnings- och aktiveringsprodukter skulle frigöras från det använda bränslet eller påverka den strukturella integriteten negativt.

Dessa slutsatser baseras på genomgång av Clabs konstruktion för bränsle och härdkomponenter med de ingående materialen, och antagandet att Clabs bassängkemi regelbundet övervakas för att uppfylla branschstandarder och rekommendationer. Utifrån bedömningen och genomgången av branschens driftserfarenhet har författarna tagit fram ett program för övervakning av material, bränsle och härdkomponenter.

Det framtagna övervakningsprogrammet bestod av fyra delar: övervakning av

bassängvattenkemi, inspektion och rengöring av bassängliner, undersökning av övervakningskuponger för material och inspektion av BWR och PWR bränsle och härdkomponenter.

Övervakning av bassängvattenkemi bör överensstämma med "Good Practices for Water Quality Management in Research Reactors and Spent Fuel Storage Facilities", IAEA Nuclear Energy Series, No. NP-T-5.2, juli 2011, utökad för att inkludera både bly och fluor.

Inspektion av bassänglinern bör utföras för att identifiera deponering av smuts och/eller slam. Om deponering har skett bör detta tas bort.

Övervakning av korrosion bör vara en kombination av periodisk inspektion av övervakningskuponger för material och;

Periodiska inspektioner av utvalda representativa BWR- och PWR-bränsleknippen och härdkomponenter

Abbreviations and notations

A.N.T. INTERNATIONAL	Advanced Nuclear Technology, International
appm	atom parts per million
BWR	Boiling Water Reactor
CILC	CRUD-Induced Localized Corrosion
CRUD	Chalk River Unidentified Deposit
DHC	Delayed Hydride Cracking
eV	electron volt (unit of energy gained or lost by an electron when subjected to an electric potential of one volt)
HM	Heavy Metal (uranium and/or plutonium)
IASCC	Irradiation Assisted Stress Corrosion Cracking
IGSCC	Inter-Granular Stress Corrosion Cracking
kgU	Kilograms Uranium
LWR	Light Water Reactor
MeV	Million electron Volts
MOMOs	Manganese Oxidizing MicroOrganisms
MOX	Mixed Oxide (fuel, contains plutonium and uranium)
MIC	Microbially Induced Corrosion
MTHM	Metric Tons Heavy Metal
MTU	Metric Tons Uranium
MWd	Mega-Watt days
NSSS	Nuclear Steam Supply System
PCI	Pellet-Clad Interaction
PHWR	Pressurized Heavy Water Reactor
ppm	parts per million
PWR	Pressurized Water Reactor
IAEA	International Atomic Energy Agency
SCC	Stress Corrosion Cracking
SKB	Svensk Kärnbränslehantering AB
SSM	Swedish Radiation Safety Authority
U	Uranium
UOX	Uranium Oxide (fuel)

Contents

1. Introduction

Sweden is managing the spent nuclear fuel and core components from its commercial nuclear reactors with a national strategic plan. The spent fuel and core components are initially stored at the reactor site.

After an on-site storage period to comply with transportation limits on decay heat and radiation, these materials are shipped to the Clab interim storage facility. The Clab facility operates several linked underground storage pools which are expected to have a storage capacity of 11,000 tonnes. Spent fuel and core components may reside at Clab for up to fifty years prior to final disposition or disposal in Sweden's repository.

During the storage period, aging and structural degradation may compromise the integrity of the fuel and core components to be packaged in repository containers. The release of fission and activation products into the Clab pool storage facilities due to this degradation would increase the risk and operational difficulties at Clab and later with encapsulation of spent fuel at Clink.

The Clab facility design and operation leverage the body of data demonstrating the safety and efficacy of pool storage of spent nuclear fuel and core components. Pool storage allows direct and indirect monitoring of the storage conditions and inspection of the materials in storage up to the point of final encapsulation in disposal containers.

The interested reader of Swedish Interim and Final Storage of Nuclear Waste is referred to two excellent publications by SKB, "Environmental Impact Statement Interim storage, encapsulation and final disposal of spent nuclear fuel", ISBN 978-91-978702-5-2, 2011, "Design, production and initial state of the canister", SKB Technical Report, TR-10-14, 2010 and "Spent nuclear fuel for disposal in the KBS-3 repository", SKB Technical Report, TR-10-13, 2010.

The structure of this report is as follows:

Sections 2 and 3 provide the objectives of this report and the assumptions and limitations that the A.N.T. International team worked within,

Section 4 is the core of the report and gives an overview of the Swedish management of spent nuclear fuel with the focus of management of fuel and core components to be stored at Clab as well as potential material degradation mechanisms.

Section 5 provides a surveillance program.

Sections 6 and 7 provide a summary/conclusions as well as outlooks and future activities

2. Objectives

The objectives of this work include:

1. Identify fuel and core component materials and their configurations. A particular emphasis is paid to dissimilar metal combinations and gaps/crevices.

2. Identify structural degradation mechanisms in spent nuclear fuel and core components that may adversely impact the integrity of long-term (> 50 years) storage of spent fuel and core components in the Clab facility pools. If this will be the case:

   a. The operation of the Clab pool would become problematic should degradation of the spent nuclear fuel occur to such an extent that radioactive species are released into the storage pools.
   b. The subsequent handling of the spent nuclear fuel and core components such as encapsulating of the spent fuel in copper canisters and treatment of core components could be more complicated.

3. Assess the degradation mechanisms with respect to the materials that make up the fuel and core components stored at Clab.

4. Provide surveillance and inspections that may identify the onset and track the progression of the degradation mechanisms.

3. Focus of research and key assumptions and limitations

The A.N.T. INTERNATIONAL team worked within the following assumptions and limitations:

1. The authors performed the study utilizing information available in the public domain.

2. Spent fuel that was failed or damaged because of reactor operation was placed in failed fuel isolation containers.

3. The Clab pool water chemistry complied with industry standards such as "Good Practices for Water Quality Management in Research Reactors and Spent Fuel Storage Facilities", IAEA Nuclear Energy Series, No. NP-T-5.2, July 2011.

4. Spent fuel and core components are properly handled.

The A.N.T. INTERNATIONAL team had the following key focus areas in pursuit of the objective of the study:

1. Identify the spent fuel and core components expected to be stored at the Clab facility.

2. Identify the materials and juxtaposition of these materials in the fuel and core components and the Clab pool storage canister.

3. Review industry research and operating experience regarding long-term pool storage.

4. Utilize the education and experience of the authors and peer reviewers

4. Results and Discussion

4.1 Management of Radioactive Waste in Sweden

4.1.1 Introduction

The power plant owners are responsible for safe management and disposal of the radioactive waste and have jointly established SKB[1] for this purpose. Today there is a final repository for short-lived radioactive waste (SFR) and a central interim storage facility for spent nuclear fuel (Clab), Figure 1.
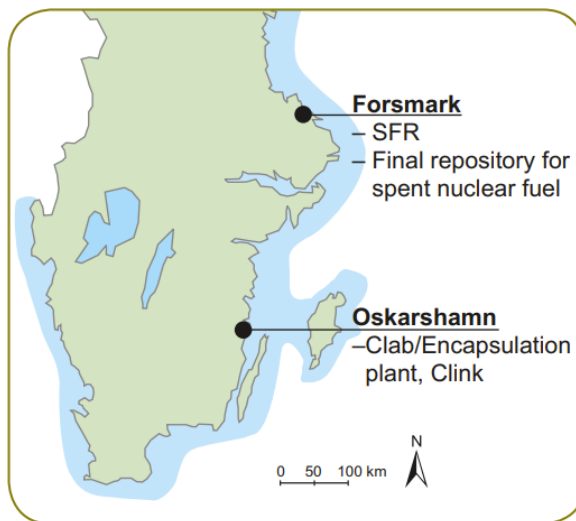
Figure 1: SKB is applying for an encapsulation plant adjacent to Clab. SKB has recently been granted a license to build a final repository in Forsmark [Svensk Kärnbränslehantering AB, 2011]

Most of the radioactive waste volume from the nuclear power plants consists of short-lived low- and intermediate level waste, Table 1. The short-lived low- and intermediate level waste is disposed of in SFR (final repository for short-lived radioactive waste) in Forsmark. The spent nuclear fuel is interim-stored in Clab in Oskarshamn.

The radioactive waste from the nuclear power plants is divided into different categories dependent on the level and duration of the radioactivity, Table 1.

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[1] https://skb.com/

Table 1: Different types of radioactive waste [Svensk Kärnbränslehantering AB, 2011]

	Short lived waste	Long lived waste	Low level waste	Intermediate level waste	High level waste
Duration of reactivity	Radioactivity declines to the level that occurs naturally within 500 years (e.g. protective clothing and tools)	Radioactivity may take around 100,000 years before it is at the level of naturally occurring uranium ore (e.g. spent nuclear fuel)			
Level of reactivity			Radioactive waste that can be handled directly without cooling or radiation shielding. For example protective clothing, tools, filters and other items that may have been contaminated with radioactivity.	Radioactive waste that requires radiation shielding, but not cooling, for handling. For example ion exchange resins	Waste which, due to its high radioactivity and heat output, requires both cooling and radiation shielding, for example spent nuclear fuel.

The system for transportation of the various waste types from the nuclear power plants to the waste facilities is shown in Figure 2.
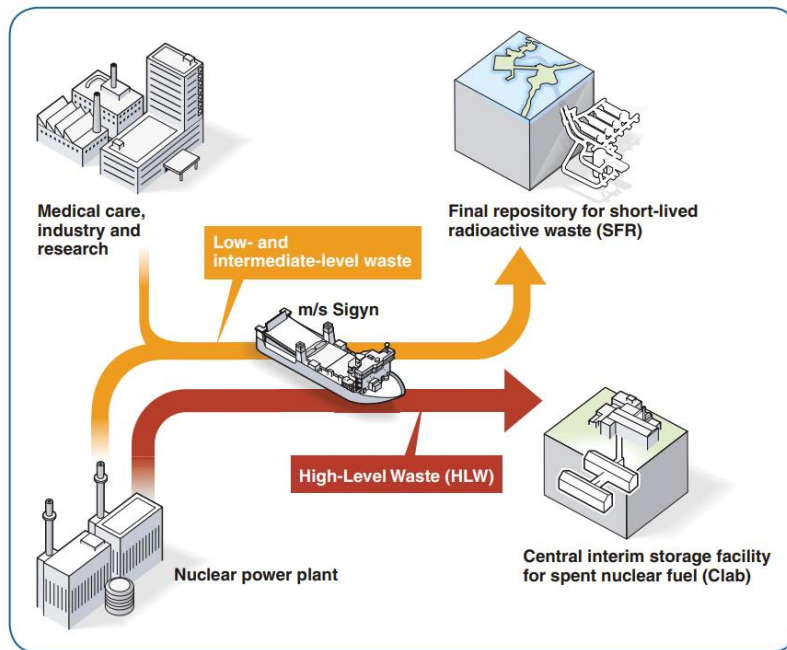
Figure 2: Existing facilities in the Swedish system for management of radioactive waste [Svensk Kärnbränslehantering AB, 2011].

The risks associated with spent nuclear fuel can be described in terms of radiotoxicity and accessibility [Svensk Kärnbränslehantering AB, 2011].

- Radiotoxicity describes the harm which the ionizing radiation from the

radionuclides can cause if people are exposed to it and is dependent on the level of activity and the type of radiation the radioactive decay gives rise to.

- Accessibility describes the degree to which a person can be exposed to radiation in different situations, for example during transport, interim storage or final disposal.

Whenever the spent nuclear fuel is handled, its accessibility is limited:

- during transport by means of special transport casks, and
- during storage periods by keeping the fuel in water pools.

The transport casks shield off the radiation and dissipate heat. The water in the storage pools at the nuclear power plants and Clab cools the fuel and shields off the radiation emitted by the fuel.

The radioactivity of the fuel increases sharply during the operation of a reactor. After about five years of use, the fuel is taken out of the reactor and is then at its peak radiotoxicity. After discharge, the quantity of radionuclides, and thereby the radiotoxicity of the fuel, declines as they decay. The spent nuclear fuel is first stored for some time in pools at the power plants, after which it is interim stored in Clab awaiting encapsulation and final disposal. The radioactivity decreases sharply during the Clab storage time, Figure 3.
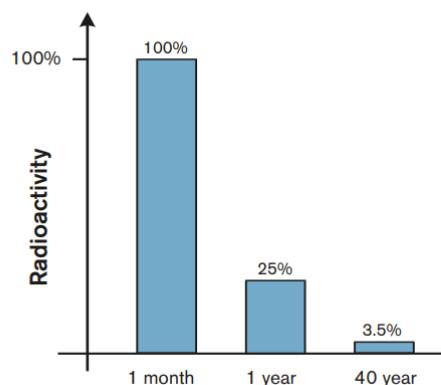
Figure 3: Typical decrease of radioactivity in a tonne of spent nuclear fuel, modifed Figure 3 according to [Svensk Kärnbränslehantering AB, 2011].

4.1.2 Clab

After an on-site storage period in the pool at the nuclear power plant to comply with transportation limits on decay heat and radiation, these materials are shipped to the Clab interim storage facility in accordance with applicable regulations for transports of radioactive material [Svensk Kärnbränslehantering AB, 2010].

During interim storage at Clab, the spent fuel is kept in deep storage pools about 30 metres below ground in the rock, Figure 4 and Figure 5. Covering the fuel with eight metres of water shields against radiation and cools down the hot fuel. In 2022 SKB received the permit to expand Clab capacity to 11.000 tonnes [World Nuclear News, 2022].
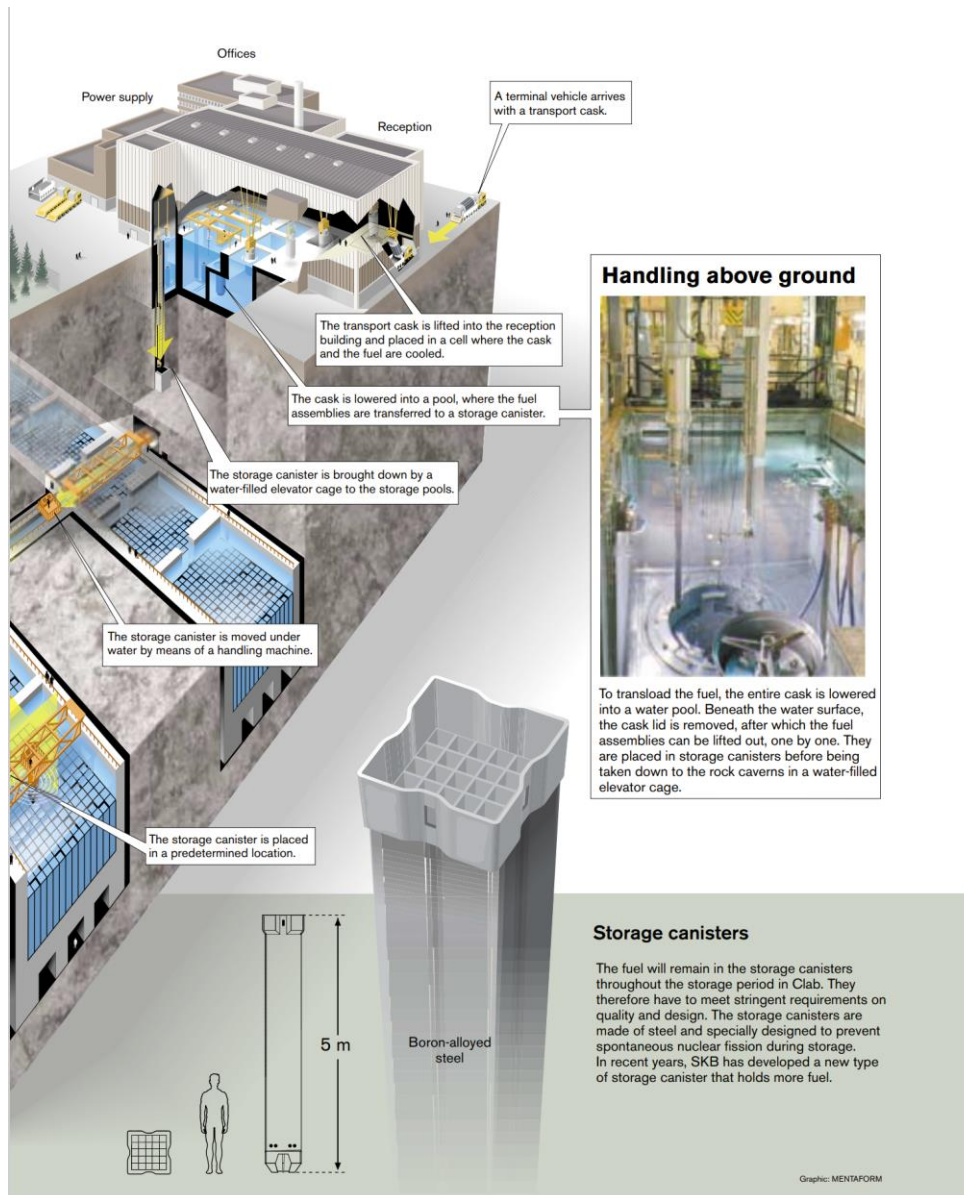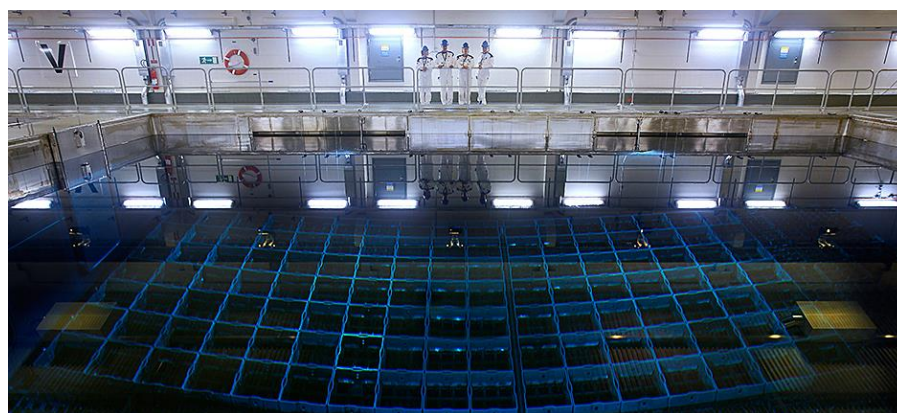
**Figure 4:** Schematics of Clab[2]

Figure 5: Spent fuel storage pool at Clab (Source: SKB – "Clab – Central Interim Storage Facility for Spent Nuclear Fuel", Photo Curt-Robert Lindqvist)

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[2] https://www.skb.se/publikation/1109049/Clab.pdf

In the current operating scenario for the nuclear power plants, the average storage period will be about 45 years. Besides spent nuclear fuel, the radioactive operational waste to which Clab gives rise is also managed in the facility. However, it is not stored but is packaged and sent to SFR. Core components are also stored temporarily in Clab awaiting transport to suitable final disposal. During this storage period, aging and structural degradation may compromise the integrity of the fuel and core components to be packaged in repository containers. The release of fission and activation products into the Clab pool storage facilities due to this degradation would increase the risk and operational difficulties at Clab.

