STR on Interim Dry Storage of Commercial Spent Nuclear Fuel

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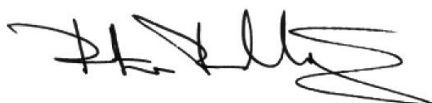Brentwood, California, USA

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Contents

1   Introduction

Dry storage of commercial spent nuclear fuel (CSNF) is a well-established technology, which, over the past 35 years, has emerged in several countries as the technology of choice for interim storage prior to final disposal of CSNF in geologic formations. This Special Topic Report addresses the degradation mechanisms that could potentially affect the performance of spent fuel stored in a dry, inert environment for periods up to ~100 years. The focus of the review is on the spent nuclear fuel rods, and not on the storage system components such as the casks or the canisters and their internal hardware elements. For the storage system components, the reader is referred to [NRC, 2019].

Topics related to long-term, dry storage of CSNF and to its subsequent handling and transport have been the subjects of many reports, papers and reviews, primarily in the context of their applicability for identifying, quantifying, or resolving regulatory requirements. In Section 2, today's regulatory requirements or guidance will be briefly summarized. In Section 3, postulated cladding degradation mechanisms will then be reviewed for their applicability to intact fuel rods, i.e., rods with no cladding through-wall defect(s); the assessment will focus on maintaining fuel rod integrity or on minimizing degradation of the cladding's mechanical properties; only the effects of residual water in the cask/canister after drying will be relevant to failed fuel rods. The results from an exercise to rank the significance and current state of knowledge of the different mechanisms will form the basis for the conclusion in Section 4.

2   Overview of Regulations and Regulatory Guidance

Spent fuel is generated continually by operating nuclear reactors. It is stored in the reactor fuel storage pool, or spent-fuel pool, for a period of time for cooling and then may be transferred to a designated wet or dry spent fuel storage facility, where it will await reprocessing or final disposal.

Storage in the spent-fuel pool was initially intended for very limited periods of time (as short as 90 days) prior to shipment of the spent fuel from the reactor sites to a reprocessing facility. However, because of delays in developing disposal facilities in most countries and limited market appeal for reprocessing, storage periods have been progressively extended, in many cases beyond the design lifetime of the reactor facility.

In this report, only dry storage of commercial spent LWR fuel is considered.

2.1   Basic Requirements [IAEA, 2012a]

Dry storage of spent LWR fuel has been safely implemented for over 35 years. IAEA Safety Standards Series No. SSG-15, Storage of Spent Nuclear Fuel, provides generic guidance for ensuring the safety of spent fuel storage. As stated in §1.3 of SSG-15:

> "The safety of a spent fuel storage facility, and the spent fuel stored within it, is ensured by: appropriate containment of the radionuclides involved, criticality safety, heat removal, radiation shielding and retrievability."

2.1.1   Containment

Containment prevents the release of radioactive material into the environment and is provided by the spent fuel cladding and the storage system (e.g., welded or bolted cask or welded canister). Therefore, maintaining spent fuel cladding integrity during dry storage is an important component of regulations.

For storage of spent fuel that has been characterized as failed or damaged, consideration of the condition of the fuel may lead to the inclusion of additional engineered methods for the safe handling of damaged fuel during loading and unloading, for examples, instrument tube tie rods for assemblies where stress corrosion cracking of the top nozzle is of concern, canning of damaged fuel assemblies to maintain spent configuration and ensure criticality control, and additional measures, such as encapsulation, to ensure the robustness of containment since, for damaged fuel, the primary containment feature, i.e., the spent fuel cladding, cannot be relied upon for control of the spent fuel material.

2.1.2   Criticality Safety

Criticality safety precludes an unplanned criticality event. Ensuring subcriticality often relies on geometry control. However, maintaining the geometry of spent fuel, especially during hypothetical accident conditions, may be challenging in terms of maintaining geometry. As a result, criticality control functions are often enhanced by other structures, systems and components (SSCs). These may include controlling fissile content, inclusion of neutron absorbers, and preclusion of moderators.

2.1.3   Decay heat removal

Effective removal of decay heat is important because degradation phenomena that could affect spent fuel integrity and some SSCs important to safety, such as polymers, are thermally activated.

2.1.4   Radiation Shielding

Shielding ensures that radiation exposure remains within safe limits and must be provided by the storage systems.

