Spanish R&D in dry storage of spent fuel: outcomes and outlook

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Abstract

The Spanish R&D efforts in dry interim storage of spent nuclear fuel (SNF) are mainly focused on supporting safety under storage and transportation, including intermediate operations. Major actors involved are research entities, academia, the waste management organization and industry. Experimental and modelling activities are carried out to understand and predict the SNF response under the prevailing conditions. Particularly, thermal, chemical, mechanical and neutronics investigations are performed. The main outcomes achieved as well as the outlook in each research line are related to improvements in the characterization of dry stored SNF, due to its essential importance for the safety assessment of the back-end fuel cycle.

Keywords: dry storage, spent fuel, Spanish R&D

1. Introduction

The radioactive waste generated from Nuclear Power Plants (NPPs) requires treatment, confinement and storage systems. This fact must be consistent with specific safety, environmental and human protection standards. In case of the irradiated fuel, due to its high heat load and radioactivity, it is initially stored in onsite spent fuel water pools to provide both cooling and shielding of the fuel elements. Thereafter, the technology used for the management of the irradiated fuel depends on the strategy followed by each country.

In Spain, the fuel is not reused once irradiated (open cycle). The strategy followed once the pools reach their maximum capacity is based on interim dry storages, given the lack of a final deep repository or a central temporal repository. The Spent Nuclear Fuel (SNF) is managed by ENRESA (Spanish Radioactive Waste Management Company) and comes from seven commercial nuclear power reactors under operation (six Pressurized Water Reactors (PWRs) and one Boiling Water Reactor (BWR)), as well as from another BWR reactor that has recently entered permanent shutdown and two additional power reactors that were decommissioned in the last decades (one PWR and one gas-cooled reactor). Table 1 summarizes the current status of the NPPs in Spain.

The Spanish policy on radioactive waste management is set in the Radioactive Waste General Plan (PGRR). Based on the one currently in force (6th, approved in 2006) (Ministerio de Industria y Turismo, 2006), the spent fuel shall be managed in a Centralized Interim Storage facility (CTS). Additionally, Interim Spent Fuel Storage Installations (ISFSI) are being built at the reactor sites (first one licensed in 2002) to increase the spent fuel storage capability onsite. In case of the ISFSI, two types of dry storage systems are already in use: dual-purpose bolted metal casks containing bare fuel (DPT, ENUN, HI-STAR) and welded canisters in concrete modules (HI-STORM); the former also valid for transportation (to the CTS or the final repository). Details of cask types in each NPP are shown in Table 1.

Table 1: NPPs (ISFSI included) that have been operating in Spain.

NPP	**Reactor supplier**	**Reactor type**	**Shutdown date**	**ISFSI Cask type**
Vandellós II	Westinghouse	PWR	2035*	-
Vandellós I	HIFRENSA	GCR	1990	
Cofrentes	GeneralElectric	BWR-6	2030*	HI-STAR 150
Trillo I	KWU	PWR	2035*	DPT & ENUN-32P
Sta Maria de Garoña	GeneralElectric	BWR-3	2017	ENUN-52B
José Cabrera	Westinghouse	PWR	2006	HI-STORM-100Z
Ascó I	Westinghouse	PWR	2030*	HI-STORM 100
Ascó II	Westinghouse	PWR	2032*	
Almaraz I	Westinghouse	PWR	2027*	ENUN-32P
Almaraz II	Westinghouse	PWR	2028*	

*Planned shutdown dates

Currently, a revision of the PGRR is being drafted for Government approval that will update the Spanish status considering an agreed NPPs shutdown schedule (Monforte, 2019). According to the draft of the 7th PGRR (Ministerio para la Transición Ecológica y el Reto Demográfico (MITECO), 2022) released in 2022, the total inventory of spent fuel assemblies in Spain as of December 31, 2021, was 17,065. Among these, 14,461 assemblies were stored in spent fuel pools, while 2,461 were housed in dry storage canisters. It is important to note that these figures are subject to change as more fuel is discharged from reactors and subsequently sent to storage. Furthermore, the PGRR estimates the production of approximately 4,600 additional fuel assemblies until the reactors expected end-of-life. Recently, the Ministry for Ecological Transition has ruled out the construction of the CTS in Spain, so that Decentralized Temporary Storage Sites are planned on the sites of nuclear power plants.

From the regulatory point of view, the Spanish requirements go through the following key safety functions for storage under normal, off-normal conditions and design basis accidents (DBA) (CSN, 2009): retrievability (SNF should remain retrievable), heat removal (the storage system must provide for heat removal), confinement (the storage system must maintain confinement of the radioactive material), radiation shielding (the storage systems must provide for controlling and limiting occupational radiation exposures within the regulatory limits) and subcriticality must be demonstrated. The same safety functions apply to transportation. Additionally, dry storage systems at the plant sites must be licensed for both storage and transportation before any SNF can be loaded in the cask.

In this context, the research and development (R&D) activities from the Spanish organizations (Research Centers, Universities and Industry) are mainly focused on supporting the continued safe dry storage and transportation of SNF. This paper summarizes the main contributions from different organizations. Table 2 shows the main actors involved and classifies them according to the sort of activity carried out: experimental or modelling (methodologies and data management included). The paper is split into three main parts: research synthesis, key outcomes and future developments.

Table 2: Main actors in R&D activities in Spain.

	EXPERIMENTAL	**MODELLING**
INDUSTRY	**Enusa** Industrias Avanzadas, S.A.S.M.E.	**Enusa** Industrias Avanzadas, S.A.S.M.E. **IDOM** Consulting, Engineering, Architecture Equipos Nucleares, S.A. (**ENSA**)
RESEARCH ENTITY	CIEMAT's High Level Waste Unit (**CIEMAT-HLWU**)	CIEMAT's Unit of Nuclear Safety Research (**CIEMAT-UNSR**)
UNIVERSITY	Universidad Politécnica de Madrid - Escuela de Ingenieros de Caminos, Canales y Puertos (**UPM-ETSI Caminos**)	Universidad Politécnica de Madrid - Escuela Técnica Superior de Ingenieros Industriales (**UPM-ETSII**)

2. Research synthesis

2.1. Research entities

2.1.1. CIEMAT-HLWU

The challenge of the CIEMAT-HLWU is to study the effect of key parameters that promote UO_2 oxidation. If the fuel pellet is exposed to an oxidizing atmosphere under certain temperatures, something that could happen at normal or accidental scenarios, the fuel matrix (UO_2) oxidation may occur (Sjöland et al., 2023). If oxidation progresses up to the formation of U_3O_8, it results in a 36% volume increase (Taylor et al., 1989), leading to fuel grain de-cohesion and spallation of oxidized product. In the event of a defected fuel cladding, this could entail the release of radioactive material to the environment.

In order to reach the general purpose of this research, the fundamental objectives are the following:

- Developing and implementing both traditional and advanced methods for characterizing and monitoring the oxidation of non-irradiated UO_2 fuel to improve the understanding of its kinetics and main parameters affecting its potential degradation (Milena-Pérez et al., 2023a). These demonstrated protocols could be then applied to irradiated fuel, which samples are relatively limited and whose handling is subjected to strict safety criteria and regulation.
- Maintaining and expanding the databases of fuel oxidation compounds, as a way to fulfil oxidation knowledge gaps that subject the fuel matrix to experimental oxidation conditions.
- Performing experiments to address the study of the different decoupled and coupled effects affecting SNF oxidation.
- Understanding the multiparametric effects (with the focus on temperature and oxygen percentage) on chemical oxidation resistance of UO_2 to U_3O_8 as a consequence of:
    - The $UO_2/ZrO_2/Zr$ system from cladding (in case of a potential pellet-cladding contact).
    - The impact of trivalent and tetravalent dopants (fission products and actinides).
    - The influence of induced change in the morphological and mechanical properties of the fuel (cracks, density, porosity or grain size).

The study of the different parameters that may affect this reaction, such as temperature and oxygen partial pressure, is of interest for irradiated fuel in dry interim storage of SNF scenarios, including intermediate handling operations, under both, normal and accidental scenarios. The scope details that this study explores are the impact of time-dependence of variables coupled and decoupled by means of a combination of traditional and novel techniques:

- Extrinsic factors governing the oxidation: time, temperature, oxygen partial pressure and moisture.
- Intrinsic parameters or attributes associated with the fuel properties: fuel chemistry (simulated burnup), morphometric properties such as particle and grain size, morphology, specific surface area, porosity or density.

A wide range of techniques, traditional and advanced tools, are being applied to characterize the oxidation extent on UO_2 (using both, in-situ and ex-situ approaches). For in-situ oxidation experiments, techniques employed are thermogravimetric analysis (TGA) and Raman spectroscopy using a Linkam stage, and for ex-situ experiments are performed by means of isothermal reactors.

Samples are usually characterized by complementary techniques such as specific surface area, microhardness, Scanning Electron Microscopy with Energy Dispersive X-ray Spectroscopy (SEM-EDX), X-Ray Diffraction (XRD) and Raman spectroscopy, particle size distribution by laser diffraction or Inductively Coupled Plasma-Mass Spectrometry (ICP-MS), among others.

2.1.2. CIEMAT-UNSR

The main objective of the research carried out in dry storage by the CIEMAT-UNSR is the development of methodologies for assessing the dry stored fuel integrity. To do so, credited predictive tools are being extended and validated for an accurate thermo-mechanical characterization of fuel rods. The research scope encompasses conventional fuel from Light Water Reactors (LWR) (UO_2 pellets with Zirconium alloy cladding), from in-reactor end-of-life to transport and handling operations required by its final geological disposal repository, both under nominal and off-nominal conditions.

Therefore, the main pillar of this research is a sound fuel performance code valid for the anticipated conditions. The FRAPCON code, developed by Pacific Northwest National Laboratory (PNNL) and extensively credited, was used (Geelhood et al., 2015) and has been extended to what is called FRAPCON-xt (Feria et al., 2015). The advantage of this strategy lies in its capacity to transmit straight the rod end-of-life conditions as initial storage conditions.

The work carried out on FRAPCON-xt involved the derivation of new modelling associated with postulated cladding degradation mechanisms such as creep (viscoplastic deformation), UO_2 oxidation in non-detected defective rods (related to pellet swelling and mechanical interaction with the cladding) and radial reorientation of hydrides formed from in-reactor hydrogen pickup (related to alloy embrittlement). The two former were modelled through semi-empirical correlations (Feria and Herranz, 2011; Herranz and Feria, 2010, 2009).

