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# Cladding stress during extended storage of high burnup spent nuclear fuel

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A R T I C L E I N F O

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A B S T R A C T

In an effort to assess the potential for low temperature creep and delayed hydride cracking failures in high burnup spent fuel cladding during extended dry storage, the U.S. NRC analytical fuel performance tools were used to predict cladding stress during a 300 year dry storage period for $UO_2$ fuel burned up to 65 GWd/MTU. Fuel swelling correlations were developed and used along with decay gas production and release fractions to produce circumferential average cladding stress predictions with the FRAPCON-3.5 fuel performance code. The resulting stresses did not result in cladding creep failures. The maximum creep strains accumulated were on the order of 0.54–1.04%, but creep failures are not expected below at least 2% strain. The potential for delayed hydride cracking was assessed by calculating the critical flaw size required to trigger this failure mechanism. The critical flaw size far exceeded any realistic flaw expected in spent fuel at end of reactor life.

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## 1. Introduction and background

Gap analyses by NRC [1] and DOE [2] have identified low temperature creep (LTC) and delayed hydride cracking (DHC) as potential cladding breach mechanisms after ~100 years of dry storage. Both of these mechanisms require the presence of a stress in order to be active. Predicting the cladding stress over an extended storage period is necessary to evaluate the potential breach of the spent fuel cladding due to LTC and DHC during long term spent fuel storage, from 100 to 300 years.

Potential sources of stress are plenum gas pressure, phase change of the hydrides upon cooling from drying temperatures [3], and swelling of the fuel due to a buildup of helium decay product (resulting in Pellet–Cladding Mechanical Interaction – PCMI).

Although the rod pressure at the end of irradiation may be too low to drive these cladding breach mechanisms, helium production due to alpha decay and PCMI induced by swelling of the pellets via a buildup of helium have been proposed as sources for cladding stress [4]. Until stress levels in the cladding are evaluated over the extended storage period, it is unclear whether LTC or DHC cause a regulatory concern. The present study consists of cladding stress predictions over a period of 300 years of spent fuel dry storage for fuel burned to 65 GWd/MTU for different fuel designs having different power histories. The predictions account for both gas production in spent fuel and fuel pellet swelling during storage.

The calculations performed in this study do not account for any local stress concentration that might be caused by pellet–pellet interfaces, pellet fragment interfaces, friction forces that may arise between pellet fragments as the fuel swells during storage, or any circumferential heterogeneity in pellet-to-cladding mechanical interaction. Consequently, the cladding stresses calculated in this study represent a circumferential average cladding stress. As discussed in Section 4.2, this limitation is not expected to have had a major impact on the results of the study, mainly due to the fact that significant pellet inter-fragment volume was predicted to exist during the dry storage period.

## 2. Phenomenology and analytical approach

The goal of this study was to determine whether sufficient cladding stress would develop in high burnup spent fuel over a period of long term dry storage that could lead to cladding failure due to creep or static overload. Fuel pellet swelling correlations were developed and subsequently used to determine the total cladding stress and associated critical flaw size for DHC initiation, as well as predict the cladding strain accumulated during 300 years in dry storage. A modified version of the NRC's steady-state fuel performance code FRAPCON-3.5 was used to perform this analytical study.

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[1] Since writing this paper, Dr. Einziger has moved on to become a Senior Professional Staff member at the Nuclear Waste Technical Review Board.

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*2.1. Modeled sequence and cladding temperature history during dry storage*

The sequence that was modeled in this study to determine cladding stress in high burnup fuel over an extended storage period of 300 years starts at reactor shutdown then includes pool storage followed by transfer to a dry storage canister, drying, and finally 295 years in dry storage. After irradiation to 65 GWd/MTU, the fuel rods are stored in the spent fuel pool for 5 years, where the decay heat produced by the fuel rods is removed by active cooling, such that the temperature of the cladding remains at or below 353.15 K.

It is interesting to note that upon cooldown from reactor temperatures, FRAPCON predicted that the fuel-to-cladding gap reopens because the thermal contraction of the pellet exceeds that of the cladding. In reality, the physical gap between the pellet and the cladding does not reopen, particularly for higher burnup fuel where the fuel is bonded to the cladding. Nonetheless, the differential thermal contraction of the pellet and cladding opens up some inter-fragment volume inside the fuel pellet. More details on this predicted phenomenon are provided in Section 4.2. It is also important to note that low temperature[2] decay gas production (predominantly helium) and fuel pellet swelling, both primarily due to alpha decay of transuranic species and fission product decay, begin as soon as the reactor is shut down and the fuel rods enter storage.