3 Performance of Fuel Rods with No Cladding Through-Wall Defects

Prior to the discussion of the mechanisms that can result in fuel rod cladding rupture or in cladding mechanical property alterations, it is important to recognize the important role of the hydrogen, which is always present in the cladding. Hydrogen effects pose a concern for the behaviour of Zr alloys not only during in-reactor operations, but also after discharge of the spent fuel assemblies or bundles, particularly during dry storage and transportation. Factors contributing to this concern are:

- Hydrogen has limited solubility in zirconium alloys: Hydrogen in excess of the solubility limit precipitates as hydrides, which are brittle, having low critical stress intensity factors between 1 MPa√m to 5 MPa√m between 20 °C and 400 °C.

- Hydrogen moves under a temperature gradient to colder regions where it can create significant local concentrations. The flux of hydrogen in solid solution, J, driven by a temperature gradient, ∇T, is a function of the heat transport, Q^*, the hydrogen diffusion coefficient, D, and the concentration of hydrogen in solid solution, C; R is the ideal gas content.

  Equation 3-1: $$J = -D\left(\frac{Q^*C}{RT^2}\nabla T\right)$$

- Hydrogen also moves due to gradients in mean stress, from regions of low to high mean stress. The flux of hydrogen in solid solution, J, driven by a hydrostatic stress gradient, $\nabla\sigma_H$, is a function of the partial molar volume of hydrogen in the α-Zr, V_H (~1.67 cm^3/mol), the hydrogen diffusion coefficient, D, the concentration of hydrogen in solid solution, C, and the absolute temperature, T.

  Equation 3-2: $$J = D\left(\frac{V_H C}{RT}\nabla\sigma_H\right)$$

- Hydrogen can cause a growth of initial cracks by diffusion and re-precipitation of hydrogen at the tip of a notch under tensile stresses (delayed hydride cracking).

- Hydrides can adversely affect the toughness and tensile ductility of Zr-based components at temperatures that vary with concentration and morphology (orientation and effective continuity), but which may be within the temperature range expected for long-term dry storage.

Reviews of topics related to hydrogen and hydrides effects during dry storage can be found in numerous recent references, including, for examples, [ANT, 2016], [EPRI, 2020a], and [Motta et al., 2019].

3.1 Degradation Mechanism Driving Forces or Main Parameters

3.1.1 Cladding Stress

For fuel rods with no cladding through-wall defects, the transfer from the reactor environment to the spent-fuel pool results in the creation of tensile hoop stresses in the cladding, which are generated by the internal rod pressure that is no longer compensated by the water coolant pressure in the reactor.

3.1.1.1 Rod internal pressures

For PWR spent fuel rods, at a reference temperature of 25 °C and rod-average burnup ≤ ~62 GWd/MTU, the upper end-of-life (EOL) rod internal pressure (RIP) limits were determined to be:

- RIP < ~ 5 MPa for Standard PWR rods, based on experimental data reported in [Machiels et al., 2017] and updated in [Montgomery et al., 2019].

- RIP < ~6 MPa for Integral Fuel Burnable Absorber (IFBA) rods, based on limited experimental data [Pan et al., 2019], atomic scale modelling computations [Middleburgh et al., 2010], scenario modelling [Machiels, 2012], and computer code simulations using FRAPCON [Brown et al., 2004] [Richmond and Geelhood, 2018].

Cladding stresses can be then calculated using RIP values and cladding tubing dimensions, after taking into account any decrease in effective cladding wall thickness due to waterside corrosion.

Fuel performance codes generally assume that a gap is generated between the fuel column and the cladding ID upon transfer from the reactor to the spent-fuel pool. This is based on the relative changes in fuel and cladding diameters upon cooling from reactor operating conditions to wet storage conditions: thermal contraction of the fuel is more pronounced than that of the cladding tubing. However, gap reopening is not generally physically observed, especially for high-burnup fuel for which bonding is prevalent along the length of the fuel column except at pellet-pellet interfaces and toward the ends of the rod. Examination of spent fuel rod segments in hot cells show that the bonding between fuel and cladding is not visually broken by the transfer from the reactor to the spent-fuel pool. When defueling cladding segments is required for testing or PIE, defueling often requires mechanical drilling through the fuel column, followed by chemical dissolution of the residual fuel attached to the cladding in order to free the cladding from the bonded fuel. This means that the tensile stress created by the rod internal pressure may be lower than calculated due to some compensation by the compressive effects induced by the differential fuel-cladding contraction. Accurately calculating the contribution of the thermal contraction to the state of stress of the cladding is difficult because it would require knowledge of the displacement of the pellet fragments in response to the forces acting on the fuel column. Therefore, for simplicity, it is assumed that the free volume in the fuel column is located in the plenum and in the fuel-cladding gap, which permits straightforward calculations of the tensile stress existing in the cladding, using, for example, the thin-walled pressure vessel formula to calculate the cladding hoop stress, σ:

Equation 3-3: $$\sigma = P * D_m / 2 * t$$

where P is the internal pressure, D_m is the mean diameter, and t is the cladding thickness. The mean diameter is equal to the outside diameter minus the wall thickness.