In case of the hydride radial reorientation, a complex phenomenon considered one of the main threats for cladding integrity in dry storage, an in-house semi-mechanistic model has been developed: HYDCLAD (F. Feria et al., 2020; Feria and Herranz, 2023, 2018). It simulates the hydrogen migration and precipitation as a basis for the estimation of the radial hydrides formed under dry storage conditions. HYDCLAD is externally coupled with FRAPCON-xt.

Given the relevance of the thermal conditions on the cladding degrading mechanisms, a 3D thermo-fluid dynamic (TFD) model with Ansys Fluent (Herranz et al., 2015) has been conducted for several casks. To avoid the excessive overloading resulting from the coupling of a 3D TFD and FRAPCON-xt, an engineering method to estimate dry stored fuel temperature has been derived (Penalva et al., 2021).

Additionally, the computational capabilities have been extended by coupling FRAPCON-xt and DAKOTA, a statistical tool enabling multiple code realizations and later sensitivity analysis of the results. This allowed the application of bringing thermo-mechanical characterization in the frame of the BEPU (Best Estimate Plus Uncertainty) approach (Feria and Herranz, 2017).

Finally, a methodology for the characterization of the cladding ductility during dry storage has been developed. It is based on the prediction of the radial reorientation of hydrides with HYDCLAD (fed with FRAPCON-xt) and a conservative criterion of zero radial hydrides for ductile cladding (there is not a settled embrittlement criterion related to the radial hydrides content so far). For this purpose, three levels of predictions are established depending on the thermal boundary conditions applied during storage: conservative (maximum cladding temperature of 400ºC with isothermal axial profile), limiting (maximum cladding temperature of 400ºC and axial temperature profile) and realistic (maximum cladding temperature of 350ºC and axial temperature profile). Given the impact of the irradiation period on the characterization aimed, a complementary in-house methodology is available to take into account the worst power history (in terms of the cladding degradation targeted) among a wide variety of realistic irradiations statistically obtained through the FRAPCON-xt and DAKOTA coupling (Feria et al., 2020).

2.2. Universities

2.2.1. UPM-ETSI Caminos

The main goal of UPM-ETSI Caminos is to evaluate the effect of very long dry storage and transportation on fuel cladding integrity, to prevent the release of radioactive species. The secondary objectives are to investigate the effect of hydride embrittlement and hydride cracking on cladding mechanical behaviour in accident conditions.

Then, the research methodology involves the following steps:

- Hydrogen charging: a cathodic charging technique (KOH aqueous solution at room temperature) is employed to introduce controlled amounts of hydrogen in cladding samples (10 mm height rings) of unirradiated ZIRLO®. The cladding sample is used as the cathode of the electrochemical reaction, while a platinum wire coiled around the sample is the anode.
- Hydride precipitation: a thermal treatment is employed to precipitate δ-zirconium hydrides. As a result, homogeneously distributed hydrides along the hoop direction of cladding are obtained.
- Radial hydride treatment (RHT): samples from the previous step (hoop hydrides) are subjected to a thermo-mechanical treatment to reorient the hydrides along the radial direction of cladding.
- Mechanical testing: ring compression tests (RCT, representative of a "pinch-loading" accident) are performed at different temperatures (20, 135 and 300ºC) and loading rates, representative of storage and transportation conditions.
- Metallographic and fractographic analysis: post-test examinations are performed to study the effect of hydrogen content, hydride morphology and temperature on the fracture micro-mechanisms in RCT. Metallographic cross-sections are prepared and observed (by optical microscopy) to ascertain the effectiveness of the RHT on the resulting hydride morphology. Failure modes and micro-mechanisms are also investigated by optical microscopy of metallographic cross-sections and by scanning electron microscopy on the cladding fracture surfaces.

2.2.2. UPM-ETSII

Due to the current difficulties for commissioning a CTS or a Geological Deep Repository in countries like Spain, the nuclear industry started relicensing one decade ago the temporary storages with dry cask systems to contain the spent nuclear fuel for a longer period. CFD (Computational Fluid Dynamic) codes have been the major players in the effort to simulate this type of systems.

In that context, UPM-ETSII and CNAT (Centrales Nucleares Almaraz Trillo), started to collaborate in 2014 in a research line to simulate spent fuel dry cask systems with CFD codes. The scope was to be able to simulate a cask to the level of detail of fuel rods (no porous media approach in the fuel region) and simulating at the same time the surrounding environment (atmosphere and floor) without imposing thermal boundary conditions, as it was the state of the art at that time.

The first dry cask model simulated in the collaboration UPM-CNAT project was the DPT cask design, simulated with the Ansys Fluent code. One of the first hurdles when simulating a dry cask is the complexity of the fuel bundles and the cask itself, especially difficult in the case of the DPT cask, with its multiple disks to increase the heat transfer surrounding the fuel bundles. To alleviate this problem, the problem was divided into a two-step process: an Interior model and an Exterior model (Benavides, 2021). In the first step, the cask is simulated from the inner lining of the cask to the atmosphere (Exterior model), with the heat flux distribution for each surface approximated on the first iteration. Two results are obtained from this simulation: the thermal impact the cask has on the environment, useful to ensure safety parameters when designing temporal storage facilities, and the boundary conditions to use for the second step. On the second step the interior of the cask is simulated with as much detail as it is possible (Interior model), obtaining mainly the Peak Cladding Temperature (PCT), and other parameters that may be of interest such as the thermohydraulics of the system. This two-step process is repeated until convergence is reached on both models.

Afterwards, the previously introduced Interior and Exterior models were adapted to be used with the Ansys CFX 2019 R3 code, in order to assess the ENUN-32P, a metallic nuclear spent dry cask which houses 32 PWR fuel elements (Hernandez-Avellaneda et al., 2022a). This cask has been built with the available public information from different sources (Consejo de Seguridad Nuclear, 2018a, 2018b; Grey et al., 2018; Hernández et al., 2017; Transnuclear Inc., 2002, 2000).

The models were coupled following similar steps to those explained previously for the DPT cask until achieving good convergence, which was measured using heat flux distributions on the main surfaces of the cask. The Interior model, a far more complex model and with a greater mesh count (6.52 million cells, in contrast with 1.77 million cells for the Exterior model), required several simplifications in the construction of the geometry to make it viable. Some simplifications were related to model complex solid bodies with different layers of materials as a single body, featuring a material with equivalent thermal properties. The equivalent materials were modelled using both isotropic and anisotropic properties.

In the next step of the methodology, the Interior and Exterior models were combined into a single model, the so-called Integral model (Hernandez-Avellaneda et al., 2022b). This new step in the methodology allowed to observe the Heat Transfer Coefficient (HTC) in the exterior surfaces of the cask, while also taking into account the effects of the complex geometry of the interior of the cask. The HTC was found to be heterogeneous, both axially and radially, with a greater contrast in function of the height.

In the last step of the ENUN-32P research, the Integral model was tailored and used to assess the importance of modelling the fuel assembly flow nozzles (Hernandez-Avellaneda et al., 2023), components which are rarely modelled explicitly in CFD models of the state of the art in nuclear spent fuel casks. Even though the nozzles were simplified, mainly reducing the number of small surfaces, especially the curved surfaces in the small holes in the nozzles; the computational effort to include them in the model was significant. The mesh count of the Integral model increased from 10.56 to 15.81 million cells; an increase close to 50%.

The Interior/Exterior two-step methodology was also applied to the TN24P dry cask in a collaborative research project with Enusa, comparing the simulation results with the experimental measurements of the cask with good agreement (Benavides et al., 2019a). Additionally, to help the validation of the methodology, experimental results from the Extended Storage Collaboration Programme (ESCP), coordinated by EPRI, were also used (Benavides et al., 2020). Some studies were done for several casks configuration in a closed repository (Benavides et al., 2019b). Finally, the TN24P CFD model was one-way coupled with ABAQUS to estimate the thermal displacements and stresses (Benavides et al., 2021).

2.3. Industry

2.3.1. Enusa

Enusa, as designer, manufacturer, and supplier of nuclear fuel, has provided different services in the SNF management area for its customers with PWR and BWR technologies in the past 20 years. The main activities related with SNF management performed by Enusa are focus on the SNF characterization and classification processes to its dry storage. This characterization and classification of SNF is a critical and complex activity in the process of waste nuclear management. The Spanish current legislation just provides definitions for classifying the SNF to assure fuel-specific and system (dry cask)-related function. There is no clear process or methodology defined by the industry or regulators to do that, this is why Enusa, in collaboration with Spanish NPP (PWR and BWR type), ENRESA and other technological partners, had defined its own process to characterize and classify SNF for its storage and transportation in dry cask (Viñas-Peña, 2019; Viñas-Peña and García-Infanta, 2014).

This methodology developed for PWR and BWR technologies is based on the SNF nuclear safety functions fulfillment analysis regarding to cladding and fuel assembly structural integrity. It identifies degradation mechanisms such as hydriding, Stress Corrosion Cracking (SCC) that could compromise the SNF integrity during dry storage and set acceptance criteria to classify fuel assembly affected by these degradation mechanisms. Over last 15 years, Enusa has classified more than 1000 SNF assemblies for its storage and transportation according to its methodology with the approval of CSN and ENRESA.

Fulfilling the fuel nuclear safety functions during dry storage of SNF requires an assessment of the condition for each fuel assembly along with technical solutions that reduce the SNF population classified initially as damaged fuel. Different kind of data are required such as materials and dimensional related data, irradiation history (data related to power history and chemistry of the coolant during the cycle among others) and SNF characterization data (e.g., inspection results, oxide thickness, hoop stress…) for each fuel assembly. These data have been gathered by Enusa for more than 40 years and provided to the Spanish NPP for its waste management related activities according to ENRESA’s requirements. This information is used to classify each fuel assembly as damaged or undamaged.

However, in some cases the lack of data on the behavior of SNF under certain dry storage operational conditions, conservatively forces to classify these SNF as damaged. Enusa has promoted different R&D projects in order to get that information and reduce the SNF population classified initially as damaged fuel.

The main R&D lines developed by Enusa could be grouped in the following programs:

- Pellet and cladding: the final aim of these projects is to provide the data necessary to justify the performance during dry storage of SNF with defects or under demanding operational conditions resulting in excessive corrosion, hydriding, bow, etc. These programs include an extensive experimental in-pile and out-of-pile characterization and testing of irradiated and fresh nuclear fuels, respectively.
- SNF structural assembly integrity: the objective is the development of technical solutions that assure the retrievability of fuel assembly with defects.

- Modelling: mechanical and thermal calculations of SNF under dry storage conditions are some examples of modelling programs carried out by Enusa in the past years.
- Data management: this program is focused on the tool development for the management of all the data required to classify SNF.