When the transport arrives at Clab (later Clink), the transport cask is inspected and transported to a cell for cooling. When the temperature has stabilised, the transport cask is transported to the cask pool and connected to the unloading pool. The fuel assemblies are unloaded and placed in the storage canisters in which they are kept during the interim storage. **If the assemblies are damaged, or suspected to be leaking, the storage canister is transported to a service pool and the assemblies are searched for damage. If necessary, damaged assemblies are put in protective boxes and placed in special storage canisters.**
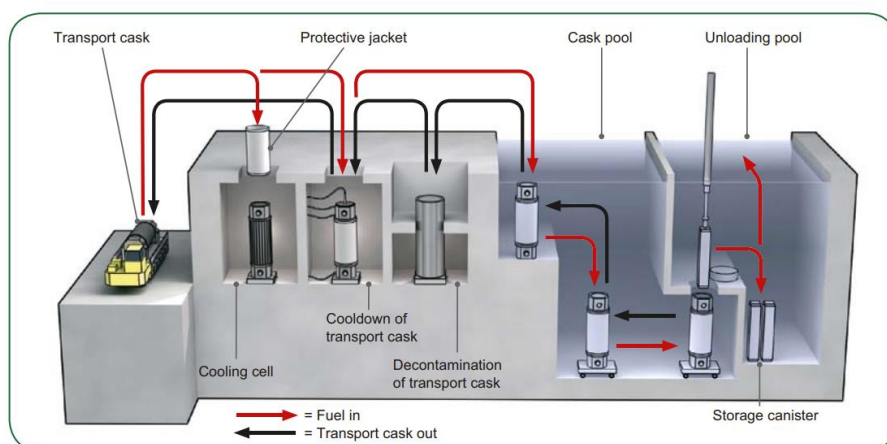
Figure 6: Handling of transport casks in Clab's reception building [Svensk Kärnbränslehantering AB, 2011].

However, since several years back, all failed fuel, Figure 7, identified at the nuclear power site is loaded into a protection box, see Figure 8.

Figure 7: Leaking fuel rod (left), Damaged rod (right)

Figure 8: Westinghouse Quiver (Left: PWR version - Right: BWR version)

When the placement and registration of assemblies in the storage canister is finished, the storage canister is transported to a fixed position with a unique identity in one of the storage pools. The assemblies are stored until they are selected for encapsulation, see Section 4.1.3.

Table 2 shows that the actual concentration of radioactivity in Clab is far below the licensing values. Table 2 shows the calculated concentrations of Co-60 and Cs-137 in the storage pools and the cleanup systems, along with measured values for comparison. The table includes Co-60 and Cs-137 because they are the two most important radionuclides in the releases from the facility. The table shows that measured concentrations in pools, and radioactivity quantities that are collected in the cleanup systems, are far below the licensing calculations. The input of activity is expected to increase as the amount of fuel stored in the pools increases, but at the same time the decay of certain nuclides will offset the increasing storage.

Table 2: Pessimistic calculations and measured concentrations of radioactivity in storage pools and cleanup systems [Svensk Kärnbränslehantering AB, 2011].

Nuclide	Radioactivity concentration in the storage pools [Bq/m³]	Radioactivity in cleanup systems [Bq/year]
Co-60 calculated for maximum storage (8,000 t U)	$8.9 \cdot 10^7$	$5.1 \cdot 10^{14}$
Co-60 measured average 2000-2009	$1.8 \cdot 10^6$	$1.1 \cdot 10^{13}$
Cs-137 calculated for maximum storage (8,000 t U)	$2.1 \cdot 10^8$	$1.2 \cdot 10^{15}$
Cs-137 measured average 2000-2009	$2.2 \cdot 10^5$	$1.2 \cdot 10^{12}$

During the storage period, aging and structural degradation may compromise the integrity of the fuel and core components to be packaged in repository containers. The release of fission and activation products into the Clab pool storage facilities due to this degradation would increase the risk and operational difficulties at Clab.

The Clab facility design and operation leverage the body of data demonstrating the safety and efficacy of pool storage of spent nuclear fuel and core components. Pool storage allows direct and indirect monitoring of the storage conditions and inspection of the materials in

storage up to the point of final encapsulation in disposal containers.

4.1.3 Clink

SKB has chosen Simpevarp in Oskarshamn Municipality as the site for the encapsulation plant since this makes it possible to exploit the personnel's experience of fuel handling at the same time as many of the existing systems and plant parts in Clab can be utilized for the encapsulation plant as well.

The main activity during the operation of the encapsulation plant is to enclose spent nuclear fuel in tightly sealed copper canisters. The copper canisters, which are five meters long and have a diameter of about one meter, arrive ready-made at the encapsulation plant. The different steps in the encapsulation process are shown in Figure 9.
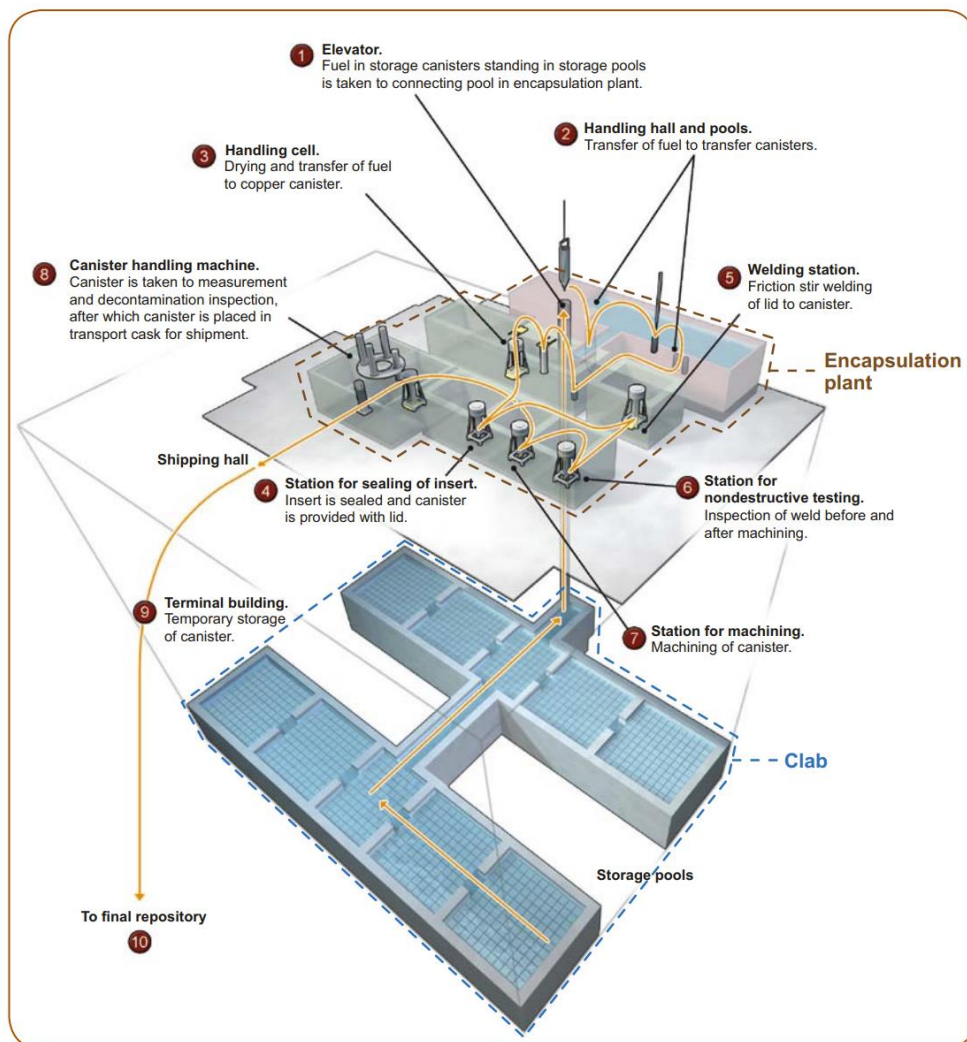
Figure 9: The different steps for handling of spent nuclear fuel in the encapsulation plant [Svensk Kärnbränslehantering AB, 2011]

Figure 10 shows the different steps transported nuclear fuel from the Clab storage pools goes through during the process the canister is moved in a remote-controlled transport frame between workstations in the plant where different steps are performed. During the entire process the canister is covered by a radiation shield. The end product is a

filled copper canister placed in a transport cask and prepared for transport to the final repository, see Figure 11.
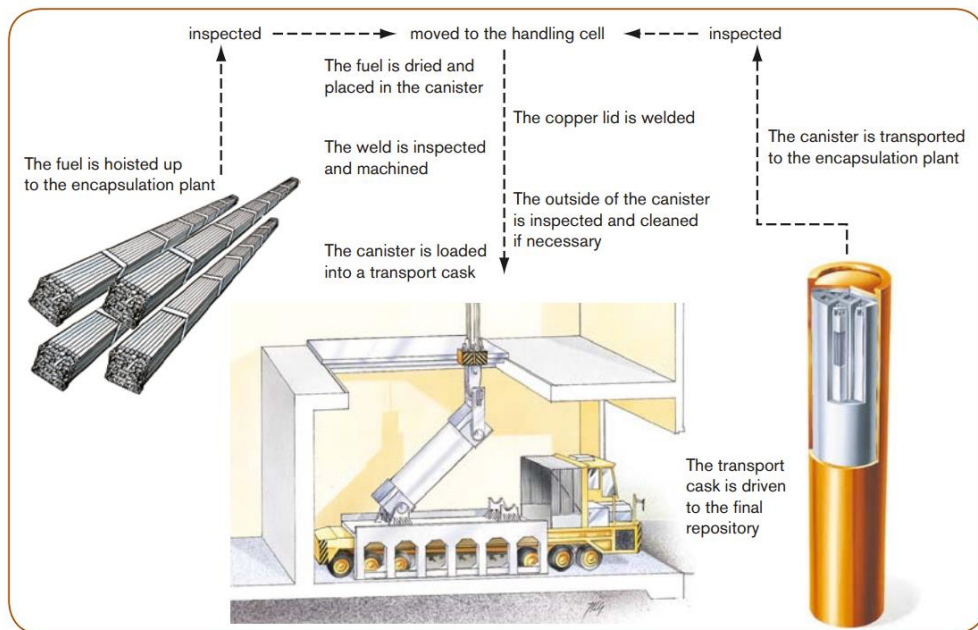
Figure 10: The filled copper canister is placed in a transport cask prior to transport to the final repository [Svensk Kärnbränslehantering AB, 2011]
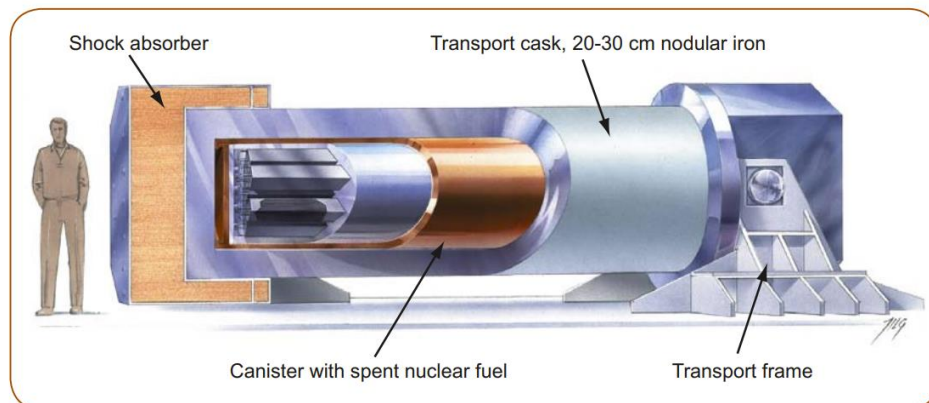
Figure 11: Canister transport cask and transport frame [Svensk Kärnbränslehantering AB, 2011].

Figure 12 shows how the spent nuclear fuel is encapsulated in copper canisters which are then emplaced, surrounded by a buffer of bentonite clay, in deposition holes in a tunnel system at a depth of about 500 metres in the bedrock.
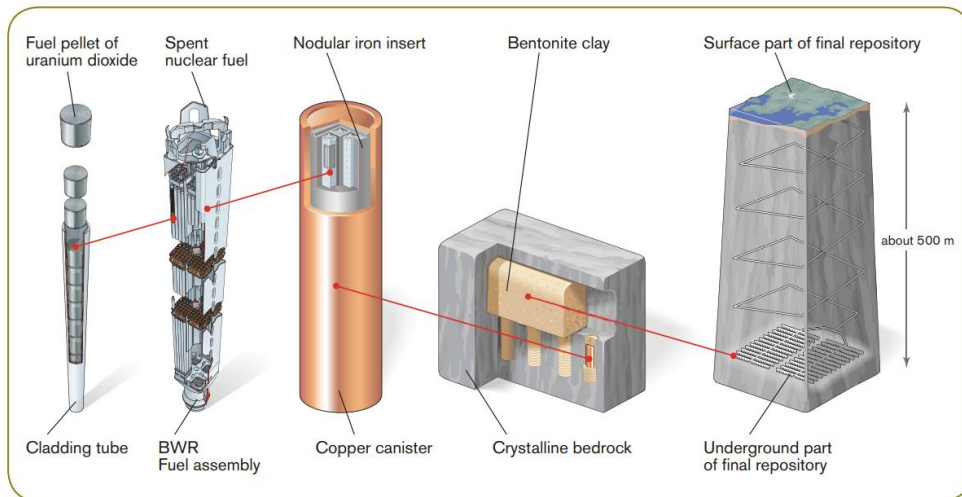
Figure 12: The KBS-3 method [Svensk Kärnbränslehantering AB, 2011].

The final repository for spent nuclear fuel consists of a surface (above-ground) part and an underground part, see Figure 13.
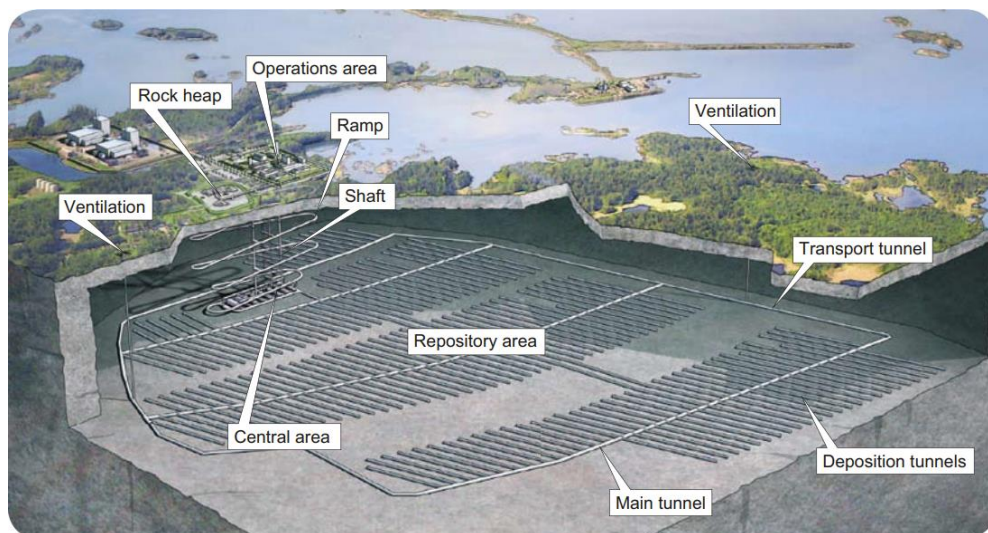

Figure 13: The final repository with surface and underground parts [Svensk Kärnbränslehantering AB, 2011].

4.2 Fuel and core components to be stored at Clab

4.2.1 Introduction

The dominant source of spent fuel and core components to be stored at Clab are Swedish commercial Light Water Reactors. These include BWRs Barsebäck (B1 and B2), Forsmark (F1, F2 and F3), Ringhals (R1) and Oskarshamn (O1, O2 and O3) and PWRs Ringhals (R2, R3 and R4). The locations of Sweden's commercial nuclear power plants and Clab are provided in Figure 15. The fuel used in these reactors have a maximum enrichment of 5% U-235. The maximum assembly average burnup of UOX is 60 MWd/kgU for both PWR and BWR fuel. The maximum assembly average burnup of MOX is 50 MWd/kg HM for only BWR fuel.
Table **3** provides details regarding the volume and type of fuel to be stored at Clab.

