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Rod internal pressure of spent nuclear fuel and its effects on cladding degradation during dry storage

Ju-Seong Kim*, Jong-Dae Hong, Yong-Sik Yang, Dong-Hak Kook

Korea Atomic Energy Research Institute, 989-111 Daedeokdae-Ro, Yuseong-Gu, Daejeon 34057, Republic of Korea

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ABSTRACT

Temperature and hoop stress limits have been used to prevent the gross rupture of spent nuclear fuel during dry storage. The stress due to rod internal pressure can induce cladding degradation such as creep, hydride reorientation, and delayed hydride cracking. Creep is a self-limiting phenomenon in a dry storage system; in contrast, hydride reorientation and delayed hydride cracking are potential degradation mechanisms activated at low temperatures when the cladding material is brittle. In this work, a conservative rod internal pressure and corresponding hoop stress were calculated using FRAPCON-4.0 fuel performance code. Based on the hoop stresses during storage, a study on the onset of hydride reorientation and delayed hydride cracking in spent nuclear fuel was conducted under the current storage guidelines. Hydride reorientation is hard to occur in most of the low burn-up fuel while some high burn-up fuel can experience hydride reorientation, but their effect may not be significant. On the other hand, delayed hydride cracking will not occur in spent nuclear fuel from pressurized water reactor; however, there is a lack of confirmatory data on threshold intensity factor for delayed hydride cracking and crack size distribution in the fuel.

1. Introduction

In general, spent nuclear fuels (SNFs) are burned in a nuclear reactor for around 4.5 years. As of 2013, the number of fuel assemblies (FAs) from pressurized water reactor (PWR) stored in wet storage at a nuclear power plant in Korea is around 10500. Among them, the average burn-up of FAs lower than 45 GWd/tU is about 85%. Each of the fuel assemblies has an intrinsic burn-up, which is mainly dependent upon the loading pattern and power history. The burn-up level of SNF can be simply classified into two groups: high burn-up fuel (HBU) is greater than the average fuel assembly (FA) burn-up of 45 GWd/tU, and low burn-up (LBU) fuel is equal to or less than that of 45 GWd/tU. Recently, the storage of SNF from PWRs in Korea has become a critical issue owing to the expected saturation of wet storage capacity in the near future. As an alternative, dry storage is considered to be a realistic option prior to the pyro-processing or final disposal of the SNFs.

The technical storage criteria of the United States (US) on spent fuel in dry storage are based on 10 CFR 72. By this regulation, the SNF must be protected against a gross rupture and maintain its retrievability indicating that the integrity of the SNF will remain intact during a short-term storage operation, long-term storage, and even transportation. In order to preclude creep rupture of the cladding during dry storage, temperature limit of the cladding was proposed in NUREG-1536 [1]. At that time, creep was a primary failure mechanism during dry storage. The US Nuclear Regulatory Commission (NRC) recommended a specific peak cladding temperature during normal storage dependent upon the fuel burn-up and cooling period in wet storage, and also proposed a temperature limit of 570 °C under off-normal and accident conditions. As an accumulation of creep data on irradiated cladding, the NRC proposed 1% creep strain of cladding instead of the peak temperature of cladding in ISG-11 rev. 1 [2]. However, in ISG-11 rev. 2 [3], the NRC tentatively concluded that creep rupture is hard to occur under the dry storage conditions and cannot severely degrade the integrity of cladding if the peak cladding temperature can be maintained below 400 °C, but they considered the detrimental effects of hydride reorientation (HR) on the ductility. In their third revision, ISG-11 rev. 3 [4], NRC limits the number of thermal cycles to less than 10 to prevent HR in cladding material; moreover, they allow high temperature greater than 400 °C for LBU fuel if the cladding stress is equal to or less than 90 MPa [5]. In other words, the current

* Corresponding author.
E-mail address: juskim@kaeri.re.kr (J.-S. Kim).

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storage guidelines of NRC is maintaining the cladding hoop stress of less than 90 MPa for the LBU fuel and a peak cladding temperature of less than 400 °C for the HBU fuel [5].

The current lifetime of the dry storage system in the US is up to 80 years: however, the US has considered very long-term storage beyond 120 years and even up to 300 years including the wet storage period. As the needs of very long-term storage of spent nuclear fuel increase, NRC [6], the Electric Power Research Institute (EPRI) [7], Department of Energy (DOE) [8] and Nuclear Waste Technical Review Board (NWTRB) [9] reviewed the operable degradation mechanisms in SNF during long-term dry storage and analyzed a technical gap. In their reports, creep is recognized as a minor degradation mechanism because the creep is a self-limiting mechanism in the dry storage system. On the other hand, hydrogen effects, i.e., HR, hydride embrittlement, and delayed hydride cracking (DHC) have been pronounced as major degradation mechanisms during storage. DHC is a time-dependent degradation phenomenon degrading the integrity of fuel cladding with a gross rupture. DHC has been suspected as a failure mechanism of Zr-2.5Nb [10,11] and Zircaloy-2; in contrast, DHC has not been observed in PWR cladding material, and thus the onset of DHC was not of interests for PWR cladding. However, owing to the extension of the storage period, hydride effects including DHC are reconsidered in PWR cladding as major potential degradation mechanisms [12]. HR is not a direct failure mechanism but it can degrade the integrity of material with even a small hydrogen concentration. HR occurs when excessive stress (so-called threshold stress) and soluble hydrogen exist in cladding during a cool-down. Unfortunately, some of the hoop stresses of HBU cladding may be close to the threshold stress for HR under the dry storage condition; therefore, most of the concern of HR is related to the possibility of the onset of HR and their effects on the material ductility.