Enusa also gives support on the design and licensing of dry storage installations, as the R&D projects developed in the framework of collaboration with ENRESA to design of the CTS facility. Moreover, Enusa carries out different activities for the design of dry cask containers for different manufacturers such as shielding, criticality and structural assessments among others.

2.3.2. ENSA

The goal of this research is the optimization of spent fuel cask regionalization. Spent fuel cask regionalization is a loading strategy in which usually the hottest spent fuel assemblies (SFA) are placed in the inner region of a cask basket, while the coldest ones are placed in the outer region. This is due to the heterogeneous characteristics of spent fuel that may be loaded in the same cask (burnup, enrichments, cooling times, bundle designs, operating history, etc., which in turn imply differences in neutron and gamma yield) and the fact that SFAs shield one another when stored in spent fuel casks. Then, a legitimate concern is if this regionalization may be optimized and how.

This work aims to define an analysis strategy by which the cask regionalization may be optimized. This was one of the strategies followed by ENSA for the ENUN casks. These are a series of bare fuel metal casks designed and manufactured by ENSA for the storage and transport of Light Water Reactor (LWR) SFA. The cask comprises a low-alloy carbon steel body, enclosed by a neutron shielding material and covered with an inner and an outer lid. The cask cavity contains an egg-crate-type basket made of steel and neutron absorber plates, to accommodate 24, 32 or 52 SFA, depending on the specific ENUN design.

The individual contribution of each SFA loaded in a cask to dose rates outside the cask was analysed, depending on each SFA relative position within the cask basket. Monte Carlo calculations of gamma and neutron particles have allowed ENSA to create cask basket maps that show which positions of the cask basket affect the most (and which ones the least) to dose rates at different representative locations in its proximity. Dose rates have been calculated at representative locations around, above and below the cask as shown in Fig 1, which are:

- A ring surrounding the lateral side of the cask at mid-height, whose radius is 2 m greater than the outermost shell of the cask.
- A point detector located 2 meters above the centre of the cask outer lid.
- A point detector located 2 meters below the centre of the cask bottom.

Individual contributions of each fuel assembly to total dose rates at the specified locations have been separately evaluated for cask fully loaded, i.e., only one SFA source has been considered at a time. This procedure allows self-shielding effects to be considered for each of the calculated terms separately (gamma, with a constant intensity for all basket positions and the $^{137m}Ba/^{137}Cs$ spectra; and neutrons, with a constant intensity for all basket positions and the ^{244}Cm spectrum). $^{137m}Ba/^{137}Cs$ and ^{244}Cm spectra are typical of LWR SFA. The source term parameters (e.g.: in the case study shown in this paper: 60 GWd/MTU burnup, initial ^{235}U enrichment of 4.62% wt. and 7.9 years of cooling time) are chosen as representative of SFAs in Spain.

Results have been used to identify and evaluate SFA self-shielding effects and radiation streaming significance when the cask is loaded. These results have been used to develop smart cask-loading strategies which minimize dose rates in the vicinity of a loaded cask, but may be also used for cask design improvements if necessary.

Monte Carlo calculations have been performed with SCALE (Standardized Computer Analyses for Licensing Evaluation) neutron transport code, developed by ORNL (Oak Ridge National Laboratory), version 6.1.2, using its MAVRIC sequence (Monaco with Automated Variance Reduction using Importance Calculations), with ENDF/B-VII-based libraries of SCALE: v7-27n19g (27 neutron energy groups and 19 gamma energy groups) for the importance function calculations for the variance reduction functionality, and v7-200n47g (200 neutron energy groups and 47 gamma energy groups) for the actual Monte Carlo task.

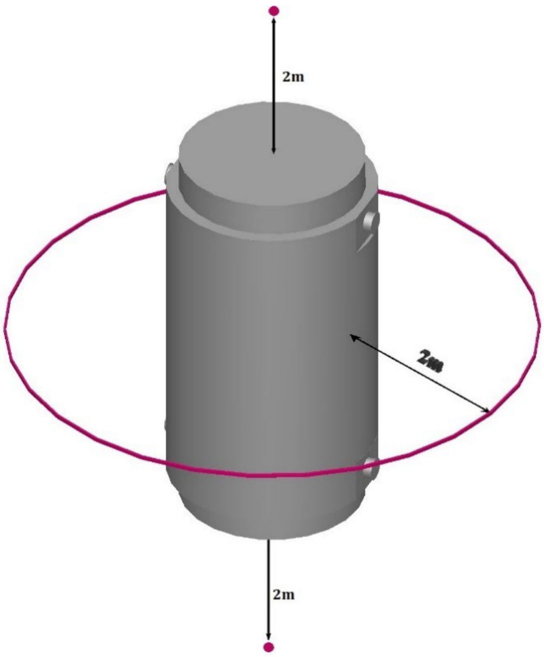


Fig 1. Location of calculation points and ring where dose rates were calculated around the cask.

2.3.3. IDOM

IDOM is playing a crucial role in the implementation of an acceptance system for the spent fuel of the Spanish inventory. The acceptance criteria and the acceptance methodology are being established through a collaborative effort between the NPPs and ENRESA, taking into account the requirements of regulatory authorities, international standards, and best practices. The overarching objective of this collaboration is to ensure that any activity related to spent fuel characterization and classification are carried out safely.

In order to adhere to the acceptance criteria, the availability of the following information for each spent fuel assembly is imperative:

- Pre-irradiation data, which should be available in the so called GECYRE[1] database. This includes the mechanical and nuclear design data of the fuel

[1] ENRESA's nuclear fuel database fed by Spanish NPPs.

element and its components, such as geometries, materials, enrichments, distribution of gadolinium rods, and other relevant parameters. The reference documentation justifying the values of these data must also be accessible.
- Irradiation data, also available in the GECYRE database, which includes records related to the history of use of the fuel assembly in the reactor, such as power and burnup history, date of discharge, and related details.
- Characterization results, additional data that are required for documentary acceptance of the fuel assembly but are not available in the GECYRE database. Examples of such data include oxide thickness values of fuel rods composed of a specific alloy. In these cases, the data are agreed upon between the NPPs and ENRESA, and the reports containing these data, which are prepared or verified by the producers, are also be made available to ENRESA.

By possessing all the aforementioned information, it is possible to check compliance with the acceptance criteria, including the classification of the fuel into undamaged and damaged, as specified by the ISG-1, rev. 2 (USNRC, 2007).

3. Key outcomes

3.1. Research entities

3.1.1. CIEMAT-HLWU

Among the methods and tools developed at the CIEMAT-HLWU, in-situ protocols are of particular interest, because they allow studying the fuel oxidation at specific conditions simulating the state of irradiated fuel during dry storage. In particular, one of the main outcomes of CIEMAT-HLWU's research is the ability to monitor the oxidation reaction by Raman spectroscopy, using the aforementioned Linkam stage (Raman-Linkam experiments), a device that allows selecting temperature and gaseous atmosphere at which the reaction will take place. The focus of this research is on the time needed for the formation of U_3O_8 (the so-called induction time) for each of the conditions, therefore, if this reaction takes place, it could be foreseen the timeframe available in the decision-making process in the event of certain incidental or accidental circumstances to maintain containment. For this reason, the oxidation of a fresh UO_2 (i.e. unirradiated), consisting of powder coming from milling sintered pellets, and being a realistically conservative analogue of SNF (Milena-Pérez et al., 2023b), has been studied. Some representative results of this study are presented in Fig 2, where it can be concluded that lowering the oxygen concentration in the atmosphere can delay the formation of U_3O_8; nevertheless temperature has a major impact on the UO_2 oxidation.

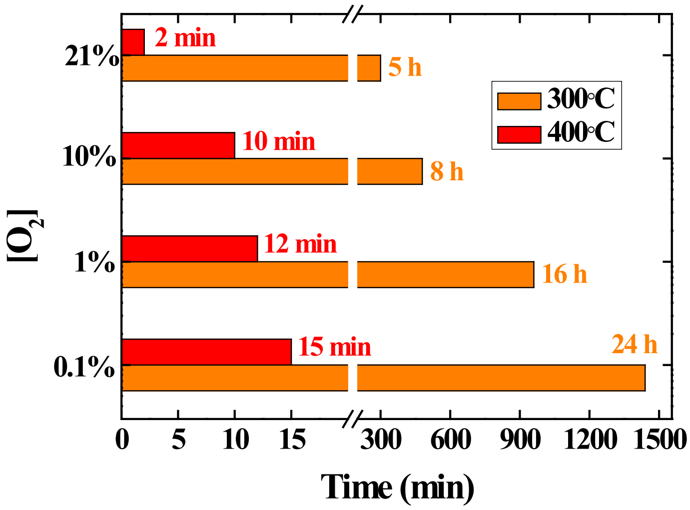


Fig 2. Induction time for U_3O_8, measured by Linkam-Raman, as a function of oxygen concentration and temperature.

Besides Raman-Linkam, another in-situ method widely used to study the fuel oxidation is TGA. In the CIEMAT-HLWU, this technique is used by means of two different approximations:

- Isothermal oxidation experiments. In order to validate the new Raman-Linkam method, the fuel oxidation with the same UO_2 surrogate and defining the same experimental conditions is studied. In this case, the oxidation of the "bulk" of the sample is obtained, defined as the average weight change.
- Non-isothermal thermoanalytical experiments. In this case, a thermal treatment consisting of a linear heating from room temperature up to 700ºC under flowing air conditions is performed. The goals here are, by one side, to study the oxidation kinetics of the formation of U_3O_8 in different materials, ensuring that this phase is formed at this high temperature; by another side, to see the number of oxidation steps and onset temperatures (beginning point of a transition). By applying this, CIEMAT-HLWU has been able to analyse as-prepared sintered pellets of different fuel analogues, which are of interest for the current scientific community. These surrogates include standard UO_2; new Accident-Tolerant Fuels (ATFs) (Milena-Pérez et al., 2021), fabricated by adding Cr to the UO_2 matrix prior to the sintering step; and ZrO_2-doped UO_2 (Rodríguez-Villagra et al., 2023), with the goal to simulate the Pellet-Cladding Interaction (PCI) that takes place in nuclear fuel irradiated at high burnup. These doped materials represent novel approaches to the oxidation behaviour of new fuel designs. Regarding ATFs, there is a slight delay with compared to non-doped UO_2, probably because the low concentration of Cr needed. On the other hand, differences in oxidation behaviour of the studied $(U,Zr)O_2$ samples as a function of Zr content are observed. In general, a delay in the UO_2 oxidation is found when increasing the Zr doping concentration (Fig 3).