Table 3: Fuel Storage at Clab

Fuel Type	Total Expected Assemblies	Total Initial MTU or MTHM	Notes
BWR (Barsebäck, Oskarshamn, Forsmark, Ringhals)	47,498	8,312	• Includes 86 BWR MOX • Fuel channels removed from 1,520 assemblies • Includes individual rods placed in storage cassettes • Assumed 175 kg U/HM per assembly
PWR (Ringhals)	6,016	2,791	Assumed 464 kg U per assembly
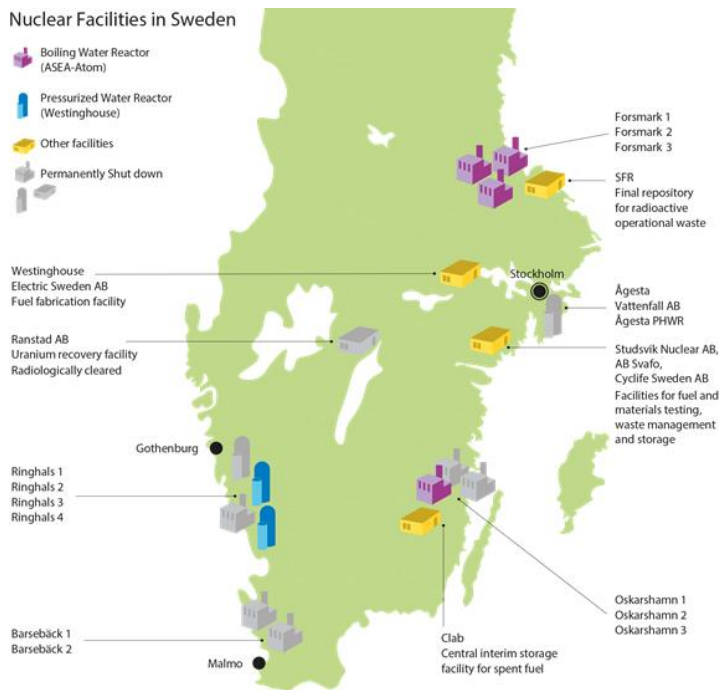
Figure 14: Sweden's nuclear power facilities [Sweden's 8th National Report under the Convention on Nuclear Safety, 2019]

While the vast bulk of spent fuel and core components may be characterized as typical LWR spent fuel, there are miscellaneous fuel assemblies from older reactors, reprocessed fuel, research facility waste and damaged fuel. Table 4 provides details regarding the miscellaneous spent fuel and high-level radioactive waste to be stored at Clab.

The A.N.T. INTERNATIONAL team determined that for the purposes of consistency with the design and licensing of Clab and Sweden's repository that degradation assessment would focus on the design basis fuel assemblies and core components cited by "Spent Nuclear Fuel for Disposal in the KBS-3 Repository", Svensk Kärnbränslehantering AB, SKB Technical Report TR-10-13, December 2010 and "A Study of Availability of Fuel Data for Sweden's Spent Nuclear Fuel Main Review Phase", Henrik Lindhl, ES-konsult, Swedish Radiation Safety Authority, December 2013.

Table 4: Miscellaneous Fuel at Clab [Svensk Kärnbränslehantering AB, 2010].

Fuel Type	Total Expected Assemblies	Total Initial MTU or MTHM	Notes
Ågesta PHWR Assemblies	222	20	• 1 assembly is unirradiated • Burnup range 0 to 10 MWd/kgU • ~70% natural enrichment • ~30% 1.35% enriched • 1 assembly enriched to 2.2%
BWR "Swap MOX"	184	14.1	• Average Burnup 16 MWd/kg HM • Higher transuranic content than UOX
PWR "Swap MOX'	33	8.4	• Average Burnup 31 MWd/kg HM • Higher transuranic content than UOX
Studsvik residue	~25	~3	• In protection boxes, PWR outer dimensions • ~ 125 kgU/box • 3% average enrichment per box
Damaged fuel*	-	-	• In protection boxes • Fuel with leaking rods in Clab • Damaged fuel currently stored at reactor sites
* All damaged nuclear fuel is either put in protection boxes at the nuclear power plant or at the checking of fuel rod integrity once a container with spent nuclear fuel has arrived at Clab.			

The representative fuel assemblies are the SVEA 96 Optima 2 (BWR) and AREVA 17x17 (PWR), Figure 15 and Figure 16. Figure 15 illustrates the SVEA 96 BWR fuel assembly. Figure 16 illustrates the AREVA 17x17 fuel assembly. It should be noted that the AREVA 17x17 fuel assembly does not include a channel or shroud. The key core component of the A.N.T. INTERNATIONAL team assessment was the PWR control rod cluster whose outer surface is composed of stainless steel.

Upon selection of the SVEA 96 Optima 2 (BWR) and AREVA 17x17 (PWR) fuel assemblies, data regarding the types of material alloys that were to be subject to corrosion assessment was retrieved. Table 5 provides details with respect to the materials, relative mass and distribution (which fuel component is composed of a particular material).
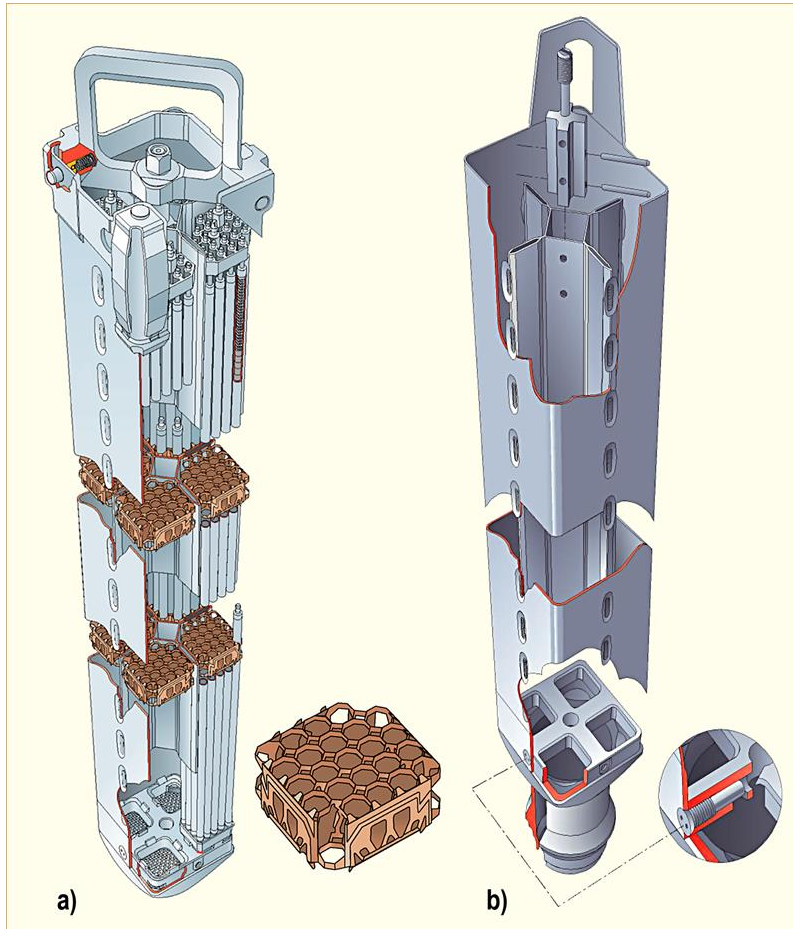
Figure 15: Fuel rod SVEA-96 Optima 2 BWR Fuel Assembly
a) The BWR Fuel SVEA-96 Optima2, provided courtesy of Westinghouse.
b) The BWR SVEA Fuel channel, bottom support plate and transition piece, provided courtesy of Westinghouse.
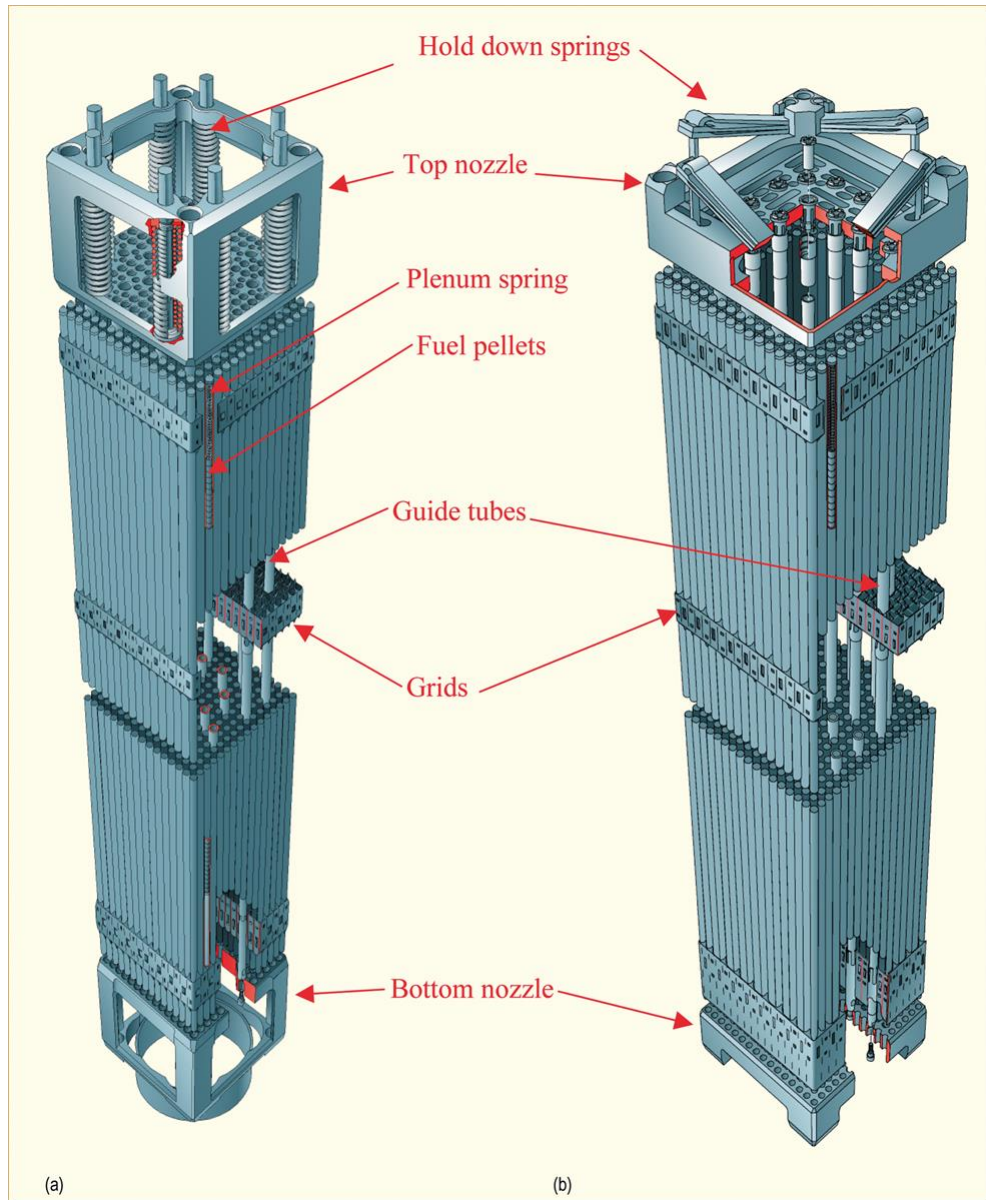
Figure 16: AREVA 17x17 PWR Fuel Assembly
a) German PWRs (16x16, 18x18) and
b) 17x17 design, provided by the courtesy of Westinghouse

In the Swedish PWRs, the fuel 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.

The control assemblies in the Swedish PWRs consist of a set of absorber rods attached to a spider assembly. The rods of each control assembly are designed to enter the guide tubes of the fuel assemblies, consequently the radial spacing of the rods is designed to match the radial spacing of the guide tubes, see Figure 17 where the guide tube cladding is made of 304 stainless steel.

The rods contain the absorber or a combination of absorbers in stainless steel or nickel alloy tubing with end plugs welded to the tubing. The primary absorber used in Swedish PWRs is an extruded alloy rod of 80% Ag, 15% In and 5% Cd (AIC) for the full length of the absorber rod. Some designs use B_4C for the entire rod length, and

some hybrid rods substitute AIC for boron carbide (B_4C) at the high exposure tips to provide a material with a lower swelling rate than B_4C.
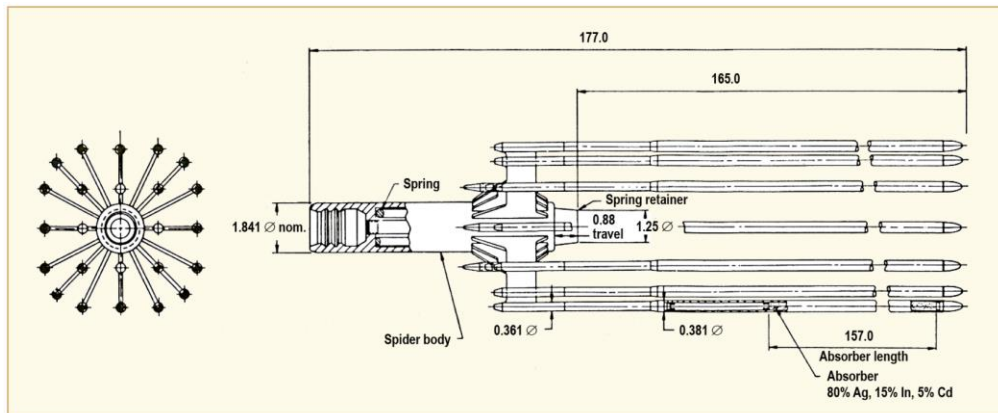
Figure 17: Control Assembly (CA) for Westinghouse 17x17 fuel NSSS.

Materials used in BWR and PWR fuel as well as PWR control assemblies are provided in Table 5, Table 6 and Table 7.

Table 5: Components and material composition for components in a BWR fuel assembly with fuel channel. Table modified according to [Svensk Kärnbränslehantering AB, 2010].

Component	Fuel Assembly						Fuel channel	
	Cladding	Spacer	Springs, part length	Springs, part length	Top plate with handle	Bottom plate	Channel	Channel bottom piece
Material	Zry-2	X-750	AISI 302	AISI 302	AISI 304	AISI 304L	Zry-2	AISI 304L
Weight (kg)	49.3	0.86	0.088	1.26	2.6	0.96	31.5	8.4
Material composition (%)								
Ti	0.0004	2.5	0.01	0.01	0.01	0.01	0.004	0.01
Cr	0.1	16	18.5	18.5	18.5	18.5	0.1	18.5
Mn	0.003	0.2	1.3	1.3	1.3	1.3	0.003	1.3
Fe	0.15	7.00	69.05	69.05	69.08	69.10	0.15	69.1
Ni	0.05	71.99	10.00	10.00	10.00	10.00	0.05	10
Zr*	98.0						98.0	
*The mass given for Zr is the total mass in the fuel assembly, the water cross is included.								

Table 6: Components and material composition for components in a PWR fuel assembly, Table modified according to [Svensk Kärnbränslehantering AB, 2010].

Component	Top nozzle SS	Top nozzle Zr	Bottom nozzle SS	Spacer Zr	Spacer Inconel	Guide thimble Zr	Guide thimble SS	Cladding Zr	Cladding SS
Material	304L	Inc718	304L	Zry4	Inc718	Zry4	316L	M5	302
Weight (kg)	6.5	1.1	4.8	7.2	0.75	14.2	0.7	108.1	3
Material composition (%)									
Ti	0.01	0.9	0.01	0.04	0.9	0.004	0.01	0.004	0.01
Cr	18.5	19	18.5	0.1	19	0.1	17		18.5
Mn	1.3	0.3	1.3	0.003	0.3	0.003	1.3		1.3
Fe	69.05	21.05	69.05	0.22	21.05	0.22	66.25		69.95
Ni	10	52.50	10	0.004	52.5	0.004	12		9
Cu	0.1	0.1	0.1	0.003	0.1	0.003	0.1		0.1
Nb	0.03	4.59	0.03	0.01	4.59	0.01	0.03	1	0.03
Mo	0.2	0.05	0.2	0.0005	0.05	0.0005	2.5		0.2
Sn	0.01	0.01	0.01	1.5	0.01	1.5	0.01		0.01
Zr	0.001			97.97		97.97		98.81	
* The stainless steel in the springs in the fission gas plenum.									

Table 7: Material composition for control rod clusters in PWR fuel assemblies, Table modified according to [Svensk Kärnbränslehantering AB, 2010].