The objectives of this study are to assess which failure mechanisms can deteriorate the cladding integrity, and what guidelines will be required for the mitigation of the major potential degradation in SNF cladding. In dry storage, there are no controllable variables except temperature due to the intrinsic passive storage system, and thus, the RIP and the corresponding hoop stress of cladding at the initial dry storage have a prime importance in the assessment of the integrity of SNFs. However, few available data on RIP at high temperatures and corresponding hoop stress have been presented in literature because we can measure RIP only in a hot cell through a perforation. In this study, the maximum RIP and hoop stresses of Korean SNFs were calculated using FRAPCON-4.0 with certain assumptions. Subsequently, the onset of HR and DHC was discussed based on the current guidelines of the NRC.

2. Calculation of rod internal pressure and corresponding hoop stress of Korean spent nuclear fuel

In this study, RIPs and hoop stress of SNF were calculated using FRAPCON-4.0. FRAPCON is a fuel performance evaluation code during steady-state operation of light water reactors, which calculates rod internal pressure, deformation of cladding, cladding oxidation and pellet cladding mechanical interaction (PCMI). The previous version, FRAPCON-3, had been steadily updated [13–15], and recently, FRAPCON-4.0 was released [16].

In the RIP calculation, two types of PWR fuel rod designs were considered: 16 × 16 Combustion Engineering (CE), and 17 × 17 Westinghouse (WH) types, whose design parameters are given in Table 1. The major differences in design parameters of the fuel are plenum length and initially charged helium pressure. Power history of the fuel rod is a prerequisite for RIP calculation. The previous study showed that power history can affect the RIP significantly at high burn-up [17]. In particular, the release of fission gases can be increased when the heat generation rate of fuel in the last cycle is high. It is obvious that hoop stress of SNF cladding value is the most conservative when RIP is equal to the system pressure at the end of reactor operation. The power histories used in this study are presented in Fig. 1. For conservative RIP calculation, case A power history was used for which the linear heat generation rate (LHGR) of last cycle operation is higher; however, RIP can exceed the system pressure at a higher burn-up greater than 50 GWd/tU when case A power history was used. In that case, we used case B power history and adjusted the RIP around the system pressure of 15.5 MPa at the end of cycle (EOC) using the bias option in the code. In order to simulate five years wet storage and vacuum drying operation prior to dry storage, the cladding temperature was set at 25 °C for five years and the cladding temperature was then changed to 400 °C.

The RIP calculation was conducted with a burn-up range of 30–60 GWd/tU with some assumptions. Firstly, uniform temperature in the entire fuel rod was assumed for the stress calculation during the vacuum drying operation, i.e., no axial temperature variation in the axial direction. The RIP of SNF is predominantly dependent upon the plenum volume and its temperature, but generally, the plenum position is not the peak temperature location in dry storage operation; therefore, non-uniform temperature distribution can give best-estimate results but a uniform temperature assumption is rather conservative in evaluating the hoop stress of the cladding. Secondly, additional gas release and swelling due to helium production is not considered after the discharging. A recent study [18] showed that both the fuel swelling due to helium and the release of helium might be negligible during the dry storage period if the fuel temperature is maintained below 700 K, and thus, considering the storage temperature history the release of helium might be negligible. Thirdly, we assumed open gap regime of the fuel rod. Practically, the gap between fuel and cladding is closed during the reactor operation, and the bonding with fuel and cladding is observed in SNF; however, the gap will be re-open due to outward creep deformation.

3. Rod internal pressure and corresponding hoop stress of spent nuclear fuel

Recently EPRI analyzed the non-proprietary RIP data independent of fuel type. According to their results, RIP values tend to begin to rise exponentially at 60 GWd/tU and following empirical correlation between RIP and burn-up was proposed for burn-up less than 60 GWd/tU.

$$\mathrm{RIP(MPa)} = 2.8781 + 0.0224 \times \mathrm{Burn-up(GWd/tU)}$$

However, the given RIP does not represent the conservative value. Fig. 2 shows the calculated RIP with reference data [17,19].

Table 1
Design parameters of fuel rods.

Parameter	17 × 17 WH	16 × 16 CE
Fuel pellet stack length	3657 mm	3810 mm
Fuel enrichment	4.5%	4.5%
Pellet density	94.8%	95.2%
Pellet diameter	8.19 mm	8.25 mm
Pellet length	9.82 mm	9.90 mm
Plenum length	185 mm	245 mm
Cladding material	Zircaloy-4	Zircaloy-4
Cladding outside diameter	9.5 mm	9.7 mm
Cladding thickness	0.57 mm	0.65 mm
Helium charging pressure	1.98 MPa	2.62 MPa