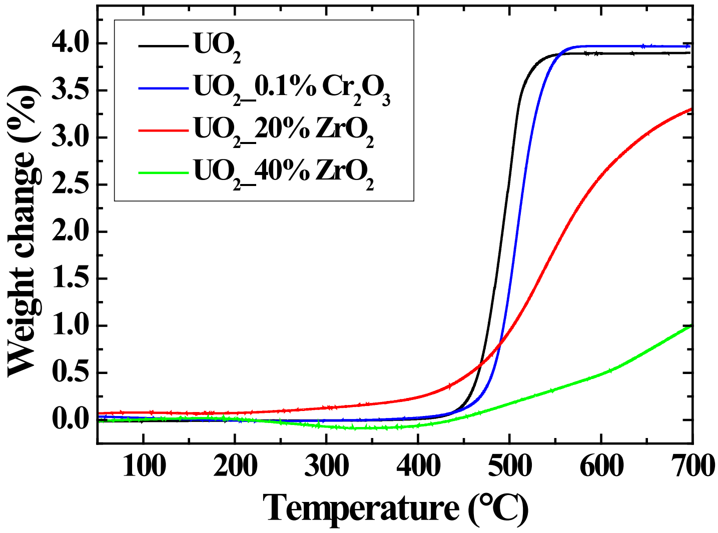


Fig 3. Air-heating (up to 700ºC) of different materials fabricated in the CIEMAT-HLWU as sintered pellets: standard UO_2, ATFs (0.1%Cr_2O_3-UO_2), PCI surrogate (20 and 40%ZrO_2-UO_2 fragments).

3.1.2. CIEMAT-UNSR

The key outcomes from the research performed by the CIEMAT-UNSR can be split into three main blocks: verification of the analytical thermo-mechanical capabilities gained, validation of the related modelling against data and application of the methodologies developed from the analytical tools. The former has been focused on consistency checks of FRAPCON-xt and HYDCLAD based on the simulations of postulated scenarios, which has allowed verifying the qualitative response of the tools (Feria et al., 2015; Feria and Herranz, 2011; Herranz and Feria, 2010).

Concerning validation, the lack of data available from representative dry stored irradiated rods prevents from performing a sound integral assessment of FRAPCON-xt and HYDCLAD under dry storage condition. However, post-irradiation examinations (PIEs) open data from commercial fuel rods operation have been found to carry out the assessment of the initial characterization previous to storage. This study has been focused on two relevant Figures of Merit (FOMs) like the rod internal pressure, RIP (direct impact on cladding stress as a driver of degrading mechanisms such as creep or hydride radial reorientation) and the in-clad hydrogen distribution (impact on ductility related to hydride rim formed in high burnup irradiations and the potential hydride radial reorientation in dry storage).

In case of RIP, FRAPCON-xt is found to be accurate for burnups lower than 60 GWd/tU, with maximum deviations of around 10%. For higher burnups, though, deviations grow progressively up to nearly 30% at 69 GWd/tU, being not conservative predictions (underestimation). An analysis of the results pointed to FGR as the main source of discrepancy, so a recalibration of an alternative FGR model (more detailed for high burnup) has been done. Fig 4a shows an enhancement of the RIP prediction accuracy with respect to the modelling by default (relative error reduction from 33 % to 12% in the highest burnup fuel rod). Regarding the in-clad hydrogen distribution estimation with HYDCLAD, two modelling options have been assessed: previous (more empirical) and new (more phenomenological). Comparison against EOL PIE data from a 69 GWd/tU fuel rod indicates that the newest HYDCLAD version is more accurate in the prediction of the hydride rim (Fig 4b).

Regarding the methodologies adopted, the application of the BEPU methodology to postulated dry storage scenarios has shown the importance of including uncertainties in associated safety studies (Feria and Herranz, 2017). In case of the methodology for the characterization of the cladding ductility during dry storage, it has been applied to scenarios of irradiated PWR rods (up to around 60 GWd/tU) dry stored during 20 years in a metallic cask. The results obtained allow concluding that under realistic thermal conditions, the cladding of the analysed rods would retain ductility since no radial hydrides have been predicted at the end of the storage period (Feria et al., 2020b).

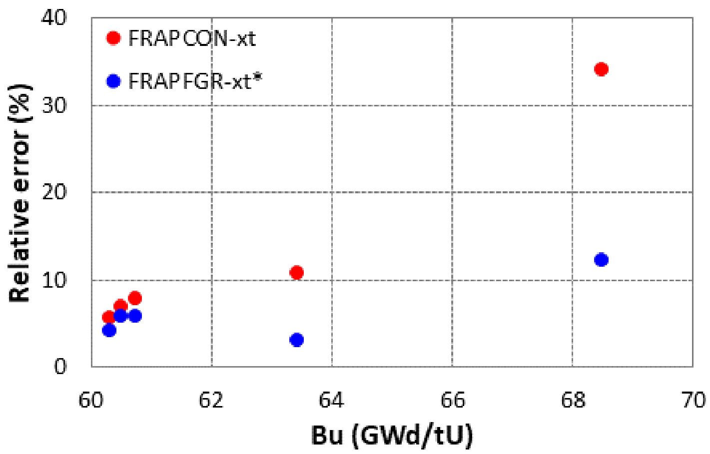


Fig 4a. RIP relative errors (absolute value) above 60 GWd/tU. Predictions with FRAPCON-xt (before calibration) and FRAPCON-xt* (after calibration).

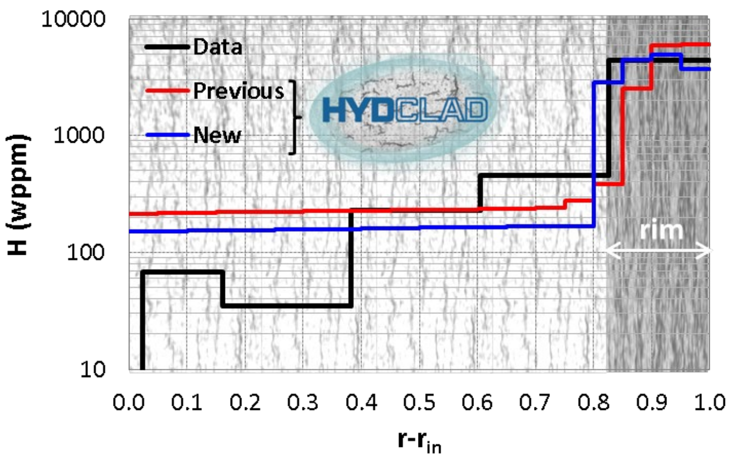


Fig 4b. Distribution of the hydrogen content (in logarithmic scale) across the cladding thickness (normalization of the radius, r, minus the inner radius, rin).

3.2. Universities

3.2.1. UPM-ETSI Caminos

Among the research activities carried out by the UPM-ETSI Caminos, below it is shown key outcomes achieved.

<u>Effect of the reorientation treatment on the hydride morphology</u>

Fig 5 corresponds to the cross-section of a cladding sample after the hydride precipitation treatment (Fig 5a) and after the RHT (Fig 5b). In this case the hydrogen content is approximately 75 wppm and the reorientation stress is 140 MPa. The hydride morphology after the hydrogen charging and precipitation treatment (before the RHT) consists of hydride plates oriented in the hoop direction (typical of cold-work stress-relieved cladding material) and mostly located at the mid-thickness of cladding. After the RHT, a considerable fraction of hydrides has precipitated in the radial direction, and strings of connected hydrides cover the cladding thickness, in a very homogeneous fashion for all generatrices. The radial hydride continuity factor (RHCF, (Billone et al., 2013)) is approximately 80-90% of the wall thickness.

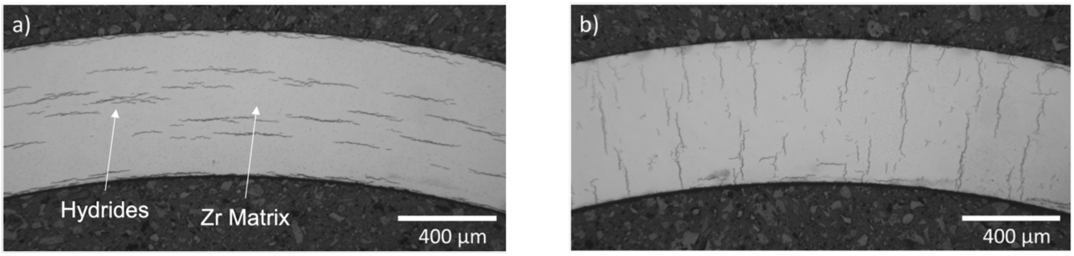


Fig 5. Micrograph of a ZIRLO® cladding sample: a) after the precipitation treatment and before the reorientation treatment, and b) after the RHT. Hydrogen content was approximately ≈75 wppm (measured by the hot extraction method with a LECO® ONH-836 analyzer). The reorientation stress was 140 MPa in this case.

Ring compression test results

Fig 6 shows the load-displacement plots for 3 samples subjected to the precipitation treatment (PT) and other 3 after the RHT (constant reorientation stress of 140 MPa), at 20, 135 and 300 ºC, respectively. When the hydrides are oriented along the hoop direction (PT samples) load increases with displacement monotonically and the maximum load is found at displacements larger than 5 mm, the maximum value decreasing with temperature (1 kN at 20ºC and 0.7 kN at 300ºC). On the contrary, the samples with radial hydrides (at 20 y 135°C) fail catastrophically for loads smaller than 0.25 kN. Sharp load drops (associated to the appearance of radial cracks) are observed for displacements around 0.2 mm. At 300 °C, the load-displacement plot is very similar for all samples, irrespective of the hydride morphology. The deleterious effect of the radial hydrides seems to disappear at this temperature.

