Annals of Nuclear Energy 119 (2018) 129–138

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Calculation chain for the analysis of spent nuclear fuel in long-term interim dry storage

Asko Arkoma *, Risto Huhtanen, Jaakko Leppänen, Juho Peltola, Timo Pättikangas

VTT Technical Research Centre of Finland Ltd., Kivimiehentie 3, P.O. Box 1000, FI-02044 VTT, Finland

ARTICLE INFO

Article history:
Received 1 December 2017
Received in revised form 20 April 2018
Accepted 22 April 2018

Keywords:
Dry storage
Creep
Fuel behaviour
OpenFOAM
Serpent
VTT-ENIGMA

ABSTRACT

In this paper, a calculation chain for the analysis of spent nuclear fuel in long-term interim dry storage is demonstrated. The calculation chain consists of reactor physics burnup simulation for obtaining the decay heat, Computational Fluid Dynamics (CFD) simulation for the temperatures of fuel rods and the dry storage cask, and finally, cladding integrity analysis. Each succeeding calculation step uses the results from the previous step as boundary conditions. The applied codes are Serpent Monte Carlo reactor physics burnup code, open-source OpenFOAM CFD code, and VTT-ENIGMA fuel performance code. The peak cladding temperatures are calculated with OpenFOAM at different instants of time, ranging from 3.4 to 300 years after unloading from the reactor. The chosen demonstration case considers 17 × 17 PWR fuel stored in a CASTOR® V/21 type dry storage cask. The main parameters of interest in the last stage of the analysis are the cladding creep hoop strain and stress during dry storage. The developed analysis methodology helps to ensure the safety of long-term dry storage.

1. Introduction

Spent fuel rods from nuclear power plants have to be contained in interim storages for several decades before the final disposal. Even longer times up to 100–300 years have been considered (Raynaud and Einziger, 2015). The safety aspects of prolonged interim storage include radiation shielding, criticality safety, containment and structural safety, and heat removal capacity. The interim storage can be either water or air-cooled, i.e., the fuel assemblies can be stored in water pool or in dry storage casks, respectively. This study focuses on heat removal capacity and the long-term fuel behaviour during interim dry storage.

From the fuel rod point of view, the most important issues to be considered during the dry storage are the cladding creep and the behaviour of hydrides in the cladding. Hydrides may induce failures such as delayed hydride cracking, and these also embrittle the cladding which complicates fuel handling.

The loading caused by the difference between the fuel rod external and internal pressures may result in creep-out of the cladding and eventually to creep rupture. The stress in the cladding is dependent on the fuel temperature (decay heat), internal overpressure enhanced by the fission gases released during the irradiation, fuel rod dimensions, cladding oxidation during the irradiation, and dry storage cask pressure, which is different in various designs. As the decay heat diminishes, the stress slowly relaxes. Thus, the creep rate is negligible after a few years of dry storage.

Radially oriented hydrides are detrimental to cladding tensile ductility in the hoop direction. Significant dissolution of hydrides occurs during drying and at the beginning of dry storage. Precipitation follows later on as the fuel cools down. Hydrides tend to reorient perpendicular to the direction of prevailing tensile hoop stress, and thus the preferred orientation is radial (Colas et al., 2013). The amount of hydrides that precipitate radially is a function of many parameters: hoop stress, cooling rate, initial hydrogen content, solubility of hydrogen in a given temperature, number of thermal cycles, and cladding material properties (Min et al., 2014; U.S.NRC, 2010).

In order to limit the possibility of cladding rupture resulting from creep and hydride effects in dry storage, cladding stress and temperature are constrained. The regulations vary from country to country: for example in USA, a maximum temperature limit of 400 °C under dry storage and the preceding drying and backfilling has been set (U.S.NRC, 2010). If the burnup is less than 45 MWd/kgU, the temperature limit may be exceeded if the hoop stress limit of 90 MPa is met (U.S.NRC, 2010). The 90 MPa limit is set to minimize the hydride radial reorientation (U.S.NRC, 2010). Specific safety limit for radial reorientation in irradiated cladding is not accurately known. It might require stresses higher than 120 MPa (U.S.NRC, 2010), while in another study, the threshold for

* Corresponding author.
E-mail address: asko.arkoma@vtt.fi (A. Arkoma).

https://doi.org/10.1016/j.anucene.2018.04.037
0306-4549/

Nomenclature

CEA	Le Commissariat à l'énergie atomique et aux énergies alternatives
CFD	Computational Fluid Dynamics
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas
EDF	Électricité de France
IFPE	International Fuel Performance Experiments database
NNL	National Nuclear Laboratory, United Kingdom
PWR	Pressurized Water Reactor
STL	STereo Lithography, file format used in 3D description
VTT	VTT Technical Research Centre of Finland Ltd.
ε	strain
σ	stress
k-ω SST	k-ω Shear Stress Transport turbulence model

Zircaloy-4 from high burnup rod (average rod burnup 67 MWd/kgU) was determined as low as 75 ± 12 MPa at 400 °C (Daum et al., 2005, 2006). The safety limit can also be defined in terms of hoop strain, in which case 1% limit is considered in some countries (Kim et al., 2010).