Component	Absorber pins	Absorber pins	Top piece
Material	304	AginCd	304
Weight (kg)	12.5	51.4	3.8
Material composition (%)			
Cr	18.5		18.5
Mn	1.3		1.3
Fe	68.9		68.9
Ni	10		10
Ag	0.0001	80	0.0001
Cd		5	
In		15	

4.2.2 Identification and assessment of corrosion degradation mechanisms

The A.N.T. INTERNATIONAL team identified the mechanisms that have

potential to have an adverse effect on the structural and fission product/ actinide containment integrity of spent fuel and core components that may reside for an extended period in the Clab interim storage facility pools. Since all the identified mechanisms are related to corrosion, it should be noted that most corrosion rates increase with temperature. Measurements of decay heat at Clab show that BWR and PWR fuel stored at Clab will experience significantly lower temperatures within a decade or so. This appears to be especially the case with BWR fuel. Figure 18 shows a plot of decay heat vs. cooling time for BWR and PWR fuel stored at Clab. Therefore, the rate of temperature-driven corrosion degradation should decrease with storage time at Clab.
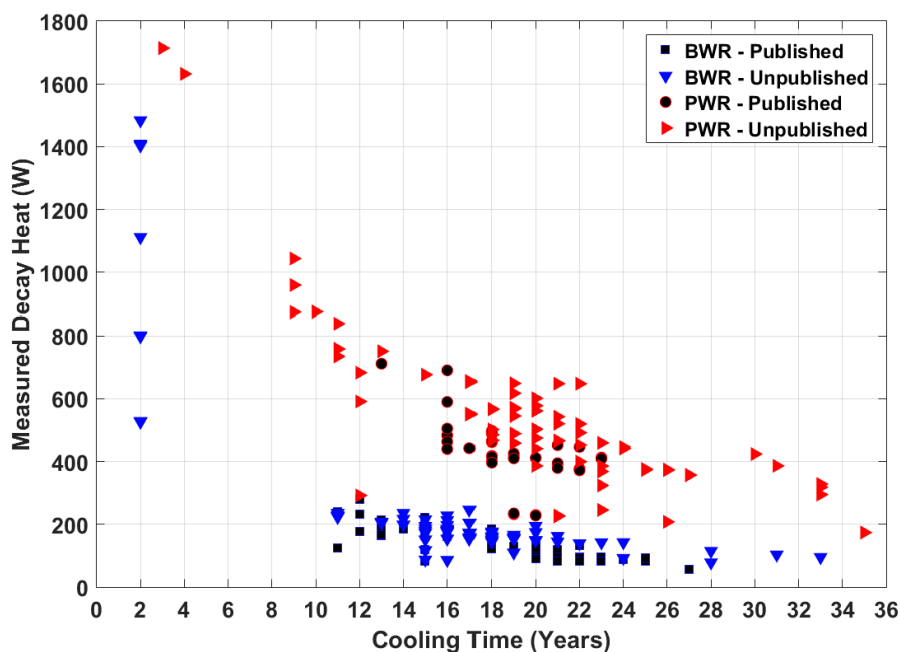

Figure 18: Measured decay heat as a function of cooling time [Svensk Kärnbränsle-hantering AB, 2006] and [PATRAM 22, 2023].

4.3 Potential Material Degradation Mechanisms

The following assessment for each material potential degradation mechanism is based upon operating experience and industry research as well as the direct observations, research and experience of the authors. However, after careful review of the potential degradation mechanisms it turns out that some potential mechanisms can be ruled out.

4.3.1 Material embrittlement

Zirconium Alloy Hydriding
Zirconium Alloys Corrosion Process
Hydrogen generation, absorption and diffusion in the zirconium alloy matrix occur during operation of the fuel and core component at reactor power operation conditions. Oxidation of fuel cladding during reactor operation leads to the release of hydrogen some of which is absorbed by the zirconium alloy.

The absorption of hydrogen (H) by zirconium (Zr) and its alloys is the result of

the corrosion reaction at operating temperatures:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

A fraction of the H produced is absorbed by the Zr metal and the "hydrogen pickup fraction" or "percentage hydrogen uptake" is defined as the ratio of H entering the metal to that formed by the corrosion reaction during the same period, Figure 20.

The H that is absorbed by the metal does so as an integral part of the reaction of Zr with the water molecule rather than by the reaction with any dissolved H in the water.
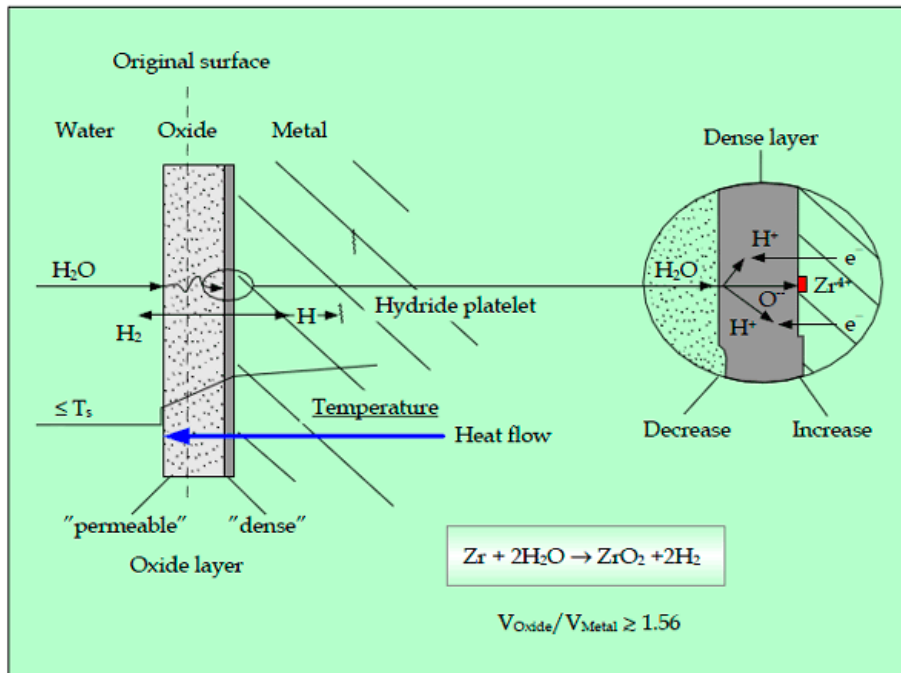
Figure 19: Schematics showing the Zr alloy corrosion and hydrogen pickup process of the fuel cladding, [Cox and Johnston, 1962] and [IAEA TECDOC-996, 1998].

Instantaneous mechanical failure due to brittle hydrides

Typical, discharged fuel average hydrogen concentrations in fuel cladding have been measured to be in the range of 50 to 600 ppm. In very high burn-up fuel, hydrogen concentrations may exceed 1000 ppm. Once the hydrogen solubility limit in zircaloy is reached, hydrides form which cause local embrittlement. This is especially the case in fuel materials subject to dual side corrosion such as BWR water rods and PWR control rod guide tubes. The solubility of hydrogen in Zirconium alloys at pool storage temperatures is less than 1 ppm. Therefore, essentially all hydrogen in LWR fuel cladding in pool storage will exist as hydrides. Irradiated Zirconium alloys containing significant amounts hydrides have very low ductility, Figure 20 and Figure 21.
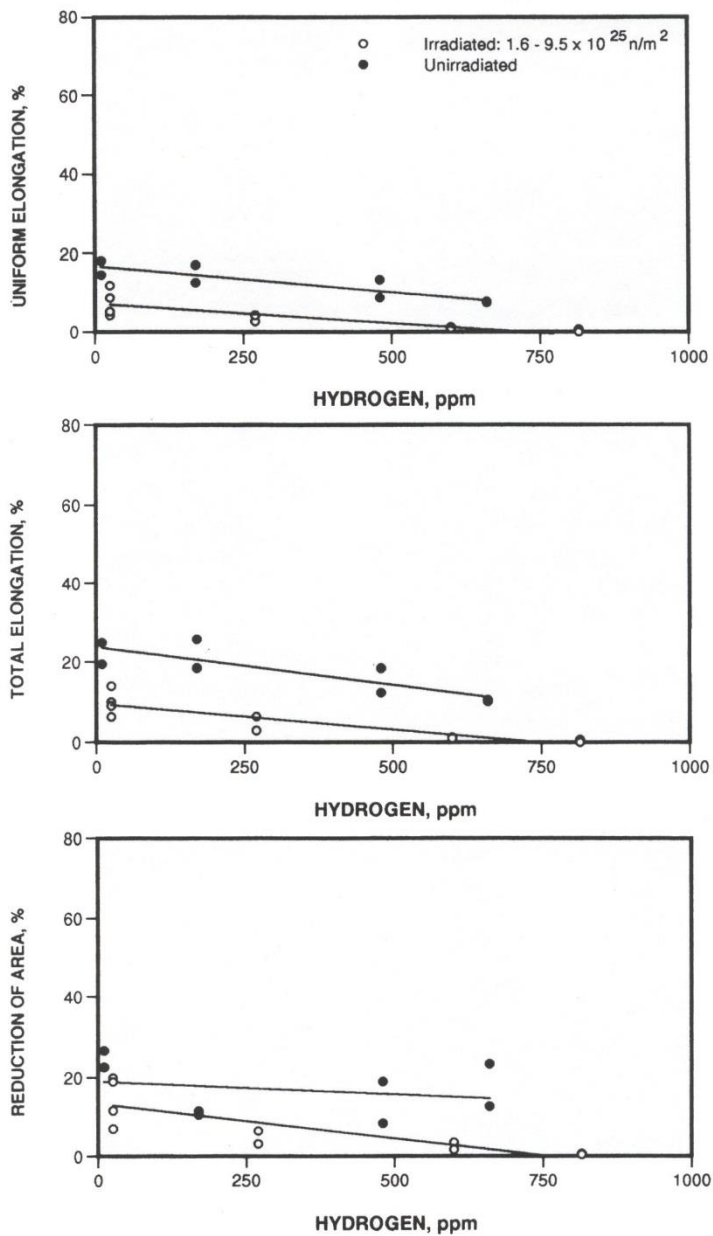
Figure 20: Uniform and total elongation and reduction of area as a function of hydrogen for unirradiated and irradiated Zircaloy-2 LDA specimens tested at 295 K [Wisner and Adamson, 1998].
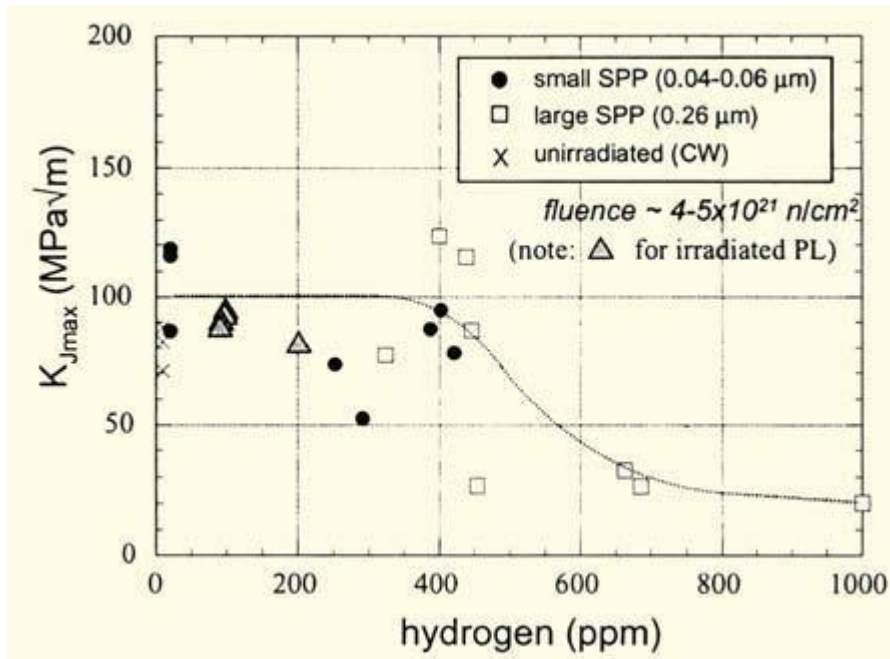
Figure 21: Fracture toughness results from irradiated Zircaloy as a function of hydrogen, using VEC and PLT techniques [Edsinger et al., 2000]

Figure 22 shows an example of hydrides in nuclear fuel cladding. The radially oriented hydrides can provide a path of low ductility for the propagation of cracks through the wall of zirconium alloy tubing.
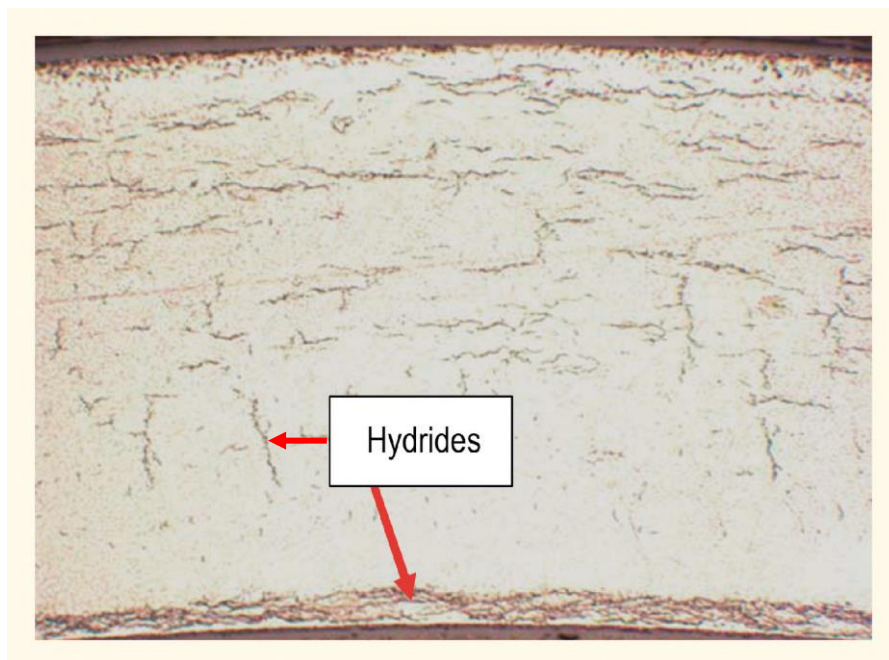
Figure 22: Hydrides in a BWR fuel cladding cross section at mid height position from KKL with an H-content of 220 ppm after a bundle burnup of 60 MWd/kgU [Strasser et al., 2008]

Zirconium hydrides are brittle and their presence represents a threat to the structural integrity of components made from zirconium alloys. The concentration, distribution and orientation of hydrides can have a wide variation but their most damaging form appears by light metallography as

28

platelets after slow cooling, Figure 23.
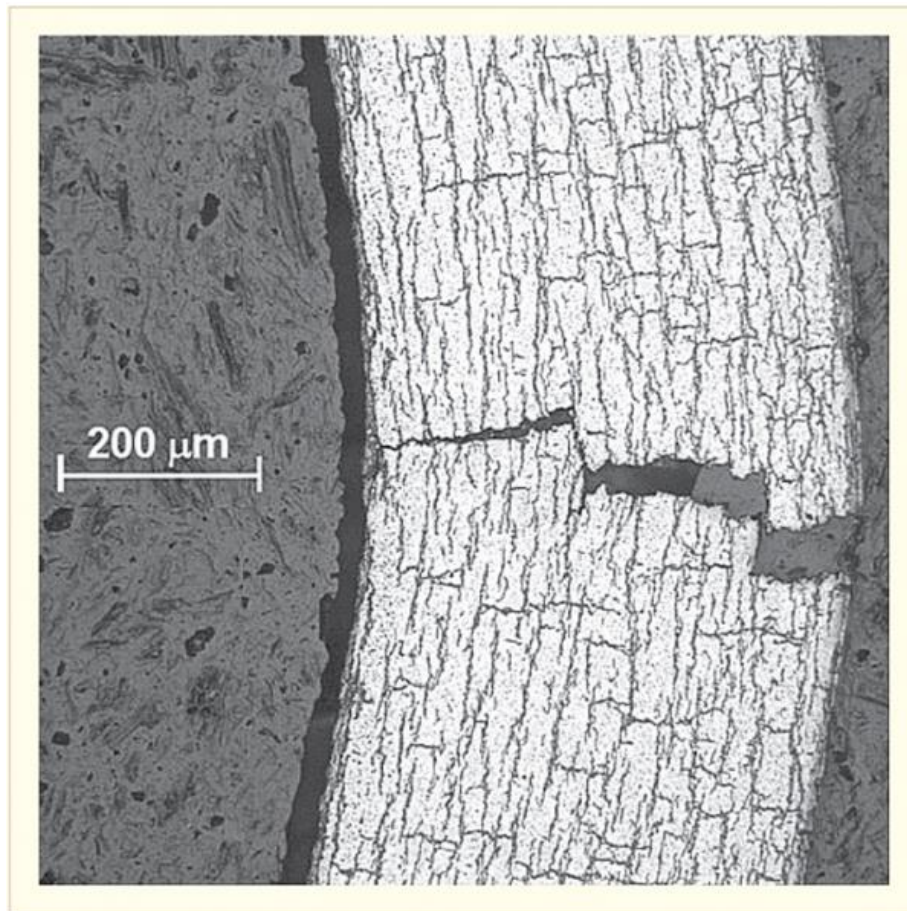
Figure 23: Distribution of hydride platelets on a transverse-radial plane in ZIRLO (Zr1Sn1Nb0.1Fe) fuel cladding containing 630 ppm hydrogen cooled from 400°C with hoop stress of 190 MPa [Billone et al., 2013].

Delayed mechanical failure

One aspect of hydrogen or hydrides in zirconium alloys is the potential for delayed hydride cracking (DHC). DHC is a time-dependent mechanism of crack propagation in zirconium alloys.

It involves the diffusion of hydrogen, but also the microscopic fracture of zirconium hydride, leading to the initiation and propagation of a macroscopic crack. A threshold mechanical loading (stress) for crack initiation is required. If no flaw is present, the process can only start with very large tensile stresses; if a sharp flaw is present in the component, a threshold stress intensity factor must be exceeded. The subsequent crack propagation rate is essentially independent of the applied stress intensity factor.