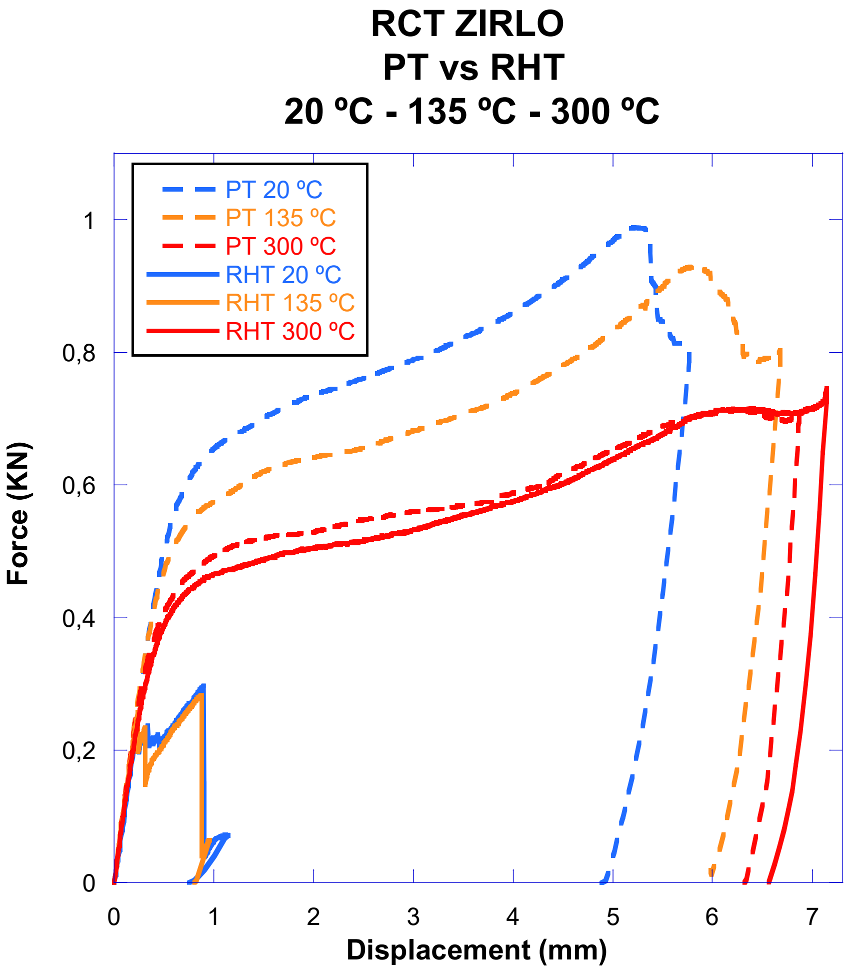


Fig 6. Load-displacement plots for samples subjected to the precipitation treatment (PT) and after the radial hydride treatment (RHT with a constant reorientation stress of 140 MPa), at 20, 135 and 300ºC, respectively.

<u>Failure micro-mechanisms in the presence of radial hydrides</u>

Fig 7 shows micrographs corresponding to the cross-section after RCT (9 o'clock position) at 20 °C, 135 °C and 300 °C. Cracks covering the whole wall thickness and growing along neighbouring radial hydrides are found at 20 y 135 °C (Fig 7a and Fig 7b). Secondary cracks close to the main crack are observed (marked with arrows in Fig 7a). On the contrary, no cracks are found in the samples with radial hydrides tested at 300 °C (Fig 7c) and most hydrides have disappeared. As the TSSD at 300 °C is approximately 50 wppm (Kim et al., 2020) and the hydrogen content is 75 wppm in this case, it is possible that part of the hydrides are re-dissolved during the test.

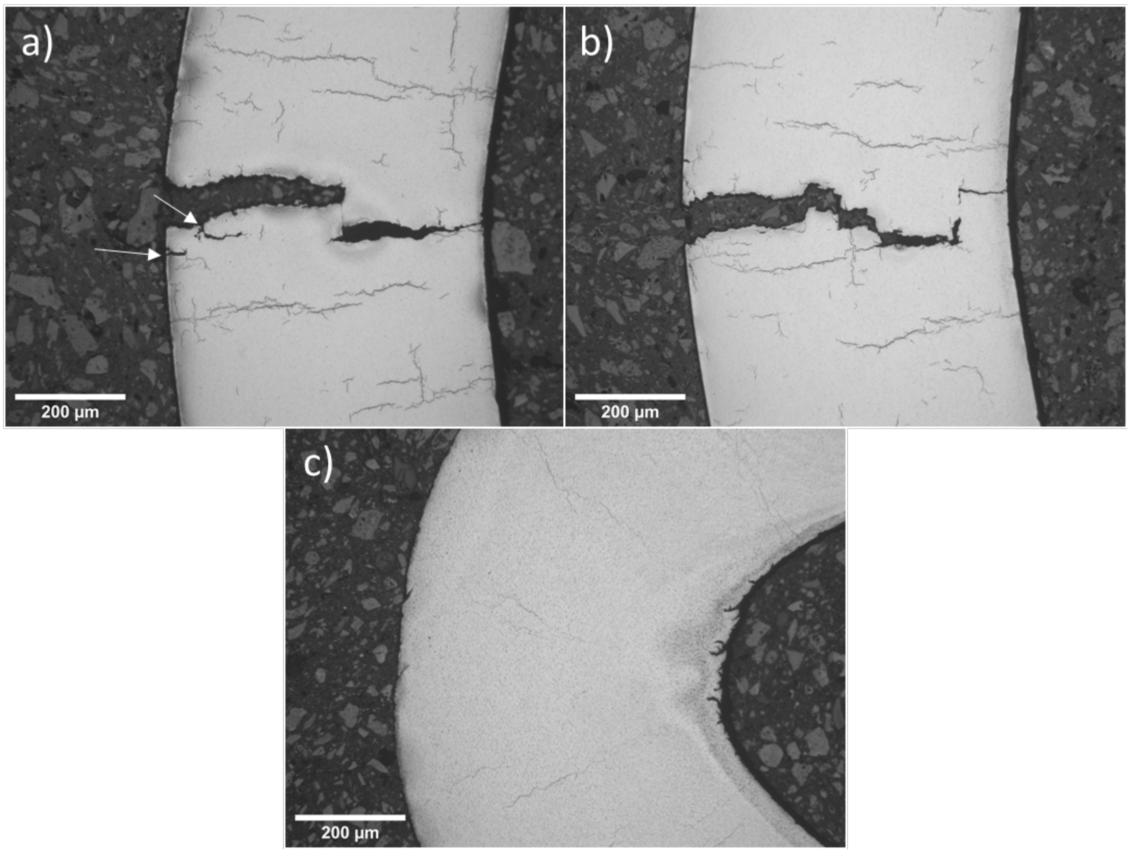


Fig 7. Micrographs corresponding to samples subjected to the radial hydride treatment after RCT at: a) 20 °C, b) 135 °C and c) 300 °C. 9 o'clock position.

3.2.2. UPM-ETSII

From the UPM-ETSII´s experience simulating dry cask with CFD codes some outcomes can be underline:

- The Interior/Exterior two-step model is an adequate methodology to save complexity in the modelling of a dry cask with its surroundings (Benavides et al., 2019a; Hernandez-Avellaneda et al., 2022a).
- The Integral model, see the geometry in Fig 8 (Hernandez-Avellaneda et al., 2022b), although requires a greater mesh count and higher RAM memory cost, showed a superior performance related to CPU time, as the coupling process between the Interior and Exterior models could be skipped, only simulating the Integral model a single time, instead of couples of Interior/Exterior models.
- The use of anisotropic properties in the materials of the cask, which is believed to be closer to the real situation, gives rise to lower overall temperatures, with a PCT up to 16°C lower under the boundary conditions used. Although simplifications cannot be avoided in these complex models, materials should be carefully simplified to avoid excessive conservatism.
- Regarding the studies of the Exterior model, the sensitivities of the modelling near the wall can be highlighted. Due to the high Rayleigh number, the air around the cask was deemed to be turbulent, which required the use of a turbulence model (SST model was chosen (Menter, 1994, 1993)). It was found out that a poor modelling of the near wall could introduce significant differences in the model results. Refining the mesh near the wall, to obtain an y+ under 1, negated those differences, although an y+ under 5 also produced similar results.

- The Integral model was also used to study the relevance of the thermal radiation in the cooling of a nuclear spent fuel cask (Hernandez-Avellaneda et al., 2022b). Under the boundary conditions used, the heat transferred from the cask to environment via radiation represented roughly a 68 % of the total evacuated heat. The thermal radiation inside the cask was also evaluated, and it was observed that over 23 % of the energy emitted from the claddings was neatly evacuated though radiation (this number is calculated by summing both the radiation out and into the claddings, which have opposite signs, meaning that the heat transferred though radiation is even higher). The radiation neatly absorbed by the basket represents around 27 % (hottest fuel assemblies) and 19 % (coldest fuel assemblies) of the total heat transfer. These results proceed from an Integral model with updated emissivity values of the main solid bodies inside the cask. Even though the updated emissivity's did not differ greatly from the original values, the PCT of the model increased around 7°C due to the reduction of the emissivity's. As the thermal radiation plays an important role in the cooling of a nuclear spent fuel cask, attention should be paid to the emissivity values.
- The comparison of the Integral nozzle model with the Integral base model showed only subtle changes (Hernandez-Avellaneda et al., 2023), with probably the biggest difference of including the nozzles being a reduction in the strength of the natural circulation of the helium inside the cask (around 10.8 %, see the streamlines in Fig 9.). The reduction of mass flow should cause a disruption in the temperatures, but the differences in temperatures were mainly irrelevant, having the model with nozzles a PCT increase of 1.5 ◦C. The nozzles also impacted the interior heat flux distributions, due to additional solid-solid interfaces, but it was observed that this effect was limited and compensated by the reduction of thermal radiation, as the modification of shape factors, made the thermal radiation less effective in certain areas of the cask.

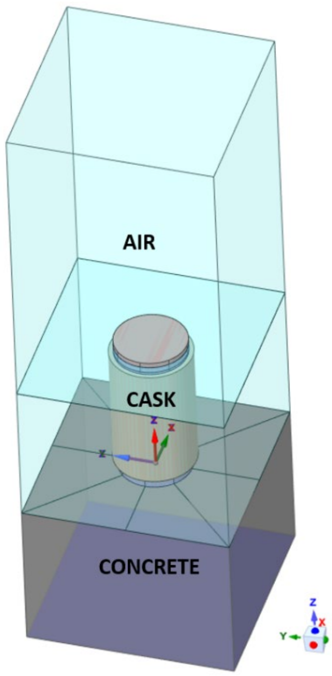


Fig 8. Integral model geometry

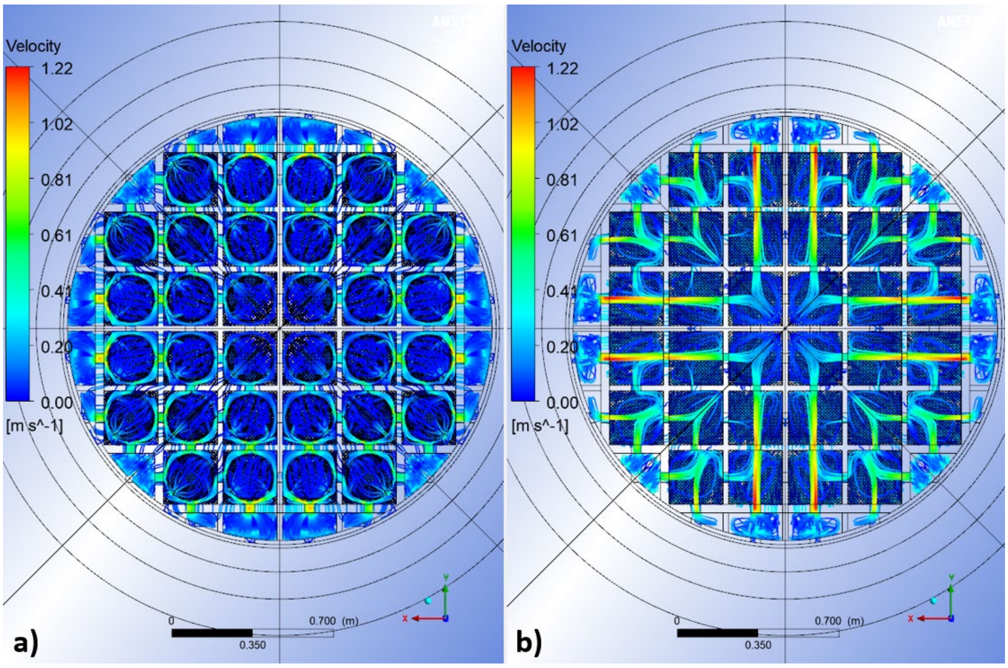


Fig 9. Helium streamlines. Inverted bottom view. a) Integral Nozzle model b) Integral model.