At a given rod elevation, the stress is generally assumed to be constant in the annular cross section of the cladding, unless it is otherwise specified, for example in rods that are severely affected by spalling, blistering, grid-to-rod fretting, etc.

3.1.1.2   Stress gradients

During dry storage, incipient defects, or part-wall flaws, in cladding create stress gradients between the front, or tip, of the flaw and the bulk of the cladding. Discontinuities in pellet-cladding bonding and non-uniform mechanical interactions between pellet fragments and cladding can also lead to local stress gradients.

According to Equation 3-2, gradients in tensile stress can result in diffusion of the hydrogen in solid solution toward the higher-stressed cladding area. Accumulation of hydrogen at a crack tip can result in hydride precipitation and cracking, as will be discussed in Section 3.2.3 entitled Delayed Hydride Cracking.

3.1.1.3   Radiogenic helium production

After discharge from the reactor, the main source of additional gas in a fuel rod comes from alpha decay of the actinides. The alpha decay gas, helium, can either be retained inside the fuel pellet fragments or released into the rod free volume. For the latter, the released He can result in an increase in rod internal pressure over time, which can be partly or totally compensated by the concurrent decrease in fuel rod temperature. Given that cladding stress is proportional to the rod internal pressure, it is important to capture the effect of radiogenic helium build-up to properly calculate cladding hoop stress over time.

4 Conclusion

Conclusions for the most relevant mechanisms are contained in Section 3.2.1.6 ("Thermal creep"), Section 3.2.2.5 ("Hydride reorientation"), Section 3.2.3.5 ("Delayed hydride cracking"), Section 3.2.4.4 ("Hydrogen migration"), Section 3.2.5.2 ("Mechanical overload"), and Section 3.2.6.3 ("Impact of residual water"), and will not be repeated here. For a broader perspective, the results from a "Phenomena Identification and Ranking Table (PIRT) exercise for Used Fuel Cladding Performance" [EPRI, 2020b] will summarized in this section.

A PIRT exercise is a structured and facilitated expert elicitation process. Its objectives are to:

- Identify phenomena associated with the intended application,

- Rank the significance and current state of knowledge, and

- Characterize the potential effects of each phenomenon.

The specific PIRT objective was to review the current state of knowledge regarding spent fuel cladding integrity, which could lead to a recommendation for possible revisions to the U.S. regulatory guidance that is presently applicable to dry storage and transportation operations. The document focused on phenomena that could occur to used light water reactor oxide fuel (UO_2) and cladding (zirconium alloys) under normal and off-normal conditions during cask/canister loading, transfer, dry storage up to 60 years, and extended storage up to 100 years, and under normal conditions of transport.

The expert panel was comprised of six internationally recognized nuclear fuel experts from the following organizations: Oak Ridge National Laboratory (ORNL), Argonne National Laboratory (ANL), Pacific Northwest National Laboratory (PNNL), Structural Integrity Associates (SIA), and the U.S. Nuclear Regulatory Commission (NRC). An independent consultant, formerly EPRI employed, was also a member of the panel.

Evaluation of the postulated fuel rod degradation mechanisms included:

- **Operability**: Does the phenomenon exist with effects that are detectable for the scenarios that are considered? The answer was either yes or no (Y/N).

- **Knowledge**: Are data on the phenomenon available, and are they relevant? The ranking was low, medium, or high.

- **Confidence**: What is the quality of the existing data and models? The ranking was low, medium, or high.

- **Significance**: To what extent does the phenomenon contribute to gross cladding rupture? The ranking was low, medium, or high.

The results from the PIRT process for a specific degradation mechanism, thermal creep is shown in Table 4-1, while Table 4-2 summarizes the "Significance" ranking for nine mechanisms that are covered in the present report. As can be readily seen, the group of experts reached a consensus position that most of the mechanisms had a "Low" significance associated with the five phenomena/scenarios that were considered. Two mechanisms, "Hydride reorientation" and "Mechanical Overload", received a couple of "Medium" significance, which were associated with the storage scenarios of up to 60 and up to 100 years. One scenario ("Fuel oxidation") did not show much of a consensus position, possibly because it did not clearly distinguish between air ingress and residual water.

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