Several analyses have been conducted regarding cladding dry storage creep (Feria et al., 2015; Rossiter, 2011), stress (Feria and Herranz, 2017; Kim et al., 2017; Raynaud and Einziger, 2015) and hydride behaviour (Stafford, 2015), and heat transfer CFD analysis (Herranz et al., 2015; Penalva et al., 2014; Tseng et al., 2016; Yoo et al., 2010). In general, the available cladding creep models in fuel performance codes are developed for in-reactor conditions and are not applicable as such for modelling out-of-reactor behaviour. In order to study the creep-out during the interim dry storage, the cladding creep model of VTT-ENIGMA code (Kilgour et al., 1992) is extended to describe the dry storage creep.

For the fuel performance analysis, the temperature evolution during dry storage is needed. EPRI has presented a conservative approximation for temperature evolution (EPRI, 2007), and CIEMAT has correlated the temperature with burnup and storage time based on CFD simulations (Feria et al., 2015). In the current study, the OpenFOAM (OpenFOAM Foundation, 2015) CFD code is used for obtaining the peak cladding temperatures within the cask at various instants of time during dry storage. OpenFOAM solves the heat transfer inside the cask and to the surrounding environment. For obtaining the decay heat needed by the CFD code, Serpent Monte Carlo reactor physics burnup calculation code (Leppänen et al., 2015) is used. By applying the whole calculation chain, the needs for approximate decay heat or thermal evolutions are eluded. The selected simulation case presented in this paper serves as a demonstration of the calculation chain.

The paper is structured as follows. The applied modelling codes and methods, as well as the selected demonstration case are described in Section 2. The results and discussion are given in Section 3, and the final remarks in Section 4.

2. Codes, methods and the demonstration case

2.1. Demonstration case

In the demonstration case, a single cask with its surroundings is considered. The details of the selected cask, CASTOR® V/21, and earlier evaluations are documented in an EPRI report (Dziadosz and Moore, 1986). A presentation of the cask is shown in Fig. 1. The hermetically sealed cask is made of cast iron, and the separate basket inside the cask is made of borated steel. Absorber rods for the radiation shielding are located inside the boreholes of the outer shell of the cask.

Each dry storage cask contains several fuel assemblies. The main fuel-related parameters are adopted from the BEAVRS benchmark (Horelik and Herman, 2013), and are given in Table 1. This benchmark is chosen because it provides very detailed descriptions of commonly used PWR fuel assembly types. The PWR assembly design is 17 × 17, with 24 empty guide tubes. In order to have a representative burnup from the intermediate burnup range, an average burnup of 50 MWd/kgU is applied at all successive stages of the calculation chain (assembly average in Serpent, and rod average in VTT-ENIGMA). As the calculation system consists of several modelling codes of varying fidelity, specific assumptions made with respect to this common demonstration case are brought out in the following subsections of Section 2.

2.2. Reactor physics: Serpent

Serpent is a Monte Carlo neutron and photon transport code developed at VTT since 2004 (Leppänen et al., 2015). The code is specialized in reactor physics applications, and it is capable of

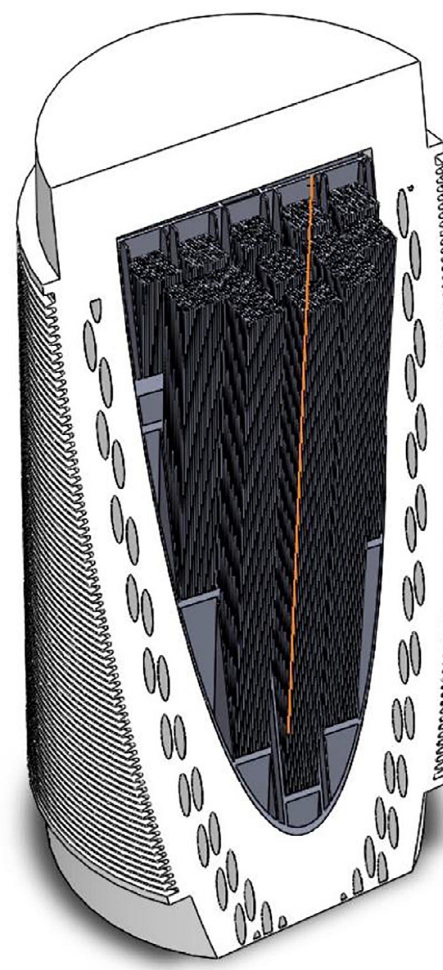

Fig. 1. Layout of the dry storage cask. Fuel assemblies, basket inside the cask, and the absorber rods are shown.