3.3. Industry

3.3.1. Enusa

<u>Pellet and Cladding related programs</u>

Enusa has promoted Post-Irradiation Examination (PIE) Programs to evaluate the performance of cladding materials (as the example in Fig 10 which shows the IFM grid's impact on cladding corrosion) and the integrity of high burnup fuel (Billone et al., 2017). Experimental characterizations in hot cell were aimed to both irradiated claddings and pellets, including metallography, hot vacuum extraction, fission gas release analysis and dimensional measurements, among others.

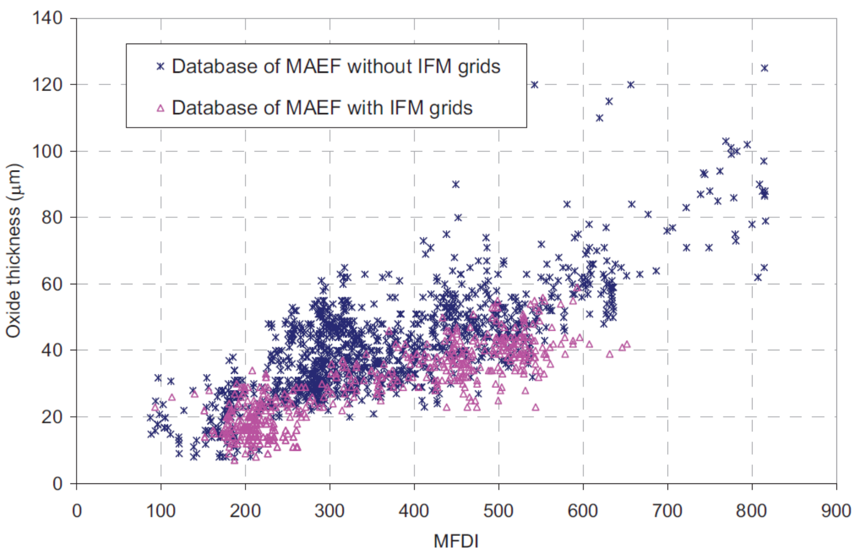


Fig 10. Corrosion database from Enusa's surveillance program (Muñoz-Reja et al., 2010).

The spalling of oxide layers during fuel operation creates cold spots which lead to form hydride blisters at the cladding outer surface and embrittle it locally. In this regard, Enusa developed a surveillance program in collaboration with 17x17 PWR Spanish NPPs to analyse the cladding zirconium alloys which may be affected by this phenomenon. Oxide thickness measurements and visual inspections have been performed in fuel assemblies with different burnup and cladding materials.

The effect of a hydride blister on the mechanical behaviour of spent fuel under conditions representative for drying, storage and transportation became an important objective in the framework of Enusa's activities related to the SNF classification. This R&D Enusa´s project has been developed in collaboration with 17x17 PWR Spanish NPP, ENRESA and CSN (Consejo de Seguridad Nuclear) and it is focused on the characterization of irradiated and non-irradiated material with hydride blister through different non-destructive test, metallographic examinations, and mechanical testing carried out by different laboratories including hot cells. The experimental data obtained in this program has been used to validate the behaviour during dry storage of spalled fuel.

SNF structural assembly integrity programs

Fracture of structural components of the fuel assembly due to SCC phenomenon was an issue for PWR fuel in the past. Although these problems have disappeared thanks to fuel assembly design improvements, from the point of view of dry storage it is a challenge to deal with because the presence of cracks due to SCC on the top nozzle and skeleton joints implies that the SNF could not be handled as usual and therefore it is not possible to transfer from pool to dry storage. Main outcomes from Enusa in this regard come from the following R&D projects:

- PIE program oriented to surveillance the performance of structural components (e.g., grids, guide tubes…) under high burnup conditions (García-Infanta et al., 2013).
- Experimental tests aimed to verify the behavior of the joints between top nozzle and skeleton of SNF under different dry storage conditions such as temperatures or material conditions.
- Development of a device named Espiga that allows the handling of SNF assembly along all back-end cycle (pool storage, dry cask containers storage and further steps). It supports the weight of the fuel assembly complying with the design criteria. The device consists of two rods (inner and outer bars) and two nuts: the rods are introduced through the instrument tube and fixed to the top and the bottom nozzles (Fig 11). The design has required some challenging issues as deal with the installation of the device in the rack location without moving the fuel assembly, assure the compatibility with core components after installation and allow its uninstallation, if it would be necessary. Since its creation, more than 400 Espiga devices have been successfully installed on fuel assemblies and, after it, the fuel assemblies have been reclassified as "Undamaged" (Canencia et al., 2016).

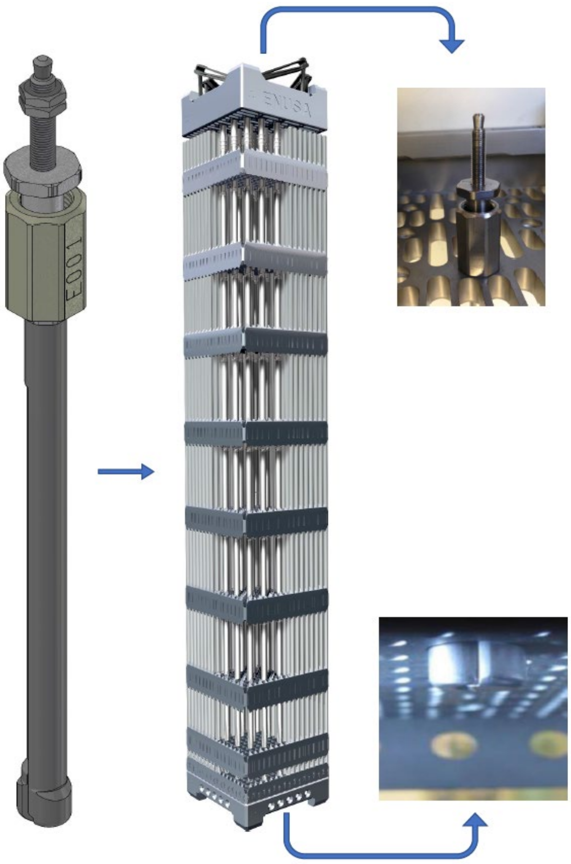
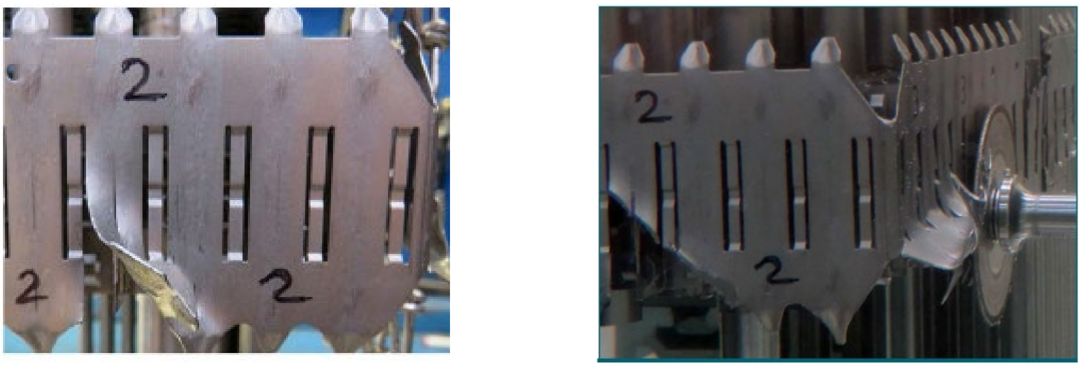


Fig 11. Espiga device: uninstalled (left) and installed into the fuel assembly (right)

Other issue that Enusa has solved to guarantee the undamaged condition of fuel assemblies has been the conditioning of distorted components from the fuel assembly structure (spacer grids, nozzles, etc). For example, there are fuel assemblies where the outer straps of the spacer grids are bent which could prevent the loading of the fuel assembly into the cells of the spent fuel canister (Fig 12a). To solve this situation Enusa has performed an equipment (Fig 12b) incorporating a cutting wheel to reconstitute the fuel assembly envelope allowing for the fuel assembly loading into the canister cells (Castaño-Marcos and Bas-Molina, 2021).



(a) **(b)**

Fig 12. (a) Replica of typical grid defect that could be repaired and (b) repairing process (Castaño-Marcos and Bas-Molina, 2021).

Modelling programs

As previously cited, cladding temperature and hoop stress are key to evaluate the SNF performance under dry storage conditions and accurate values of them are necessary to eliminate conservatism in the evaluations. Therefore, a better understanding of both parameters during dry storage is one of the most challenging issues in the spent nuclear fuel safety assessment.

Enusa has developed several lines of investigation using CFD codes over the last decade to simulate in the most realistic way the temperature of fuel under dry storage conditions. These activities, described in section 3.2.2, have been carried out in collaboration with UPM-ETSII and, based on these studies, it has been possible to qualify a methodology for thermal calculation of spent nuclear fuel containers (Benavides et al., 2020, 2019a, 2019b). Currently, Enusa is working on a project to thermally characterize one dry cask container considering the fuel assemblies storage in it, in detail.

Regarding hoop stress, Enusa has developed its own procedure and tools to determine the hoop stress of the fuel cladding under dry storage conditions for SNF. This procedure reproduces the most realistic conditions of irradiation and dry storage and was implemented to be used for different dry cask container designs (Viñas-Peña and Lloret, 2015). More than 500 fuel assemblies have been evaluated by this tool regarding to radial reorientation and has provided peak values for analysis related to mechanical properties of cladding with defects.