Under tensile stresses, the hydrides may precipitate with a harmful orientation. Since the hydrides have low toughness and cause embrittlement, in time, cracking in components may be possible. The outcome is either DHC or cracking of hydride blisters or layers. Figure 24 is a schematic illustration of the sequence leading to DHC. For example, assume a surface flaw exists in a zirconium alloy component containing hydrides, (a). Typically, a component

may have hydrides parallel with the surface. In fuel cladding, hydrides tend to have a circumferential orientation. If a tensile stress (σ) is applied, (b), hydrogen diffuses up the stress gradient close to the flaw tip [Dutton and Puls, 1976] and [Dutton et al., 1977]. New hydrides will precipitate at the flaw if sufficient hydrogen accumulates above the solubility limit. In fuel cladding stressed in the hoop direction, the new hydrides will be radially oriented. With time, the flaw tip hydride grows larger and more hydride accumulates (c). If the critical conditions of hydride size, stress and strain are reached, the hydride at the flaw tip cracks and the crack extends by the length of the hydride (d). The process of hydride nucleation, growth and fracture may repeat and the crack extend potentially becoming a through-wall defect.
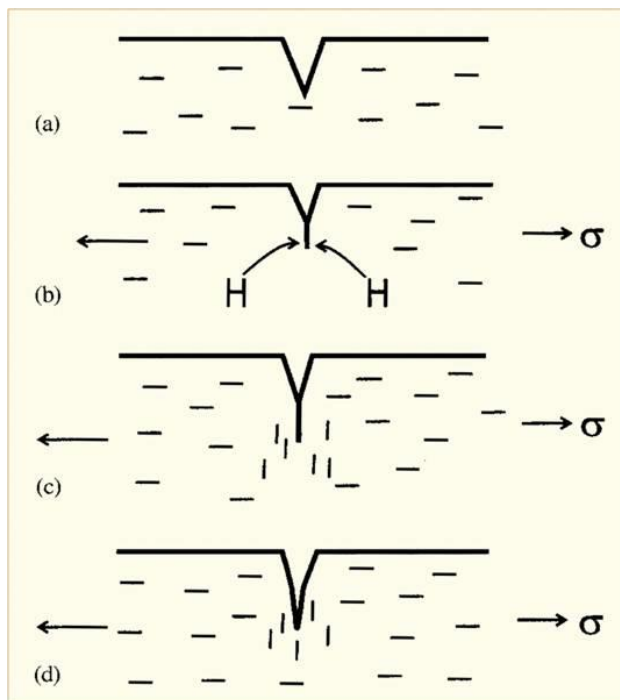
Figure 24: Schematic diagram of the traditional mechanism of DHC. A component or specimen of zirconium alloy contains circumferential hydrides and a surface flaw, (a). When a tensile stress is applied, hydrogen in solution diffuses up the stress gradient at the flaw tip and forms a hydride, which is reoriented in this example, (b). With time this hydride grows and more reoriented hydrides are formed around the flaw tip, (c). Once a critical condition is reached the hydride closest to the flaw tip cracks, (d). The process is then repeated [Coleman, 2003].

To consider DHC at pool conditions, the source of cladding stress must be identified. Upon cooling from reactor operating conditions, the fuel pellets will shrink away from the cladding inner surface and will be only loose contact. Therefore, the main stress source for the cladding and associated hydrides will be the internal pressure from the gasses (helium from fabrication and fission gases).

Flaws may exist in the cladding inner surface from pellet-clad mechanical interaction during operation in reactor core. Flaws may also exist on the cladding outer surface due to grid-to-rod fretting wear that occurred during reactor operation.

The large reduction in fuel rod internal pressure due to the Clab pool temperature greatly reduces the risk of DHC. For DHC to start, the flaw must be very large. Analyses performed by [Machiels, 2011] suggest that flaws that could propagate by DHC would be detected before the fuel was placed in storage or that DHC will be absent during pool storage of intact fuel.

Helium embrittlement of zirconium alloys

Most engineering alloys containing Fe, Cr and Ni such as zirconium alloys used in nuclear fuel cladding have (n,α) (neutron interaction with a particular element or isotope that produces an alpha particle or energetic helium nucleus) cross-sections that are large at high neutron energies (E > 1 MeV) [Griffith, 2023]. Ni, in particular, has a high (n,α) and appreciable He can be produced, especially in fast reactors. Ni has another property for a reactor that has a high thermal neutron flux. The main isotope ^{58}Ni reacts with thermal (low energy neutrons) to produce ^{59}Ni. This non-natural isotope has a very high (n,α) cross-section across a broad range of neutron energies. That is one reason why Ni-alloys produce so much He in a thermal neutron environment (~30,000 appm over the life of an Inconel X-750 component in a CANDU reactor). The amount of He produced is a product of the abundance and cross-section for each isotope. See Figure 23 which illustrates the (n,α) cross-sections for the main isotopes of Fe, Cr and Ni as well as for ^{59}Ni. The effect of ^{59}Ni only becomes important once a substantial amount has been produced by transmutation (peaks at approximately 2-3 at% in a PWR after 20-30 years of operation). Considering the main isotope of Zr, ^{90}Zr, one can see in the plot below, Figure 25 that the high energy (n,α) cross-section is shifted – peaking at about 10 MeV. There is very little overlap of the Zr (n,α) cross-section with the neutron spectra for most reactors and which is one reason why there is little or no swelling in Zr and why there is no He-embrittlement. There may exist a few cavities in Zircaloys from the precipitates.
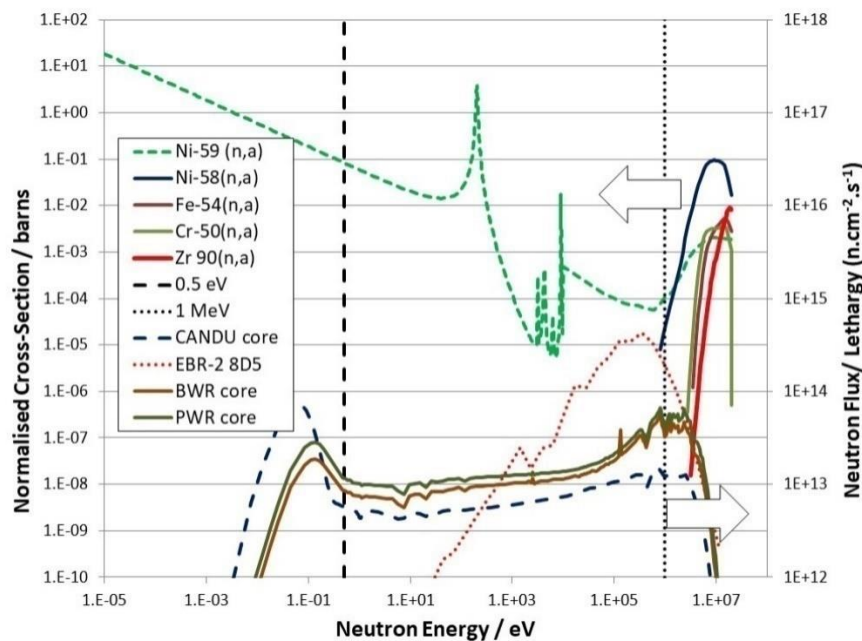
Figure 25: Normalised cross-section as a function of neutron energy for different reactor types [Griffith, 2023]

Clab Storage

In contrast to power reactor operating conditions, the comparatively low temperature and relatively unaggressive pool water chemistry postulated for fuel and core component storage at Clab rule out further significant hydrogen generation, diffusion and redistribution. Further, the relatively low storage temperatures at Clab minimize the likelihood of hydride-associated failures such as delayed hydride cracking and progressive through-wall cracking at localized concentrations of zirconium hydrides (e.g. hydride nodules or "sunbursts").

The storage temperatures at Clab and the relatively low concentration of helium in the zircaloy material make it unlikely to be a credible mechanism for zirconium alloy failure.

Precautions

To prevent structural failure, minimize the handling of very high burnup fuel (> 60 GWd/MTU assembly average) or fuel cladding types with documented hydride nodules. Vigorous rotation or twisting of fuel and associated core components such as might be done to confirm handling tool latching should be avoided.

No precaution is suggested regarding helium embrittlement of zircaloy in long term storage at Clab.

Stainless Steel and Nickel-Base Alloys

Hydrogen induced failures of in-reactor stainless steel or nickel-base alloy components have not been identified as a significant issue. Hydrogen is generated in stainless steels by neutron irradiation from (n,p) reactions with a threshold energy of ~1 MeV. The dominant source is transmutation reactions of nickel and, as can be seen in Figure 26, several tens of appm of hydrogen can be expected to accumulate from this source. The solubility of hydrogen in stainless steels is quite high but the diffusion rate is low (relative to ferritic steels) [Ford F P et al. 2010]. However, bubbles or cavities created by helium act as deep traps for molecular hydrogen. Hydrogen may also enter stainless steels due to general corrosion at typical LWR operating temperatures and particularly in PWRs due to redox reactions with dissolved hydrogen in PWR primary water. In fact, hydrogen from the PWR primary water can substantially dominate over that resulting from (n,p) reactions with typically 400-600 appm being observed, exceptionally extending up to a few thousands appm [Garner F et al, 2002], [Garner F et al, 2006] and [Edwards D et al, 2009].
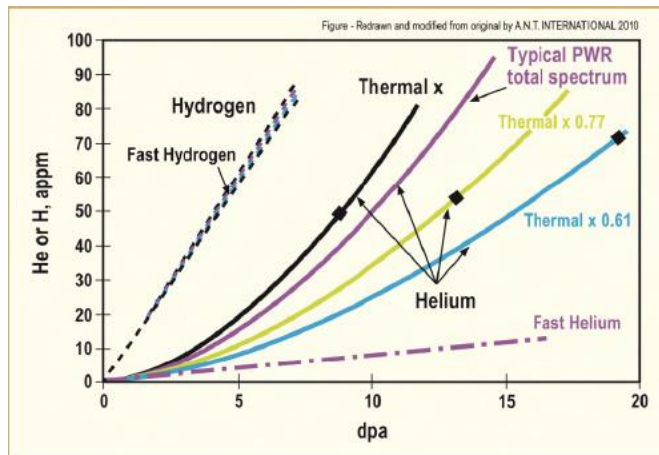
Figure 26: Helium and hydrogen production from transmutation reactions in stainless steel as a function of neutron fluence and energy [Garner F et al., 2002].

Clab Storage

Austenitic steels in the unirradiated condition are very resistant to hydrogen embrittlement at the operating temperatures of PWRs and BWRs. In unirradiated austenitic stainless steels, hydrogen concentrations orders of magnitude greater than those identified above would be necessary to cause any significant embrittlement (due to transgranular cleavage, intergranular or interphase cracking), e.g. [Ulmer and Altstetter, 1990] and [Buckley and Hardie, 1993]. Stainless steels that are hardened by martensite, or precipitation hardening, or cold work, do show evidence of susceptibility to intergranular cracking when the hardness exceeds ~350 HV. On this basis, it can be anticipated that austenitic stainless steels after irradiation doses sufficient to saturate the mechanical properties (>5 to 10 dpa) would be well over this hardness limit and consequently susceptible to hydrogen embrittlement, albeit not necessarily at normal reactor operating temperatures. Studies of hydrogen embrittlement of irradiated austenitic stainless steels are rather limited. At the maximum neutron fluences and hydrogen concentrations expected in the austenitic stainless steels of PWR internals, hydrogen embrittlement in combination with irradiation induced hardening cannot be dismissed as a potential cause of intergranular cracking [Ford et al. 2010]. Susceptibility should be limited at normal operating temperatures but could be more important during transient conditions at lower temperatures. However, studies [Pathania R et al, 2007] have failed to show evidence of hydrogen induced cracking in tests at either 100 or 320 °C.

Precautions

No precaution is suggested regarding stainless steel hydriding of fuel or core components in long term storage at Clab.
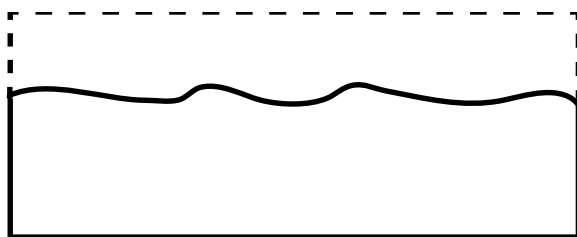
4.3.2 General oxidation

General oxidation (or corrosion) occurs uniformly across the surface of a material resulting in the gradual and overall deterioration of its properties. General oxidation may be caused by chemical or electrochemical reaction between the metal's exposed surface and the surrounding environment.

Zirconium Alloys

General or uniform oxidation is a form of corrosion that affects the whole of the exposed surface. It is characterized by a reduction of zirconium alloy metal thickness exposed to the corrosive environment. Figure 27 illustrates this type of corrosion.

Measurements of Zircaloy-2 process tubing exposed to treated Columbia River water at approximately 90°C for up to four years were conducted. Oxidation rates were estimated for "out-of-flux" or unirradiated material to be less than 0.01 microns/year. These measurements and estimations were consistent with contemporary laboratory work. Extrapolation of general corrosion data of zircaloy to 100 years was determined to be 0.3 to 0.5 microns of oxide. [Johnson, 1977]

Immersion tests of Zircaloy-4 tubing in 50°C for up to one year in a variety of potential storage chemistries were conducted. Corrosion rates were found to be below the limit of detection (50 nm/year). No spallation of the original oxide layer was noted and no change to oxide morphology was observed. Corrosion rates in pool water at 30°C to 45°C are a factor of $\sim10^{10}$ lower than those under operating reactor conditions. Therefore, zirconium-based alloys are virtually immune to uniform corrosion during wet storage, at least under controlled chemistry water conditions [IAEA-TECDOC-1862, 2019]

Figure 27: General or Uniform Oxidation (courtesy of G. Was) [Jones, 1996]

Clab Storage

The pool chemistry and temperature at Clab are bounded by the experiments and operational measurements noted above. General corrosion is not expected to be a significant material degradation concern for the anticipated storage time.

Precautions

Routine water chemistry monitoring to confirm expected pool environmental conditions. Also, ensure an action plan exists to address out of specification conditions associated with Clad storage water chemistry. A return to nominal water chemistry should occur within days.

Stainless Steel and Nickel-Base Alloys

These alloys exhibit excellent general corrosion resistance in most aqueous environments. In concentrated caustic solutions at high temperature, they can undergo wastage or degradation by uniform corrosion resulting in a decrease of thickness or mass. But this is unusual and is not a condition that will occur in a storage pool as long as the pH is not over 10 [Ford et al. 2010].

While stainless steel (e.g. 304L) and Inconel may have uniform corrosion rates higher than zirconium-based materials in storage pool water, they have excellent corrosion resistance (< 1 micron per year loss by corrosion) [Olander and Motta, 2019]

Clab Storage

Assuming Clab water chemistry remains in the assumed range, general oxidation damage to fuel and core components due to long-term storage is highly unlikely.

Precautions

Routine water chemistry monitoring to confirm expected pool environmental conditions. Also, ensure an action plan exists to address out of specification conditions associated with Clad storage water chemistry. A return to nominal water chemistry should occur within days.

4.3.3 Localized Corrosion (e.g. pitting, intergranular attack, microbe induced corrosion MIC)

Pitting corrosion is a form of extremely localized corrosion that leads to the creation of small holes or pits in the base metal [Jones DA, 1996]. Pitting can occur when, for example, a small area loses its protective corrosion layer, or when contaminants or impurities become lodged in the surface of the material. This small area acts as an anode where the oxidation reaction occurs, while the larger portion of the material acts a cathode where the reduction reaction takes place, similar to a galvanic cell between two dissimilar metals. Because of the large difference in the areas of the anode and cathode, corrosion at the anode occurs rapidly and develops into a pit. Figure 28 illustrates localized or pitting corrosion.

Intergranular attack is a form of localized corrosion that occurs at grain boundaries. Intergranular attack can occur in specific instances in which the composition of the grain boundary is different from that of the matrix, setting up a type of micro-galvanic cell. That composition difference may arise from segregation of elements to or away from the boundary or depletion of passive film forming elements by precipitation along the grain boundary. Intergranular corrosion is illustrated in Figure 28.

MIC is a form of localized corrosion caused by action or presence of microorganisms in biological films on a particular surface. The microorganisms may be bacteria, fungi or algae. These microorganisms produce metabolites that create a localized environment that promotes accelerated corrosion. Biological films have previously been found at the edges of spent fuel storage pools at Clab.
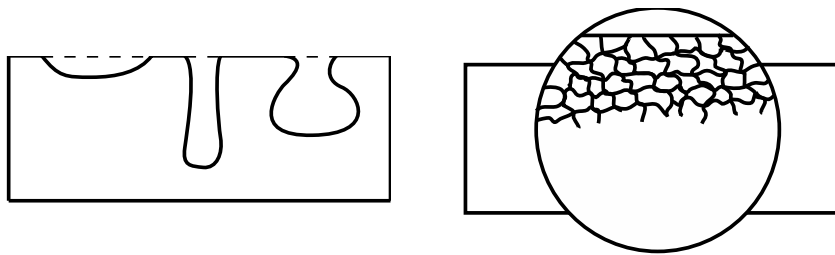
Figure 28: Pitting corrosion (left) and intergranular corrosion (right) (courtesy of G. Was) [Jones, 1996].

Zirconium Alloys

Enhanced corrosion of zirconium alloys may occur when the corroding surface is close to, or in contact with, certain other metallic components. The shape of the component is often reproduced in the shape of an area of enhanced corrosion, suggestive of a shadow cast by the component on the zirconium alloy surface. The term "shadow corrosion" is therefore often used to describe the phenomena.

Shadow corrosion may be described as irradiation-assisted galvanic corrosion. The appearance and post-operational measurements of shadow corrosion with respect to fuel cladding integrity is described below.

Inconel spacers cast a "shadow" on fuel rod cladding (Figure 29) that accelerate localized corrosion.

Figure 29: Appearance of Inconel spacer induced shadow corrosion on Zry-2 cladding. The photo to the right is the same assembly as shown to the left but after the spacer has been removed and the rods have been brushed to remove the loosely adherent CRUD [Châtelain et al., 2000] and [Ledergerber et al., 2006].