After 5 years in the spent fuel pool, the fuel is transferred to a Dry Storage Canister (DSC) and undergoes a short drying phase during which the temperature is limited to 673.15 K. Once dry, the storage canister is sealed and filled with helium gas prior to being moved to the Independent Spent Fuel Storage Installation (ISFSI). In the ISFSI, the DSC is stored either in a horizontal storage module primarily made of concrete, or in a vertical overpack either made of metal or concrete, or both. In both the horizontal and the vertical storage configurations, the outer surface of the DSC is exposed to ambient air and cooled by natural circulation of air through the overpack and around the DSC. The cladding temperature during dry storage is given by Eq. (1), where the initial storage temperature is 673.15 K, which is the highest allowed per ISG-11, Revision 3 [5]. Eq. (1) was derived from fitting the dry storage cladding temperature history used to evaluate radial hydride precipitation in Appendix D, pages 13 and 32 of Ref. [6].

$$T(\mathrm{K}) = \begin{cases} 0.265714 \times t^2 - 12.3343 \times t + 673.15 & \text{if } t < 13.0306 \text{ years} \\ -59.2015 \times \ln(t) + 709.532 & \text{if } t \geqslant 13.0306 \text{ years} \end{cases} \quad (1)$$

With $T$ the peak cladding temperature in Kelvin, and $t$ the time in storage, expressed in years.

During the time when the temperature in the DSC is above 573.15 K, the cladding rapidly creeps out and away from the pellet due to the rod gas pressure, resulting in a significant increase in the predicted gap size, and corresponding in reality to an increase in pellet inter-fragment volume, as described in more detail in Section 4.2. Once the temperature drops below ~523.15 K, cladding creep is close to zero and the cladding plastic deformation no longer evolves. This arrest in creep deformation could potentially result in a slow buildup of cladding stress due to decay gas production and release, combined with the reduction in void volume as the fuel pellets are swelling. However, this potential stress increase is counteracted by the temperature decrease during the storage period, such that the stress increase is only predicted if it is assumed that more than 10–15% of the decay gas is released, which is highly conservative and unrealistic (see Section 4.1.2). Eventually, if the pellets swell sufficiently to come in contact with the cladding, a Pellet-to-Cladding Mechanical Interaction (PCMI) stress could also develop, although this was not predicted to occur in this study.

This paper describes the results obtained for two of the most common fuel designs in operation today: PWR 17 × 17 fuel and BWR 10 × 10 fuel. In both cases, steady-state irradiation was simulated with FRAPCON to a burnup of 65 GWd/MTU. For each design, tens of power histories were modeled so as to capture the variations associated with power history for a same discharge burnup of 65 GWd/MTU.

*2.2. Fuel pellet swelling during extended spent fuel storage*

A literature survey was performed in search for information and data related to long term swelling of spent nuclear fuel (or surrogate materials). The fuel swelling is caused by alpha-decay of the radioactive nuclides in spent fuel, where the alpha particles knock on atoms in the $UO_2$ lattice. The resulting recoil nuclei result in displacement cascades and in the creation of Frenkel pairs, defects that ultimately result in lattice swelling. There was overwhelming agreement in the literature that fuel swelling due to self-irradiation saturates over time (as dose and damage are accumulated) [4,7–13]. The assumption of saturation of fuel swelling was used in the development of the swelling correlations used in this study, as reflected in Eq. (2). In addition, swelling behavior appears to be independent of dose rate within the range of specific activities relevant for LWR spent fuel [10], which supports the use of surrogates to study the accelerated aging of spent fuel.

Two types of swelling data were found in the literature survey. In both cases, the parameter measured by the experimentalists was the lattice relative expansion $\Delta a/a_0$. The relative lattice expansion parameter was either expressed as a function of displacements-per-atom (dpa) (this was usually the case for studies where only experimental data was provided, without any correlation developed to fit the data), or as a function of the cumulative α particle dose (expressed in α particles per atom: $\lambda t$, where $\lambda$ is the decay constant in α decays per atom per second, and $t$ is time in seconds). The general form of the equations used to fit data is given in Equation (2). Whenever swelling was given in the literature as a function of dose instead of displacements per atom (dpa), a conversion from dose to dpa was performed using Eq. (3), based on calculations reported in [14].

$$\frac{\Delta a}{a_0} = A \cdot (1 - e^{-B\ \mathrm{dpa}}) \quad (2)$$

where $a$ is the lattice parameter, $a_0$ is the undeformed lattice parameter, and $A$ and $B$ are constants to be determined.

$$\frac{\mathrm{dose(dpa)}}{\mathrm{dose}(\alpha/\mathrm{g})} = \mathrm{Constant} = 2.50 \times 10^{-19} \quad (3)$$

The swelling data gathered in the literature survey were analyzed and a best-estimate swelling correlation and an upper-limit swelling correlation were developed, given in Eqs. (4) and (5). The parameters $A$ and $B$ for Eq. (2) for the best estimate (average) and bounding (maximum) correlations are provided in Table 1 and shown along with all the data and curve fits relevant for extended storage of spent nuclear fuel in Fig. 1. It should be noted that one data point is not bounded by the 'bounding' correlation (Rondinella's $UO_2$ with 10% $238PuO_2$), but this was deemed acceptable because that particular set of data then saturates at a level below the bounding swelling curve. The relationship between time and dpa (shown in Fig. 1) was derived by fitting lines from [15] for $UO_2$ fuel at 60 GWd/MTU, and is given by Eq. (6).

$$\frac{\Delta a}{a_0} = 3.528 \times 10^{-3} \cdot (1 - e^{-8.492\ \mathrm{dpa}})(\text{Best-estimate}) \quad (4)$$

[2] 'Low temperature' used in this context implies dry storage temperatures, as opposed to the high temperatures used in specimen annealing tests where gas release is measured as a function of time or temperature (typically between 1000 K and 2500 K).