Additionally, it has been generated a methodology to determine the conditions that should be met by the fuel rods with spalled oxide to be classified as undamaged during dry storage and transportation conditions. The result of this project has been a methodology (Fig 13), validated with irradiated experimental results from hot cell tests, which evaluate the mechanical capacities of spalled fuel under dry storage conditions (Ascarza et al., 2017). It has been licensed by CSN and set the acceptance criteria for SNF with spalling under dry storage conditions to be classified as “Undamaged”. Currently, this methodology has been developed for two different dry cask containers and more than 800 fuel assemblies have been successfully classified as “Undamaged” for its dry storage based on the fulfilment of these criteria.

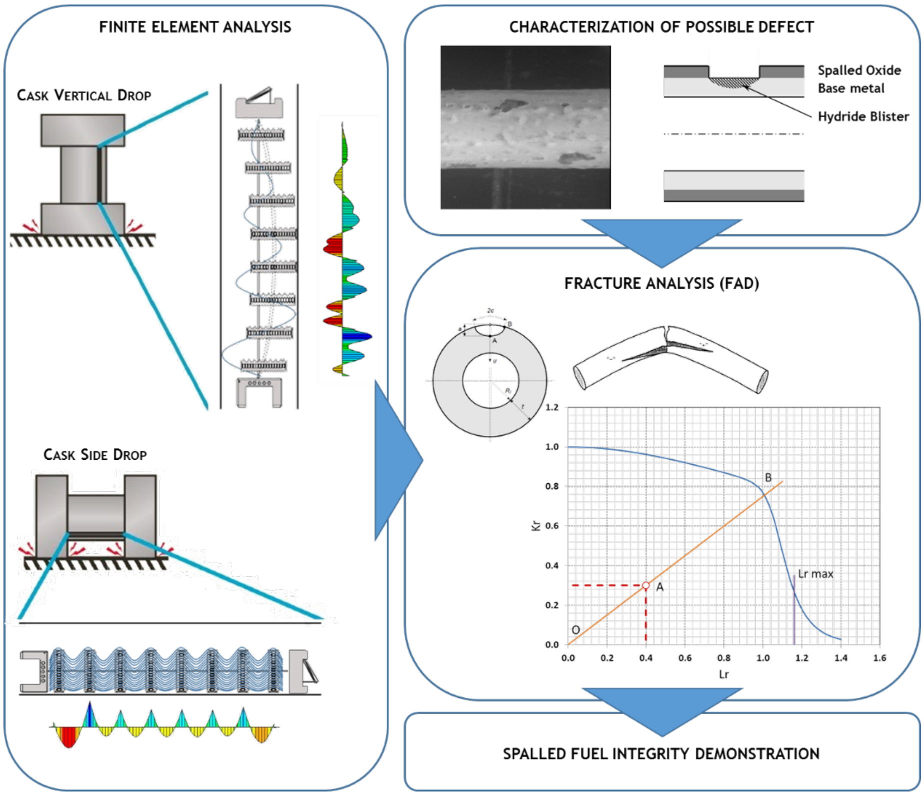


Fig 13. Methodology for the integrity evaluation of SNF with spalling (Ascarza et al., 2017).

Moreover, Enusa has been engaged in several projects to determine the isotopic composition of SNF samples over last years. It is essential to validate the prediction codes of nuclide compositions throughout accurate isotopic composition experimental results, both for burnup credit techniques applied to criticality safety analysis and source terms calculations (gamma-ray, neutron emission and decay heat). For that, experimental measurements with irradiated fuel samples, representative of the current PWR and BWR fuels, were performed in hot cells. Chemical analysis of dissolved fuel samples was performed by Thermal Ionization Mass Spectrometry (TIMS) and High-Performance Liquid Chromatography (HPLC). As an alternative, Laser ablation coupled with Inductively Coupled Plasma Mass Spectrometry (LA-ICP-MS) was applied along several diameters on pellet surfaces (Fig 14). Compositional measurements were analysed with typical computer codes used in the nuclear industry from SCALE as TRITON and Polaris (Berrios-Torres et al., 2022).

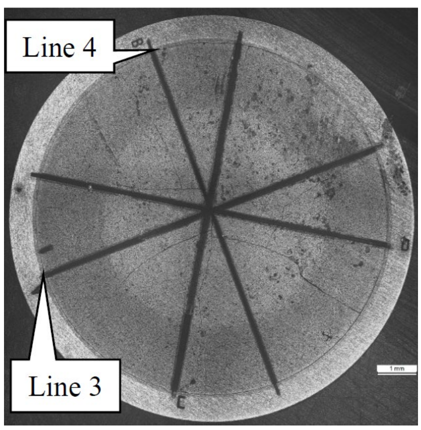


Fig 14. Example of UO_2 sample after LA-ICP-MS measurement.

Data management programs: GICOM tool

The current Spanish legislation indicates a list of definitions to classify the SNF according to its status in relation to the specific safety function of the fuel and related to the system (dry cask)-related function. Different evaluations must be done, and large amount of data are needed: manufacturing data, irradiation history and inspection results of fuel assembly examples of them. This information needed for the analysis and the big population of fuel assembly to be classified to each NPP has boosted the development of a knowledge management tool to help the characterization and classification process: GICOM, a data management tool which simplifies and support the process performed by Enusa for classifying the SNF for dry storage. GICOM provides all the necessary information to characterize and classify nuclear fuel for its storage and transport in containers, and for any other analysis regarding irradiation parameters, complying with the highest quality standards of the industry and Enusa (Viñas-Peña, 2017) and it has been used to provide to Spanish NPP all the information that ENRESA requires for SNF management activities.

3.3.2. ENSA

Regarding ENSA's research for the optimization of spent fuel cask regionalization, Fig 15 shows the individual contribution of each SFA to dose rates through the lateral ring detector shown in Fig 1 for a loaded cask of 32 positions. Fig 15 includes gamma and neutron terms separately along with the total dose rate.

Likewise, Fig 16 shows the individual contributions of each SFA to dose rates over the cask lid centre when a 32-position cask is loaded. The contributions to dose rates 2 m below the centre of the cask bottom follow similar distributions to the ones above the cask, except for the *channelling effect*, which is very limited in the case of dose rates below the cask.

Self-shielding effects are noticeable on the side of the cask. Particularly for the gamma term, which is highly attenuated by neighbouring SFAs. This means that the major contributors to gamma dose rates on the side of the cask are the SFAs loaded in the outermost region of the cask, while fuel located in the innermost cells contribute altogether with nearly 0.01% to gamma dose rates on the side of the cask. Regarding neutron flux, self-shielding effects are less prominent than for the gamma term. As a conclusion consideration of self-shielding is crucial when developing loading strategies. As for the top and bottom, the contribution to dose rates due to neutron and secondary gamma fluxes above and below the cask is slightly greater (+5% of the mean value) by SFAs located in the innermost regions of the basket, while the contribution of SFAs located in the outermost cells is lower (-15%) than the mean value per fuel assembly.

Conversely, the contribution to gamma dose rates above the cask does not follow an analogous distribution: SFAs that contribute the most to gamma dose rates at the considered locations are the ones in the basket periphery. Particularly: the "corner cells" (see reddish cells of the gamma term in Fig 16) are the greater contributors to gamma dose rates above the cask lid. This characteristic may be called *channelling effect*, as it is the result of gamma radiation being scattered or *channelled* upwards by the cask forge, due to photon scattering on the cask inner wall. This occurs even despite the presence of the two cask lids. The closer a SFA is to the cask inner-side steel wall, the greater the upwards scattering is. The effect is significant for dose rates above the lids but very limited for the calculated dose rates below the cask bottom.

Typically, nearly 90% of total dose rates above and below the cask are due to neutron flux (as the lids and bottom forge are made of steel, which mainly attenuates gamma radiation). Thus, the contributions to total dose rates above and below the cask closely follow the distribution of the neutron term. Consequently, when developing cask-loading strategies, the neutron term of fuel loaded in the innermost cells of the basket is a key concern to be considered for dose rate limitation above and below the cask. These results are shown for a 32-position cask, but analogous patterns arise for other ENUN designs with different SFA capacities.

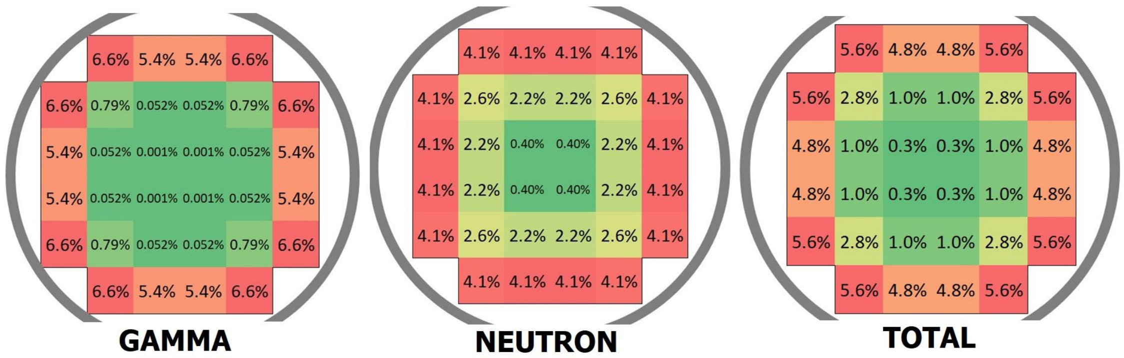


Fig 15. Contribution of each spent fuel assembly loaded in a 32-position cask to the total dose rate at 2 m from the side of the cask (ring detector) and breakdown.

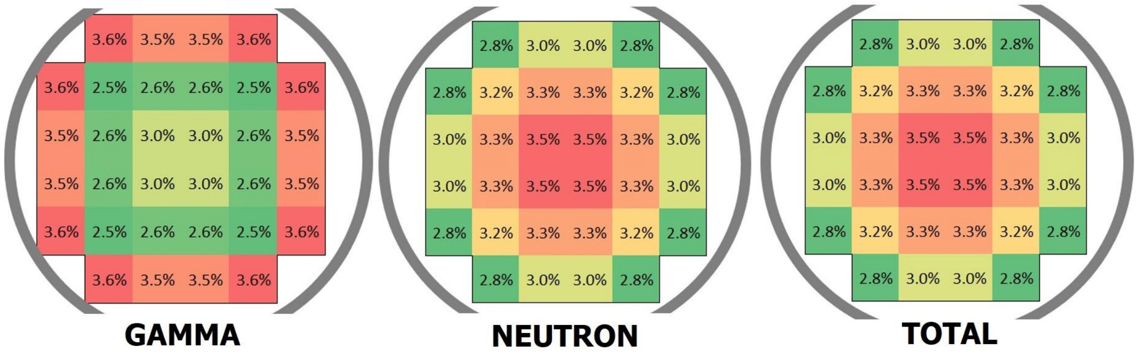


Fig 16. Contribution of each spent fuel assembly loaded in a 32-position cask to the total dose 2 m above the cask outer lid, and breakdown (neutron and gamma terms).