The oxide thickness at these positions is increased relative to the rest of the fuel rod cladding (Figure 30 and Figure 31). The red spikes in the oxide measurement occur that the spacer grid locations.
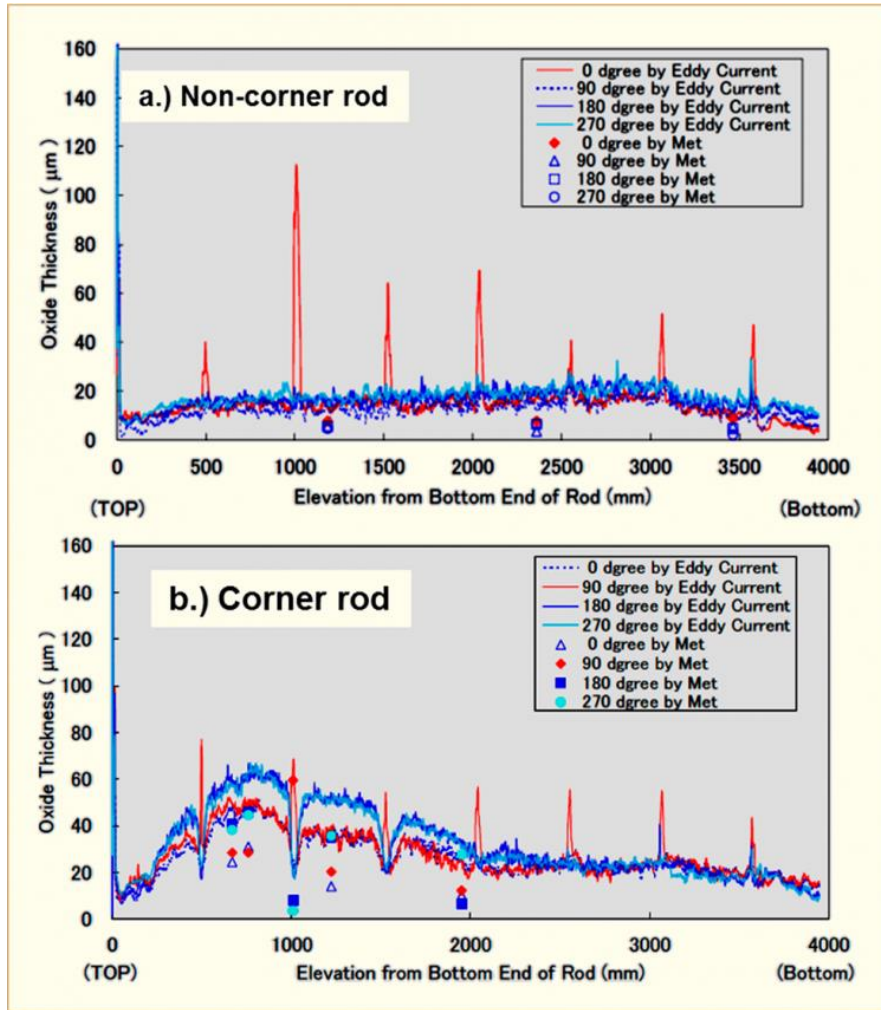
Figure 30: EC Oxide film profile line scans from different peripheral positions from 2 fuel rods of a 9×9 BWR fuel assembly with bi-metallic ferrule springs exposed in Fukushima to a burnup of 55 MWd/kgU [Miyashita T. et al., 2007]
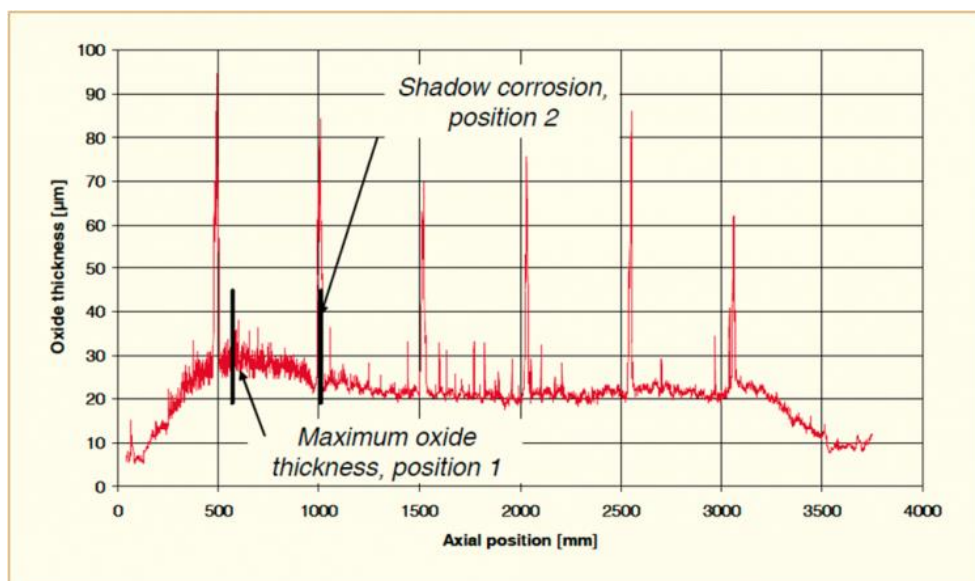

Figure 31: Oxide profile of ATRIUM 10P fuel rod with all-Inconel spacers [Hoffmann et al., 2005]

Another shadow corrosion example involves a bi-metallic ferrule spacer where the oxide thickness peaks opposite the Inconel springs (Figure 32) and opposite the Zry-4 contact points. In another series of measurements, oxide peaks were noted only at the Inconel spring contact point (Figure) [Miyashita T. et al., 2007].
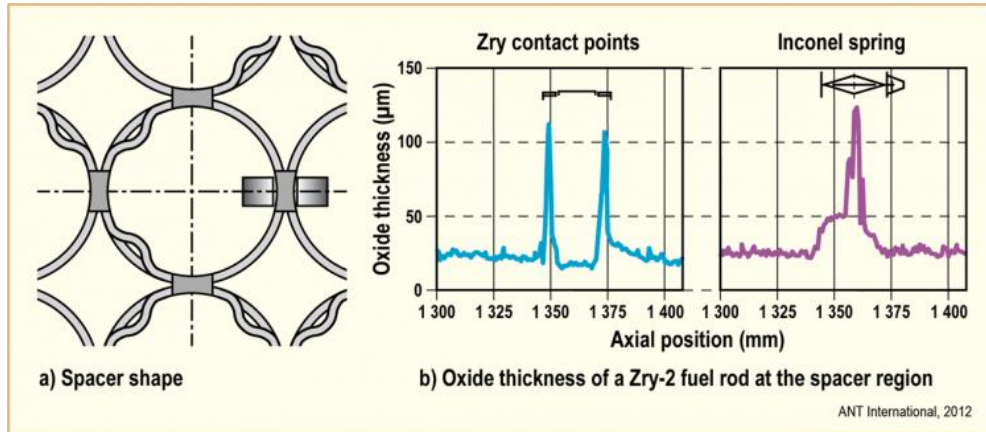
Figure 32: Oxide layer thickness profile of a BWR Zry-2 fuel rod opposite to Zry contact points and Inconel contact point after a burnup of 46 MWd/kgU [Garzarolli et al., 2001]
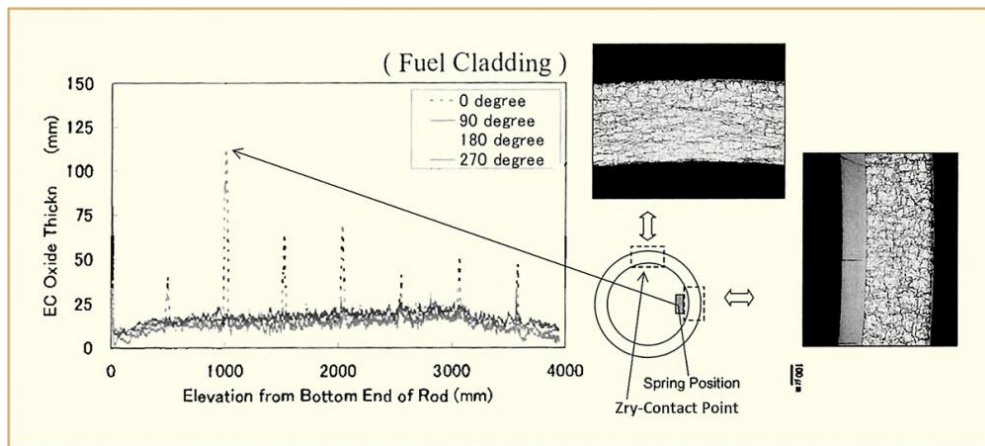
Figure 33: Oxide profile and metallography of the 2nd ferrule spacer after a burnup of 45 MWd/kgU [Miyashita T. et al.,2006]

With Inconel spacers, the oxide thickness is increased in the entire area below the spacer strip (Figure 34). The fuel cladding maximum oxide thickness is observed at positions where Inconel is in contact or close to the zirconium alloy cladding, whereas at positions at the spacer corners, where the cladding-spacer distance is larger, the oxide is only slightly increased.
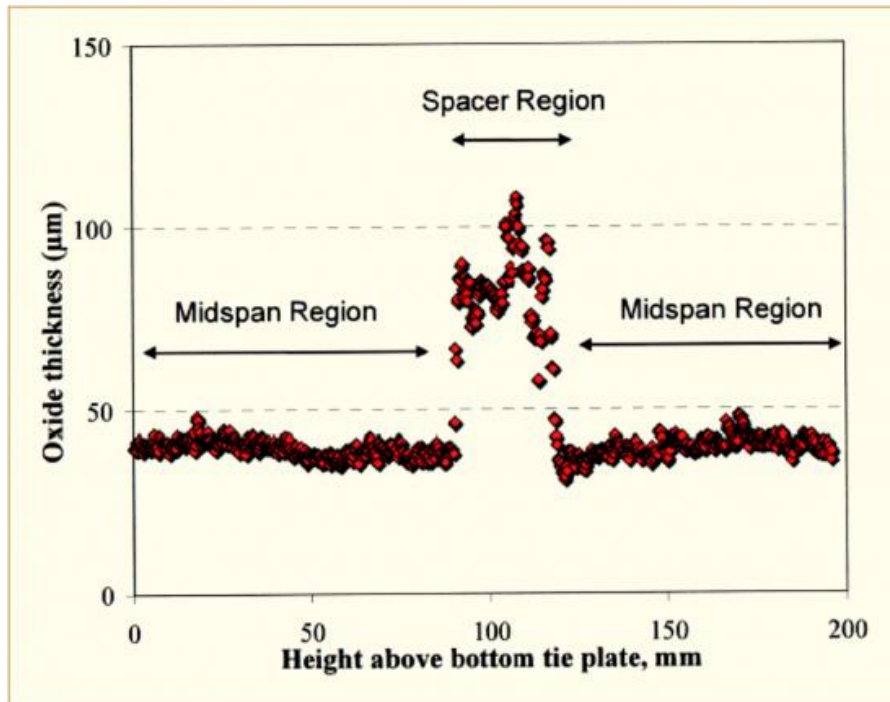
Figure 34: Oxide thickness profile on a fuel rod overall Inconel spacers [Kelen et al., 2000]

An oxidising environment and irradiation is required to cause shadow corrosion. Thus, shadow corrosion is usually not observed in reducing environments such as in PWRs and VVERs. **Furthermore, Zr alloy shadow corrosion cannot be initiated without irradiation which means that shadow corrosion is not possible in Clab.**

Also, Zirconium alloys are not susceptible to pitting corrosion under the assumed water temperature and chemistry conditions of the Clab storage pool. This is confirmed as a result of decades of commercial light water reactor spent fuel pool chemistry monitoring, incidental observations during fuel handling and inspections of fuel. MIC may occur as a result of metabolic activity of primarily sulphate reducing bacteria in stagnant water. However, MIC has not been observed as a pitting mechanism associated with spent fuel or core components in light water reactor storage pools due to storage conditions.

Stainless Steel and Nickel-Base Alloys

Localized corrosion such as pitting may occur when two conditions are met simultaneously: i) An oxidizing species other than water is dissolved in the bulk environment and is depleted within the occluded cell, and ii) a strong acid forming anionic species is present in the bulk solution that can allow a low pH to develop in the occluded cell, e.g. chloride or sulphate ions. Stainless steels are susceptible to pitting in Cl-containing environments. As such, it is important that Cl concentrations be minimized so as to avoid the breakdown of the passive film and the development of a pit. Figure 35 illustrates pitting corrosion in stainless steel.

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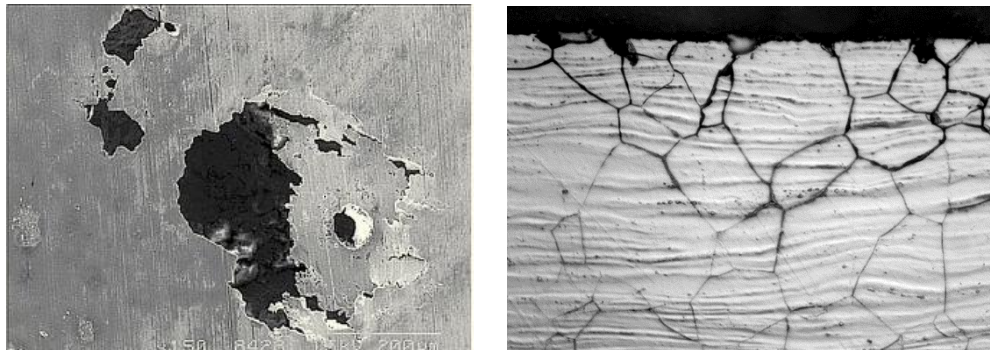
Figure 35: Pitting corrosion (left) [Laitinen T M J, 1999], and intergranular corrosion (right) in stainless steel (courtesy of G. Was).

Stainless steels can also undergo intergranular corrosion in certain instances. These include sensitization and radiation induced segregation. Sensitization occurs when stainless steels are heated into the temperature range of 600-700°C for several hours. Such heat treatments result in precipitation of chromium carbides preferentially at the grain boundaries, depleting the grain boundary region of the passive film former, chromium. Exposure to acidic solutions results in attack of the grain boundary region. Radiation induced segregation can produce a similar result as Ni and Si segregate to the grain boundary and Cr segregates away from the grain boundary. These segregation zones are extremely small (~10 nm) compared to the ~1 micron width in a sensitized condition.

Stainless steels are also susceptible to MIC but not nickel-base alloys. MIC is caused by the development microorganisms on metal surfaces. Micro-organisms can create aggressive chemical conditions within or under the bio-films they build up on metal surfaces. Several different types of microorganisms can be implicated in such processes, including bacteria, algae, yeasts and fungi. They preferentially develop at temperatures of ~30-45 °C and their development is favored by the availability of energy, carbon and phosphorus sources, and electron donors or acceptors. Formation of species that are deleterious for passive oxide layers on stainless steels include reduction of sulphate into sulphide in anaerobic conditions and ennoblement of corrosion potential by formation of very oxidizing species, for example: Oxidation of Fe^{2+} into Fe^{3+}, and oxidation of Mn^{2+} into MnO_2 by Manganese Oxidizing MicroOrganisms (MOMOs). Many examples of pitting in water circuits with chloride concentrations normally too low to cause localized corrosion are now attributed to MOMOS. This is because the manganese oxide, MnO_2, is electrochemically very active and provides cathodic current that raises the local corrosion potential of stainless steel above the pitting potential, particularly in heat tinted zones where resistance to localized corrosion is lower than on non-oxidized surfaces. Water containing more than 10 μg/l of Mn ions can produce Mn fouling if MOMOs are present and those containing more than 50 μg/l of Mn ions have a high potential for deposition

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and MIC. Figure 36 is an example of a pit caused by MIC.

Figure 36: Microbial corrosion pit [Linhart P, 2006].

The authors consider this mechanism as unlikely to be a significant contributor to fuel or core component degradation in that the radiation from decay of fission and activation products would render the spent fuel and core components sterile and thus immune to direct microbial corrosion attack. Further, the assumptions regarding water quality further reduce the likelihood of MIC affecting fuel or core components directly. MIC can affect the pool liner, cooling and circulation piping, valves and other pool cooling and clean-up equipment. This could lead to release of corrosion products downstream of filters and demineralizers allowing those corrosion products to come into direct contact with the stored fuel and core components. Build-up of these corrosion products may lead to galvanic or CRUD-assisted corrosion. Should the pool cooling and clean-up system sample water chemistry at the system inlet (as the water leaves the pool), microbial corrosion products of the pool cooling system downstream of the chemistry sample point may be undetected. This may be of particular concern in isolated system piping or components.

Clab Storage
Assuming Clab water chemistry remains in the assumed range in which the anion concentration is kept very low, local oxidation damage to fuel and core components due to long-term storage is highly unlikely.

Precautions
Routine water chemistry monitoring to confirm expected pool environmental conditions. Maintain the corrosion potential below the repassivation potential for stainless steels. Also, ensure an action plan exists to address out of specification conditions associated with Clad storage water chemistry (e.g. loss of cooling flow and sulfate contamination of pool water). A return to nominal water chemistry should occur within days.

Stress Corrosion Cracking

Stress corrosion cracking, SCC, is the formation and growth of cracks in a material under tensile stress in a corrosive environment. Figure 37 illustrates this type of corrosion-related damage.
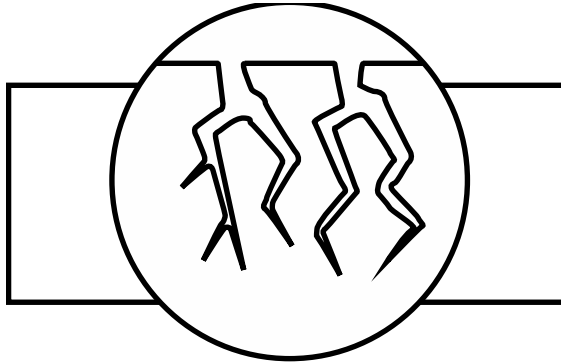
Figure 37: Stress corrosion cracking (courtesy of G. Was) [Jones, 1996].