3.3.3. IDOM

The main outcome in this case is related to the development and implementation of an acceptance system based on the characterization conducted by the producers. This methodology involves defining a series of stages in which different criteria will be applicable, ensuring the traceability of information throughout these stages, establishing the links between fuel assemblies and storage and/or transport cask, and considering the container's management in the corresponding interim storage facility.

The stages of the acceptance methodology are: documentary stage, container loading, acceptance, and delivery. As previously cited, each stage has different criteria that must be fulfilled to classify the fuel as acceptable, accepted, and delivered at the cask loading stage.

In the documentary stage, the acceptance criteria associated with the type and condition of the fuel are verified based on the information contained in the above mentioned GECYRE database. This stage also takes into account information related to fuel repairs and conditioning, where applicable.

In the container loading stage, it is necessary to ensure compliance with the documentary acceptance criteria and verify the criteria directly related to the authorized contents in each container.

In the acceptance stage, following a period of storage in a temporary storage facility, it is important to verify the criteria associated with this storage period in the interim storage facility, ensuring once again the proper traceability of information between the fuel, container, and storage facility.

Lastly, a delivery stage is defined when the management of the spent fuel is transferred from the power plants to ENRESA. The system establishes delivery deadlines for information from the power plants and evaluation by ENRESA and requires the development of a series of formats and procedures for its proper implementation.

4. Expected developments

4.1. Research entities

4.1.1. CIEMAT-HLWU

The effects affecting fuel matrix degradation can be studied and parametrized coupled or decoupled by means of using model materials. Frequently, routine characterization techniques applied for ceramic materials are not accessible for irradiated fuel, due to the difficulties involved in its handling. Currently, an experimental program is conducted within the CIEMAT-HLWU to obtain laboratory data, which could also be scaled up to dry storage conditions.

Moreover, the development of advanced methods together with traditional ones is a challenge to provide effective methods to monitor UO_2 oxidation, and could be directly applied in the study of irradiated fuel. To provide a direct signal of the oxidation progress in real time, there are currently characterization methods to get an early detection of U_3O_8 that would be advisable and would extend the knowledge to "realistic" systems. Especially interesting are the techniques exploring the surface of the sample, because of the aforementioned formation mechanism of U_3O_8 (spallation from the grain surface). In this regard, the use of alternative *in-situ* methods added to other more standard and reliable techniques, for example modifying Raman spectroscopy or XRD set-ups, would be appropriate. The success of this combination strategy lies in the fact that may be an ideal approach for reducing operational handling of irradiated materials and, at the same time, to get a more precise characterization and monitoring of the degradation processes that could occur on real irradiated fuel. The Linkam-Raman protocols, as an *in-situ* oxidation technique, have been developed under several conditions, and the next challenge would be to make use of a fiber that can be remotely controlled, opening new pathways in the analysis of spent fuel oxidation. Additionally, an *in-situ* oxidation method using XRD is currently being developed under controlled atmospheres in which diffractograms are continuously acquired to observe the U_3O_8 formation. The new capabilities provided by these techniques would allow going deeper into the safety during storage of spent fuel, making possible the early detection of U_3O_8.

Another key issue to explore is the role of the porosity, grain size, density or specific surface area on the physical properties of irradiated fuel, especially with the focus on the rim structure on the fuel periphery. New model materials are expected to be obtained simulating the SNF microstructure, to parametrize the role of morphology effects on fuel oxidation.

This research may be complemented with the Enusa´s work with irradiated material (shown below).

4.1.2. CIEMAT-UNSR

The strategy set years ago by the CIEMAT-UNSR to support even safer fuel rod designs and predisposal operations still stands. Nonetheless, a look ahead allows identifying several sources of innovation to be brought in:

- Foreseen changes in the own research domain: the fuel rod (design and operation) during the pre-disposal stages. Particularly, new designs called advanced fuels (beyond ATFs), coming with new reactor systems expected to be deployed even within the decade; more challenging operating conditions searching for a better and still profitable fit of nuclear power to evolving electricity grids; and, no less important, search for optimizing back-end fuel management till its final disposal.
- New generation of analytical tools (their way of use included) and methodologies. The FRAP family evolution to FAST (the reference USNRC analytical tool for steady and transient irradiation and dry storage of nuclear fuel) is already a fact, but there is still a path to make it as sound and robust as its predecessors have become. Additionally, and without abandoning the mainstream of analytical modeling approach, other tools like TRANSURANUS, SCIANTIX, or even multi-scale multi-dimensional codes (like BISON or OFFBEAT) are to be considered as potential sources of insights into fuel rod behavior that might be embedded in the analytical capabilities used.
- Expansion of the safety domains to be investigated. Particularly, a methodology to estimate the number of fuel rods under accidental conditions during transport/handling of spent fuel is foreseen. These further extensions will come along with in-house model developments that will be later fed into the analytical tools being used.

4.2. Universities

4.2.1. UPM-ETSI Caminos

From the outcomes previously shown, it should be noted that the reorientation stress employed (140 MPa) is very high (the typical threshold value for the beginning of reorientation being 90 MPa for this material (Ruiz-Hervias et al., 2021)). However, as the cladding material is unirradiated, its ductility should be considerably higher than the one of the irradiated material. Consequently, it could be argued that the harsher conditions would compensate for the larger ductility of the surrogate cladding. One way planned to account for this would be a damage parameter that would allow us to compare the results of the unirradiated pre-hydrided samples with the cladding after irradiation.

4.2.2. UPM-ETSII

In case of UPM-ETSII, several dry cask models with the system code TRACE v5 are under development to test the adequacy of the code to simulate the complexity of the dry cask geometry and its ability to calculate the thermal distribution and the PCT.

4.3. Industry

4.3.1. Enusa

Enusa R&D future initiatives are aimed to study and model the behavior of cladding defects such as cracks and the pellet oxidation on fresh and spent fuel at elevated temperatures in air or partially inert atmospheres with the storage conditions.

Although UO_2 oxidation has been widely studied, most of investigations were referred to unirradiated analogues or low burnup spent nuclear fuels. Moreover, many important aspects of the oxidation process of irradiated fuel are not yet fully understood such as its dependency to temperature and oxygen content in the oxidizing atmosphere.

Therefore, to determine the most appropriate parameters for storing and handling spent fuel in air or in partially inert atmospheres, Enusa and ENRESA are collaborating in carrying out hot cell oxidation tests on irradiated fuel with different burnups. The influence of time, temperature and percentage of inertization, in the controlled oxidation of the fuel, has been analyzed by thermogravimetry; while X-ray diffraction has served to determine the progress of oxidation through the identification of uranium oxides present in the evaluated materials. The local burnup of the fuel to be tested was measured by gamma scanning.

The evaluation of the oxidation behavior of the spent fuel will allow obtaining a criterion to completely avoid the risks of oxidation of the fuel, which will serve as a conservative basis for the design, safety analysis and selection of operating parameters of a hot unloading facility in the final repository of spent fuel.

4.3.2. ENSA

Following the analysis made by ENSA for optimizing the spent fuel cask regionalization, smart-loading strategies for specific SFA combinations may be defined to optimize the regionalization from a shielding (ALARA) point of view, by allowing SFA neutron and gamma intensities to follow an inverse law of the “basket maps” shown in the figures above as far as practicable, i.e.: compensating source terms with position effects, provided that other safety functions allow it (such as restrictions derived by the thermal analyses).

Additionally, some possible design variations of this cask may be applied in the future, if the cask is used for uniform-load only, to reduce dose rates around the cask, such as the installation of auxiliary neutron-shielding material in the closure system to attenuate neutron flux over the cask lids or modifying the axial position of the SFAs inside the cask cavity, to control channeling effects, etc.

4.3.3. IDOM

The expected developments in this case are related to the fact that during the implementation of the acceptance system, a continuous evaluation should be carried out to identify its strengths and weaknesses. This involves gathering feedback and input from the involved stakeholders, such as NPPs and ENRESA. This feedback can be used to improve and adjust the system based on identified needs and challenges, as it is shown in the following bullets:

- Optimization of acceptance criteria. Over time, opportunities may arise to optimize the acceptance criteria established in the system. This entails reviewing and updating the existing criteria as more information is obtained about the behavior and characteristics of spent fuel. Regular review of the criteria ensures that they are appropriate and aligned with international standards and best practices.

- Enhancement of traceability and information management. Ensuring traceability and efficient management of information related to spent fuel is crucial. Improvements can be implemented in the systems and processes used to collect, store, and access relevant data. This includes the development of tools and technologies that facilitate information management, such as tracking systems and updated databases.
- Reinforcement of coordination between NPPs and ENRESA. Collaboration between NPPs and ENRESA is essential for the success of the acceptance system. Effective communication and coordination mechanisms must be established to ensure a continuous flow of information and a shared understanding of requirements and procedures. This can be achieved through regular meetings, joint working groups, and the promotion of a culture of collaboration and transparency.

These ideas for expected developments are focused on improving and strengthening the acceptance system for spent fuel, with the aim of ensuring safety and efficiency.

5. Conclusions

The R&D in Spain about dry interim storage aims to achieve a precise characterization of the stored and transported SNF, given that it is essential for the corresponding safety assessments. To do so, experimental and modelling work is performed by Research Entities, Universities and the Industry. This paper shows the main progression in this field, as well as the perspectives for future R&D.

Regarding the experimental research, it addresses the degradative mechanisms that the pellet-cladding system may undergo. Specifically, the activities carried out are mainly focused on experiments related to the UO_2 oxidation, mechanical tests on hydrided claddings and analyses of hydrides blister at the cladding, and SCC in structural components of the fuel assembly. Concerning the analytical related activities, the final aim is to develop methodologies that accurately characterize the SNF in the storage system by using simulation codes. It implies thermo-mechanical modelling, computational fluid dynamics calculations, and isotopic composition and neutron transport estimations.

The future Spanish R&D in this field is expected to be focused on further improvements in the characterization of the SNF. This is planned to be achieved through the use of new experimental and analytical tools, as well as the study of new fuel designs. The final aim is to further enhance the capabilities gained so far, making dry interim storage of the SNF even safer for long-term.

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