Zirconium Alloys

Pellet Cladding Interaction, PCI, is a chemically-assisted stress corrosion cracking mechanism [Adamson et al., 2006] and [Jasiulevicius, 2023]. Fission products such as iodine have an affinity to cause cracking of the zirconium alloy in the presence of tensile stress, Figure 38. During reactor operation, local power increases cause fuel pellet cracking and expansion. Pellet cracks facilitate the diffusion of iodine to the clad inner surface. The pellet expansion and cracking create stress in the cladding.
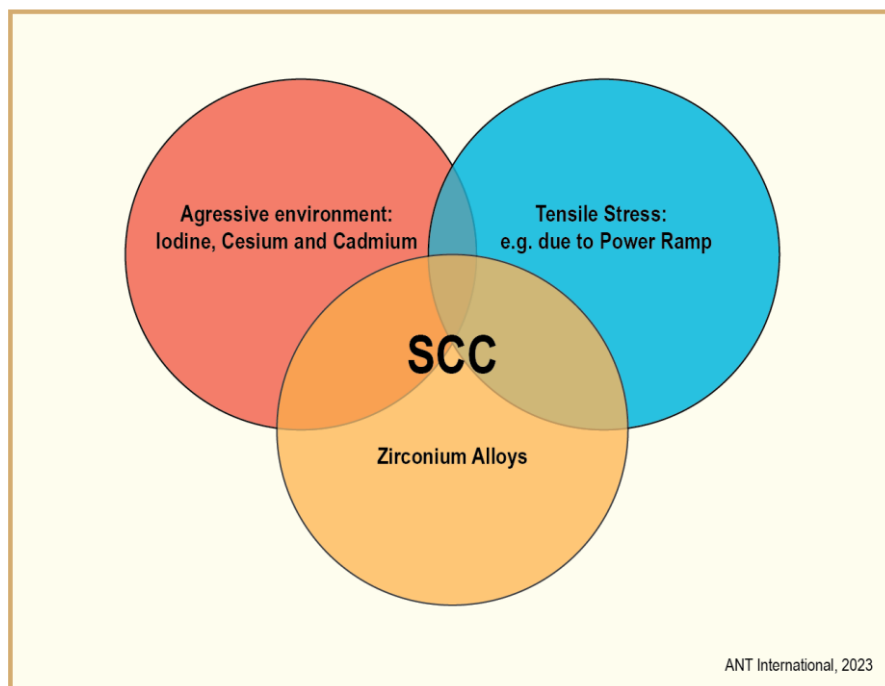
Figure 38: Schematic showing the three components involved in PCI

The production of fission products such as iodine and the migration of these aggressive, volatile species to the areas of stress are required to initiate

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cladding inner surface attack. Figure 39 illustrates a PCI crack propagating in fuel cladding.
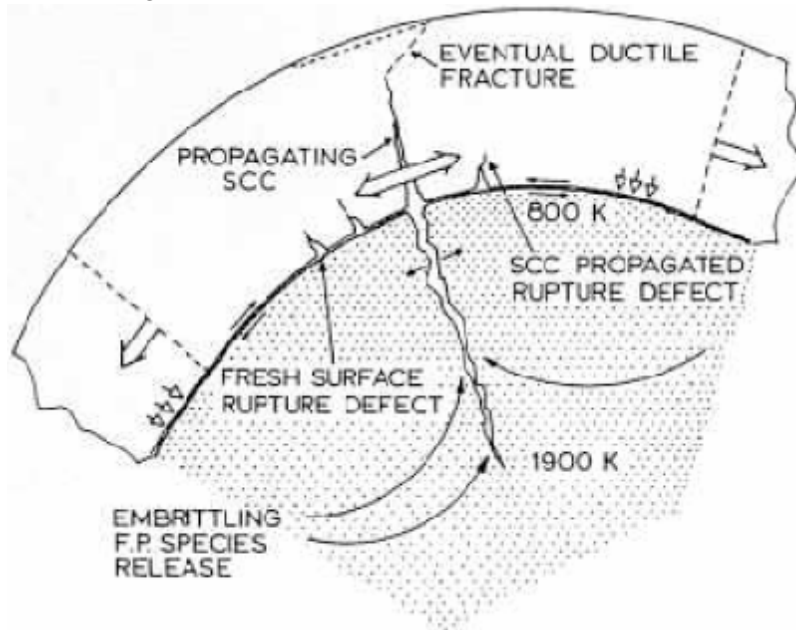
Figure 39: PCI [OECD, 2004].

Studies by [Grubb & Morgan, 1979] showed that aggressive fission-product species were either iodine or a composite Cs/Cd mixture. Of these, only iodine has the ability to form a series of Zr iodides with very different chemical and physical properties, with the end members of the series being immobile and non-volatile at reactor operating temperature [Cubicciotti et al., 1982]. This provides an explanation of the limited time within which the fission-product environment is capable of generating PCI cracks since the Zr/I reaction products will progress through the series:

$$Zr + 2I_2 \rightarrow ZrI_4 \xrightarrow{\pm Zr} ZrI_3 \xrightarrow{\pm Zr} ZrI_2 \xrightarrow{\pm Zr} ZrI$$
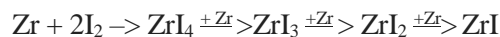
No SCC ability was detected for ZrI_2 and ZrI [Peehs et al., 1979]. **These results indicate that SCC of Zr alloys in spent nuclear fuel is not possible since the SCC harmful ZrI_4 and ZrI_3 would have decomposed into ZrI long before the fuel is transported to Clab.**

The harmful Zr iodides will have decayed into harmless Zr iodides long before the spent fuel is transported to Clab, stress corrosion failure of zirconium alloys due to long-term storage is not possible.

No action is suggested regarding fission product attack of zirconium alloys in long term storage at Clab.

Stainless Steel and Nickel-Base Alloys

Irradiated stainless steels and nickel-base alloys are susceptible to irradiation assisted stress corrosion cracking (IASCC). Environmental conditions conducive to IASCC include an oxidizing environment, high conductivity water, and high corrosion potential environment. Figure 40 shows an intergranular crack in cold-worked 316 stainless steel caused by IASCC.
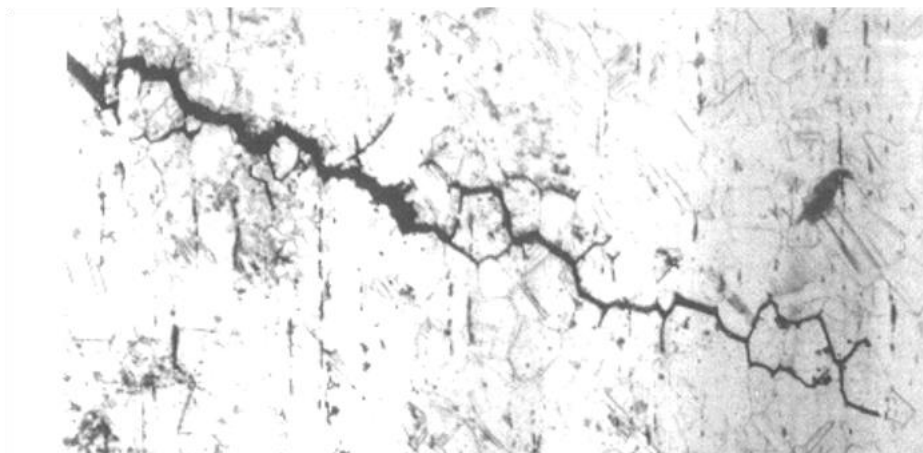
Figure 40: IASCC in 316 stainless steel (courtesy of G. Was).

The same is true for intergranular SCC (IGSCC) of sensitized stainless steels. Extensive laboratory experience has shown that irradiation and stress need not occur simultaneously for IASCC to occur. That is, stress corrosion cracking tests in the laboratory on pre-irradiated samples can readily lead to cracking initiation and growth. In fact, cracking can occur under constant load at stresses less than 50% of the irradiated yield stress. Thus, minimizing stress on irradiated stainless steel in environments conducive to SCC is important. Figure 41 illustrates IGSCC developing below a pit.

The most documented instance of Inter-Granular Stress Corrosion Cracking (IGSCC) that would affect fuel storage and handling of spent PWR fuel is that of 304 stainless steel bulge joint failure in Westinghouse fuel just below the top nozzle [USNRC, 2002]. While the phenomenon has been noted, there is no indication that IGSCC will progress in pool storage with water chemistry conditions in compliance with industry guidelines. Westinghouse has identified the affected fuel population and mitigation measures to allow safe handling of associated fuel.
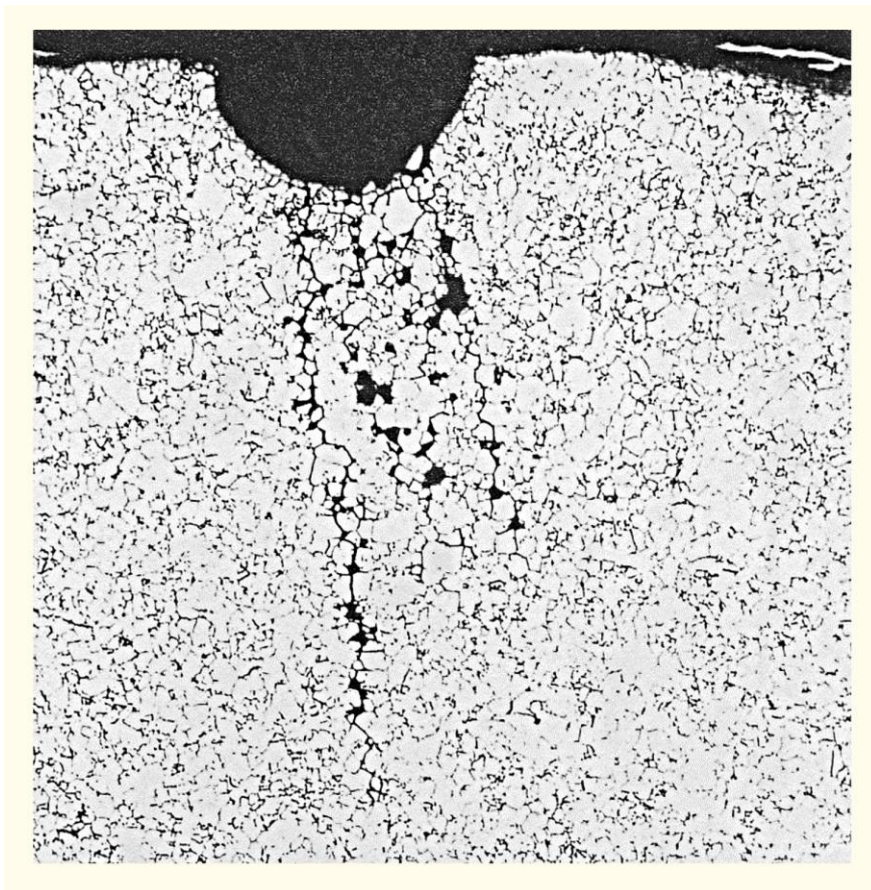
Figure 41: IGSCC developing below a corrosion pit (courtesy of G. Was).

Assuming Clab water chemistry and water temperatures remain in the assumed range, stress corrosion failure of stainless steel alloys due to long-term storage is highly unlikely.

Other than Westinghouse recommendations regarding 304 stainless steel bulge joint failure, this report makes no suggestions associated with stress corrosion cracking monitoring or mitigation.

Galvanic Corrosion

Galvanic corrosion (also known as bimetallic corrosion or dissimilar metal corrosion) is a process in which one metal preferentially corrodes when it is in electrical contact with another metal in the presence of an electrolyte. This electrolyte may be water with contaminants that provide a suitable level of conductivity. Figure 42 illustrates this type of corrosion.

Active   Noble
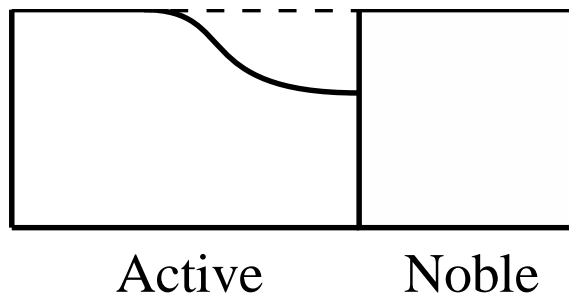
Figure 42: Galvanic corrosion (courtesy of G. Was) [Jones, 1996]

Zirconium Alloys

In spent fuel pools where the coolant acts an electrolyte, galvanic corrosion may occur when two dissimilar metals are in close contact. Stainless steel and zirconium have been observed to reside in pool storage with little to no mutual galvanic corrosion. Empirical observations confirm the supposition that little mutual galvanic corrosion would occur due to both metals being near the noble end of the galvanic series. Aluminum is an active metal and thus bare aluminum alloys adjacent to zirconium alloys in a relatively high conductivity environment would result in aluminum oxidation and hydriding of the zirconium alloy. The rapid development of passivating oxides on the surfaces of aluminum and zirconium alloys mitigate galvanic corrosion. One particular type of galvanic corrosion that may reside in Clab as a pre-existing condition primarily arising from BWR reactor operation is known as "shadow corrosion" (see previous discussion of localized corrosion). During reactor operation shadow corrosion may occur on the surface of zirconium alloys when in close proximity to non-zirconium alloys such as stainless steel or Inconel. The oxide thickness may become thick, dense and un-cracked. Areas of close contact of dissimilar metals in LWR fuel like fuel spacers or grids may be of particular concern due to the combined effect of shadow corrosion and grid-to-rod fretting during reactor operations. These fuel cladding areas may have a combination of reduced wall thickness due to grid-to-rod fretting and greater corrosion due to shadow corrosion. Thus, areas of fuel subject to possible shadow corrosion during reactor operation can become areas of concern regarding galvanic corrosion due to the close juxtaposition of dissimilar metals. Continued corrosion in areas affected by operational shadow corrosion due to dissimilar metals during Clab storage would require local coolant conditions to be outside of values noted in [IAEA, 2011].

Assuming Clab water chemistry parameters remains in the assumed range and adequate coolant flow occurs around fuel and core components in the pool storage canisters, further galvanic corrosion damage to fuel and core components due to long-term storage is highly unlikely.

Routine water chemistry monitoring to confirm expected pool environmental conditions. Confirm active pool water flow through the fuel and core components in the pool storage canisters. Also, ensure an action plan exists to address out of specification conditions associated with Clab storage water chemistry (e.g. loss of cooling flow or rise in pool water conductivity). A return to nominal water chemistry should occur within days.

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The cladding under spacer grids is suggested as part of a periodic visual examination program for pre-characterized fuel as noted in Surveillance Program 4 provided later in this report.

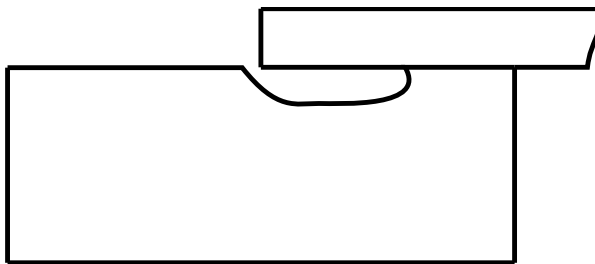### *Stainless Steel and Nickel-Base Alloys*

Galvanic corrosion can be a problem if stainless steel is coupled to a more noble metal, or by setting up an oxygen cell as in a crevice. Flowing water with low oxygen concentration helps to avoid such a situation.

Assuming Clab water chemistry parameters remains in the assumed range and adequate coolant flow occurs around fuel and core components in the pool storage canisters, galvanic corrosion damage to fuel and core components due to long-term storage is highly unlikely.

Routine water chemistry monitoring to confirm expected pool environmental conditions. Confirm active pool water flow through the fuel and core components in the pool storage canisters. Also, ensure an action plan exists to address out of specification conditions associated with Clad storage water chemistry (e.g. loss of cooling flow or rise in pool water conductivity). A return to nominal water chemistry should occur within days.

Crevice Corrosion of stainless steel and Nickel-Base Alloys

Crevice corrosion occurs in relatively closed or tight spaces between materials. The fluid in these crevices is relatively stagnant and not replaced or flushed by the bulk fluid surrounding the material. The crevice may be a gap or contact area between parts of the fuel assembly or core component. Figure 43 illustrates crevice corrosion.

Figure 43: Crevice corrosion (courtesy of G. Was) [Jones, 1996]

Stainless steels and nickel-base alloys are susceptible to crevice corrosion in Cl-containing environments by much the same process as pitting. The concentration of Cl needs to be minimized to avoid the breakdown of the passive film and the development of crevice corrosion. Stainless steels generally perform well in flowing water with low levels of Cl, SO_4 and O_2.

Figure 44 illustrates crevice corrosion in 316 stainless steel under a gasket seal in brackish water.

Figure 44: Stainless steel crevice corrosion [During, 1991]

Clab Storage

Assuming Clab water chemistry parameters remains in the assumed range and adequate coolant flow occurs around fuel and core components in the pool storage canisters, crevice corrosion damage to fuel and core components due to long-term storage is highly unlikely.

Precautions

Routine water chemistry monitoring to confirm expected pool environmental conditions. Confirm active pool water flow through the fuel and core components in the pool storage canisters. Also, ensure an action plan exists to address out of specification conditions associated with Clad storage water chemistry (e.g. loss of cooling flow or rise in pool water conductivity). A return to nominal water chemistry should occur within days.

4.3.4 CRUD-Assisted Corrosion of zirconium alloys

CRUD is a term used for corrosion products from regions of the reactor primary coolant system that deposit on the relatively hot surface of the fuel cladding. The term "CRUD" is attributed to "Chalk River Unidentified Deposit" where this phenomenon was first documented. In and of itself, CRUD does not challenge the integrity of zirconium alloys. The CRUD layer acts as an insulator which increases the outer surface temperature of the fuel cladding during reactor operation. CRUD can affect zirconium alloy corrosion by an insulating effect. This could lead to a number of corrosion rate impacting phenomena:

- Thermal insulation – increasing the zirconium alloy surface temperature leading to an increased corrosion rate. This has a greater impact potential for short decay time, high burnup fuel transported to Clab.
- Differential aeration cells – creation of electrochemical potentials between areas of relatively high and low oxygen within the CRUD layer(s).

CRUD may act to concentrate aggressive species within the CRUD layer

adjacent to the zirconium alloy surface. Figure 45 illustrates CRUD-assisted corrosion.
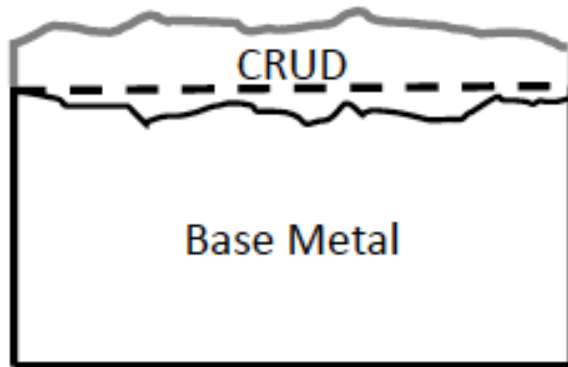
Figure 45: Assisted Corrosion (courtesy of M. Eyre) [Jones, 1996]

The range of CRUD layer thickness varies considerably in light water reactor fuel. Generally, CRUD deposits are notably thicker in BWR fuel and core components as compared to PWR. This observation is due to many BWR feedwater systems are composed of carbon steel piping (particularly in the US). Most CRUD is composed of iron and nickel oxides. Plants looking to reduce outage radiation personnel dose may inject zinc to prevent radioactive cobalt deposition on primary system piping and other components. In these plants, zinc may also be found in fuel and core component CRUD.

CRUD may lead to localized corrosion failures in LWR fuel during reactor operation. Figure 46 is an example of CRUD and localized failure in fuel cladding.

Figure 46: CRUD on BWR fuel (left) and localized corrosion failure of cladding (right) [Cheng and Yang, 2003]

The CRUD layers can have physical structures that vary from dense layers to porous ones. The porous layers can vary as well and be fluffy loose deposits or dense structures with pores and cracks or relatively loose deposits with convective coolant chimneys (wicks), all of which have a different thermal conductivity. The conductivity of these heterogeneous deposits depends partly on the CRUD composition and structure itself and partly on the composition

and thermal state of the coolant that penetrates its voids.

PWRs

The majority of the corrosion product input to the PWR coolant comes from the steam generators that represent 75 – 80% of the total surface area of the primary coolant circuit. Since the steam generator tubing is made of Ni-Fe-Cr alloys, it is not surprising that the corrosion product deposits are made of these elements. A notable difference exists between the Siemens Nuclear Steam Supply System (NSSS) that use steam generators with Incoloy 800 tubing (Fe base – 33Ni – 21 Cr) tubing and other NSSS that use Inconel 600 tubing (Ni base – 15 Cr –8Fe).

Ex-reactor (unirradiated) corrosion tests estimated a 2X – 4X higher corrosion rate of Inconel 600 compared to Incoloy 800. This, as well as some differences in water chemistry control methods account for the fact that essentially no CRUD deposits have been observed in Siemens NSSS even with nucleate boiling compared to moderate to high levels observed in plants with Inconel generators.

Excessive CRUD buildup is usually avoided even in Inconel 600 generator plants by judicious water chemistry management that maintains a positive temperature coefficient of solubility (i.e. increasing solubility with increasing temperature) for the corrosion products, the nickel ferrites, in the core. The corrosion products are maintained in solution by ensuring a pH >6.9 throughout the cycle by a balance of the B and Li in the reactor coolant.

The "normal" CRUD that results from this type of operation rarely causes excessive cladding temperatures and the corrosion that is observed is a combination of increased temperature and Li concentration that is increased significantly above its nominal level in the coolant within the CRUD and the oxide film.

The injection of Zn in reactor coolant for the reduction of radioactive material transport and worker dose reduction or steam generator tubing corrosion has resulted in the formation of zinc ferrites in a thin, dense layer on the fuel cladding. A small, but to date non-detrimental increase in corrosion has been noted due to the resulting temperature increase.

Heavy CRUD deposits that raise fuel cladding temperatures significantly can and have occurred by operation of reactor coolant pH <6.9, caused by the ingress of oxygen or loss of adequate hydrogen overpressure, or by the ingress of impurities. The Three Mile Island Unit 1 (TMI-1), Cycle 10 failures in 1995 are a good example where CRUD induced temperature increase played a major part [Tropasso, 2004]. Their first 24 month cycle started with 1820 ppm B and with Li limited to 2.2 ppm. The unit operated for 9 months at a pH range of 6.7-6.9, which caused a significant amount of CRUD to be deposited on the fuel. The failures were associated with the highest power 1st cycle fuel rods in the core and the failure locations were oriented azimuthally toward the water rods, which is typically a position of even higher local power and heat flux. The CRUD thickness was found to be at least 33 μm (not all could be

scraped off) and the oxide thickness was beyond the 100 μm limit of the measurement instrument. While Li concentration was probably a factor in the corrosion process, the clinching item indicating temperature as a factor was the hot cell examination of the cold worked-stress relieved cladding that showed recrystallized grain structure in the azimuthal positions facing the water rods near the failures. Blanketing by low thermal conductivity steam within the CRUD was proposed as an added factor that raised the temperature. Figure 47 illustrates a cross-section of failed cladding associated with the CILC failures at TMI-1 Cycle 10. The figure shows the cladding (light area) completely gone in the upper part of the cross-section photos exposing the fuel pellet to reactor coolant.
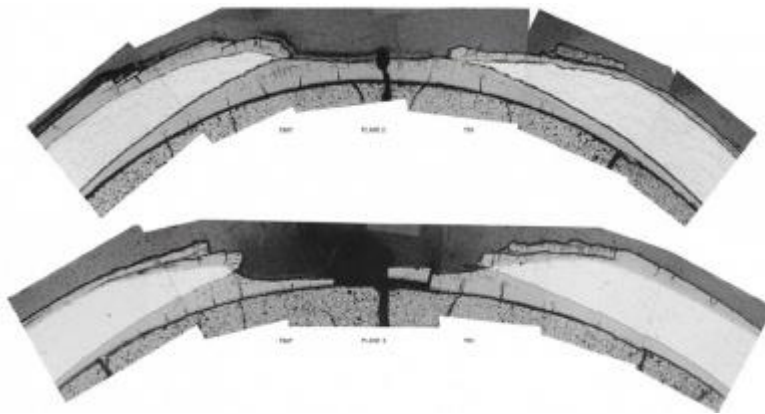
Figure 47: Cladding corrosion of TMI-1 Rod O11 at 3000 mm elevation [Garzarolli et al., 2001].

BWRs

The CRUD deposits on BWR fuels are generally loosely adherent or tightly adherent Fe_2O_3, or combinations thereof. The tightly adherent layers are generally thin, so that their effect on the clad temperature is minimal. The loosely adherent layers are very porous, filled with water, and as a result have a reasonably high thermal conductivity with only a small effect on cladding temperature.

Loose CRUD deposits can also have a morphology that may improve heat transfer. The liquid in the CRUD is drawn toward the CRUD – oxide interface through capillary passages in the loose CRUD slurry and then evaporates as steam when it reaches the interface, forming a chimney (or wick) radially through the CRUD to the coolant. The convection provides heat transfer from the interface. Thousands of such chimneys per mm^2 can provide better heat transfer than boiling on a non-crudded surface. This "wick boiling" mechanism has been observed on PWR rods as well. One should note that this is not the usual type of CRUD that is observed. A downside of this mechanism could be the concentration of some impurities at the bottom of the chimneys with their attendant corrosive effects.

Zn coolant injection for radioactive material transport reduction may result in the formation of a thin, dense layer of zinc ferrites on the cladding surfaces. The effect on cladding temperatures and corrosion has been small, but a limit on Fe content is considered appropriate when Zn injection is practiced to limit

the formation of the dense zinc ferrite type CRUD.

A particularly dense CRUD type that consists of Fe_2O_3 infiltrated by CuO or Cu caused many failures, CRUD Induced Localized Corrosion (CILC), in the 1970-1980s in plants with copper alloy condenser tubes. The thermal conductivity of the CRUD was measured as 2.2 W/m °C at 220°C. The low thermal conductivity of the CRUD by itself might not raise the cladding temperature sufficiently to cause through wall oxidation; however, steam blanketing in the circumferential cracks formed by the differential expansion of the clad and the high density CRUD can raise the temperature sufficiently to cause failure. Examples of such affected plants are, in addition to Vermont Yankee, Hatch-1, Hatch-2, Browns Ferry-1, Browns Ferry-2, all GE BWR/4 type plants in the US, and Ringhals-1 in Sweden [Strasser et al., 1985]. All the failed crud had adherent, heavy, flaking deposits that were found to consist mainly of CuO, instead of the fluffier hematite deposits normally found on BWR rods. The CILC failures were through wall "corrosion pits", as shown in the example in Figure 48.

Figure 48: Cross-section of corrosion pit in cladding caused by CILC. From [Marlowe et al., 1985]

The conclusions of [Marlowe et al., 1985] were that the CILC phenomenon requires three simultaneous conditions, namely environment (high copper concentration in the reactor water), duty (low initial heat flux, followed by higher heat flux), and material susceptibility (a cladding prone to nodular corrosion).

CILC is a failure mechanism only occurring during plant operation. Pool storage of sub-critical spent fuel will not support the corrosion noted by failed fuel inspections at operating nuclear power stations. However, one should be aware that CILC affected fuel assemblies that were not identified as failed may contain fuel rods with compromised areas of cladding.

Clab Storage
The presence of CRUD may increase the operating dosage and surface contamination potential throughout the process of transport container receipt to placement of fuel and core components in the Clab storage pools. The presence of thick (>100 microns) CRUD may increase the zirconium alloy surface temperature in spent fuel to some extent. This may lead to somewhat larger than expected long-term corrosion. One should be aware of CILC affected fuel not identified as failed in that sections of the unfailed cladding may be compromised in terms of local cladding wall thickness.

Precautions
Inspect and evaluate spent fuel cladding CRUD deposition. Should thick CRUD be observed, consider CRUD mitigation measures such as ultrasonic

cleaning prior to long-term storage in Clab. Careful handling of fuel assemblies that have been identified to have heavy CRUD deposition is appropriate.

5.   Surveillance Program

The A.N.T. INTERNATIONAL team reviewed the corrosion degradation mechanisms, assessments, and pool storage industry operating experience. Based upon those reviews, the A.N.T. INTERNATIONAL team developed a four-part surveillance program designed to monitor for adverse corrosion environments, prevent accumulation of potentially harmful materials, identify adverse corrosion mechanisms at work and identify the impact of these mechanisms on the spent fuel and core components stored at the Clab facility.

5.1 Surveillance program 1 – Storage Pool Chemistry Monitoring

Water chemistry monitoring is the cornerstone to the prevention of corrosion degradation. Monitoring assures the pool water chemistry provides an environment suitable for long-term storage of spent fuel and core components.

If pool water circulation conditions assure thorough mixing and homogenous conditions throughout the pool and storage canisters, water sampling at a single point may be performed (e.g. pool exit, circulating pump inlet, prior to filter/demineralizers). Circulation of at least two times the pool water volume per week is considered appropriate.

If homogeneity is not assured, the water could be sampled from multiple locations that can assure representation of chemistry conditions throughout the pool. This is particularly important as decay heat decreases over time and thermally driven circulation through the storage canisters is reduced.

Table 8 provides the minimum parameters, operating limits, and sample frequency for pool chemistry monitoring. This is consistent with IAEA good practices.

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Table 8: Parameters to be monitored, operating limits and sample frequency [IAEA, 2011].

Parameter	Operating Limit	Monitoring Frequency
pH	4.5 to 7	Weekly
Conductivity	< 10 uS/cm	Weekly
Solids	< 5 mg/L	Every six months
Cu	< 0.1 mg/L	Every six months
Cl	< 0.1 mg/L	Every six months
F	< 0.1 mg/L	Every six months
Pb	< 0.1 mg/L	Every six months
Nitrate (NO_3^-)	< 10 mg/L	Every six months
Sulphate (SO_4^{2-})	< 10 mg/L	Every six months
Fe	< 1.0 mg/L	Every six months
Al	< 1.0 mg/L	Every six months
Temperature	< 45°C	Monthly
Radioactivity*	See Note Below	Weekly
Turbidity**	See Note Below	See Note Below
*Water radioactivity and the presence of radioactive species should be measured whenever a water sample is drawn or once a week. A gamma scan can be used to identify the radioisotopes that may have come from failed fuel (e.g. Cs-137). The isotopic measurement should be evaluated on a case-by-case basis. Measurements of the radioactivity from filters and resins should be performed to detect the presence of leaking fuel. **Turbidity should be reduced, as necessary, to ensure visual clarity.		

5.2 Surveillance Program 2 – Pool Cleaning

The pool liner walls and floor should be visually examined periodically (i.e. every 6 months, not to exceed 1 year) for the presence of accumulated dirt, sludge, etc. These impurities may become suspended and deposit on fuel or

core components should flow, temperature or other pool chemistry conditions change. The accumulated dirt, sludge, etc. should be removed as well as the suspended solids that may result from the cleaning process.

5.3 Surveillance Program 3 – Corrosion Monitoring

Representative material coupons and galvanic couples should be part of the overall pool monitoring program.

Table 9 identifies corrosion coupon materials, configuration, inspection frequency and parameters to be measured. After inspection, the corrosion coupons should be replaced in the pool environment for inspection after the subsequent interval.

The corrosion coupons should reside in a representative area of the pool(s). The coupons should be accessible for retrieval. The coupon holder(s) should be clearly identified as part of the corrosion performance monitoring program to prevent inadvertent removal/disposal.

Table 9: Material Corrosion Monitoring

Material(s)	Configuration	Representation	Inspection Frequency	Inspection parameters
Stainless Steel	Bare coupon	Pool Liner, top & bottom nozzles, core components	5 years	Visual appearance, weight, dimensions
Boron-Stainless Steel	Bare coupon	Storage canister	5 years	Visual appearance, weight, dimensions
Boron-Stainless Steel	Bare coupon	Storage canister	10 years	Areal density
Zircaloy	Bare coupon	Fuel cladding, BWR channels	5 years	Visual appearance, weight, dimensions
Inconel (Nickel alloy)	Bare coupon	Fuel grids, springs	5 years	Visual appearance, weight, dimensions
Zircaloy/Inconel	Galvanic couple	Cladding, grid/springs	5 years	Visual appearance, weight, dimensions
Inconel/Boron-Stainless Steel	Galvanic couple	Grid/canister	5 years	Visual appearance, weight, dimensions
Zircaloy/Stainless Steel	Galvanic couple	Cladding/top and bottom nozzle	5 years	Visual appearance, weight, dimensions
Zircaloy/Boron-Stainless Steel	Galvanic couple	BWR channel/canister	5 years	Visual appearance, weight, dimensions

The choice of materials and galvanic couples were derived from descriptions of the fuel assemblies identified in [Svensk Kärnbränslehantering AB, 2010]. Table 10 identifies the materials subject to pool exposure during long-term storage at Clab.

Table 10: Materials in Typical BWR and PWR Assemblies [Svensk Kärnbränslehantering AB, 2010]

Material Composition	BWR (SVEA 96 Optima 2)	PWR (AREVA 17x17)
Fuel		
Uranium-total	175 kg	464 kg
Oxygen	23 kg	62 kg
Cladding		
Zirconium alloys	49 kg	108 kg
Stainless steel	-	3 kg
Fuel channel		
Zirconium alloys	32 kg	*
Stainless steel	8 kg	*
Other components (top & bottom fixtures, spacer grids, etc.)		
Stainless Steel	5 kg	12 kg
Zirconium alloys	-	21 kg
Nickel alloys	1 kg	2 kg
Fuel		
Uranium-total	175 kg	464 kg
Oxygen	23 kg	62 kg
*PWR fuel (AREVA 17x17) does not have a fuel channel		

5.4 Surveillance Program 4 – Corrosion Impact Inspection

In addition to the coupon program, representative PWR and BWR fuel should be visually inspected periodically to identify changes to corrosion color, apparent thickness, and overall morphology. Should changes be noted, evaluation of need for further examinations/tests should be made to determine if the observed change(s) are a sign of material degradation.

Most of the spent fuel to be stored at Clab include the design basis BWR and PWR assemblies identified in [Svensk Kärnbränslehantering AB, 2010]. The assembly designs are: SVEA 96 Optima 2 and AREVA 17x17.

When selecting representative fuel assemblies for periodic inspection, the following preferences should be considered:

- PWR assembly (AREVA 17x17)
- BWR assembly (SVEA 96 Optima 2)
- Higher burnup (e.g. > 45 GWd/MTU)
- Pre-characterization (Prior inspection records)

If the selected representative fuel assemblies have no pre-characterization/visual inspection records, the selected representatives should be inspected to create a baseline by which future examinations can be compared. Ideally, the selected and characterized PWR fuel assembly should have a control rod cluster inserted. In this way, a representative core component such as the control rod cluster may be inspected along with the PWR fuel assembly.

6. Summary and conclusions

The A.N.T. INTERNATIONAL team reviewed the spent fuel and core components to be stored at Clab and the assumed storage pool environment. The A.N.T. INTERNATIONAL team determined that corrosion is the most likely mechanism that could lead to structural degradation. The A.N.T. INTERNATIONAL team reviewed pool storage operating experience and concluded, as others have, that spent fuel and core components may be stored in a pool of high purity water for an extended period of time with no adverse consequences. The A.N.T. INTERNATIONAL team provided a surveillance program that could possibly be used to verify that the fuel and core components in storage at Clab perform as expected.

7. Outlooks and future activities

Based upon the reviewed public domain experiments, operating experience, material design and decades of direct experience regarding pool storage of spent fuel and core components, the A.N.T. INTERNATIONAL team is confident that interim storage of spent fuel and core components at Clab may be performed safely for up to fifty years. As with all things associated with commercial nuclear power, a thorough review of the operational scenario must be followed by confirmatory actions to ensure actual operation remains within the assumptions and limitations of the analysis/assessment.

8.   Acknowledgements

9.   References

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The Swedish Radiation Safety Authority (SSM) works proactively and preventively with nuclear safety, radiation protection, nuclear security, and nuclear non-proliferation to protect people and the environment from the harmful effects of radiation, now and in the future.