SAND2019-8159R

US/EURATOM I-NERI 2019 ANNUAL REVIEWASSESSING

THE INTEGRITY OF HIGH BURNUP SPENT NUCLEAR FUEL IN LONG TERMSTORAGE AND TRANSPORTATION

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US/EURATOM I-NERI 2019 Annual Review

Assessing the Integrity of High Burnup Spent Nuclear Fuel in Long Term Storage and Transportation

U.S. DOE Laboratory: Sandia National Laboratories (SNL)

Collaborating Investigating Organization:

Joint Research Centre, Institute for Transuranium Elements (JRC-ITU)

Other Key Collaborating Organizations:

Pacific Northwest National Laboratory (PNNL), United States
Swiss National Cooperative for the Disposal of Radioactive Waste (NAGRA)
Bundesanstalt für Materialforschung und -prüfung (BAM), Germany

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Program and Work Package Number: FT-SN080505

R&D Area: Used nuclear fuel storage and disposition technologies

Scope, Goal, and Objectives of this Initiative:

This project addresses the important issue of validating the integrity of spent nuclear fuel storage for extended periods of time, followed by transportation. While it is believed that this fuel is safe in its current condition for long periods of time, confirmatory data and analyses need to be obtained to validate our understanding of used fuel degradation mechanisms that may impinge on the integrity of the fuel to withstand long term storage and transportation conditions. This is especially true for high burnup fuel (> 45 GWD/MTU) that is currently being discharged.

The international community recognizes the importance of these issues. Moreover, several European countries now envisage to subject mixed oxide (MOX) fuel to extended storage and direct disposal. The institutes collaborating on this proposal all have active programs focused on resolving these very issues. Collaborating together provides a leverage of programs and funding that will benefit each program individually as well as the commercial nuclear industry, as a whole.

The goal of this I-NERI work is to collaborate through these institutes to develop the technical basis to justify the safe extended storage and transportation of high burnup fuel. Specifically, the Key Objectives include;

1. Develop data associated with realistic mechanical loads that would be expected during storage and transportation of used fuel.
2. Conduct laboratory experiments on real and surrogate fuel rods that measure the mechanical properties of cladding and the stiffness and strength of real spent fuel rods in a hot cell.
3. Given used fuel mechanical and constitutive properties, coupled with realistic loading, conduct finite element analyses to estimate fuel response to Normal Conditions of Transport (NCT) for truck and rail transport.
4. Provide a final report assessing the ability of high burnup fuel to maintain its integrity during extended storage and Normal Conditions of Transport.

The remainder of this annual report will describe specific work that has been performed over the past year. It will detail the work done at the DOE labs, followed by the contributions performed at JRC-ITU, NAGRA, and BAM.

DOE Lab Contributions:

The DOE Spent Fuel and Waste Science and Technology (SFWST) program funds the work for the labs that is performed under this I-NERI. The work described herein is divided into topical areas:

- Gap Analysis
- Fuel Clad Characteristics
- Thermal Profiles
- Stress Profiles

Gap Analysis

As part of the SFWST program, comprehensive gap analyses have been performed on an intermittent basis to ensure that the R&D being performed has been correctly identified to support the fundamental objective to ensure the integrity of spent fuel during long term storage and transportation. Two reports in this series of gap analyses have been written this year. The first report, "Gap Analysis to Support Extended Storage and Transportation of Spent Nuclear Fuel: Five Year Delta" [1], re-assesses technical gaps using work that has been done up through FY17. The second report, "Gap Analysis to Support Extended Storage and Transportation of Spent Nuclear Fuel: An FY2019 Assessment" [2], recognizes and incorporates a significant amount of R&D that has been done since 2017. This report will be finalized in the summer of 2019.

The results of these reports are consistent with past reports while recognizing the advancement in knowledge and identification of new issues resulting from operational advancements and evolving policy positions. The priority of high level gaps that are related to spent fuel and the scope of this I-NERI, based on the draft gap analysis [2] are:

1. Thermal profiles
2. Cladding issues – H_2 effects
3. Stress profiles

Each one of these gaps can be related to the three topic areas discussed in this report: Fuel Clad Characteristics, Thermal Profiles, and Stress Profiles.

Fuel Clad Characteristics

Four reports have been published this year that address high burnup spent fuel cladding performance and characteristics under realistic extended storage conditions. The first two reports provide experimental results of ring compression tests of high burnup spent fuel cladding; "Results of Ring Compression Tests" [3] and "Ductility of High-Burnup Fuel ZIRLO™ following Drying and Storage" [4]. These reports assess the ductility characteristics of spent fuel cladding relative to rod internal pressure (RIP), hydrogen content, and temperature. A general result of this work is that under the ranges of RIP, hydrogen content, and temperatures that spent fuel cladding operates under, the driving parameter effecting ductility is the RIP, which relates to cladding hoop stress. In general, for hoop stresses < 80 MPa, the cladding will maintain its ductility under storage and transportation loading conditions.

Two important reports have recently been published that detail initial experimental results on tests associated with the sister rod test program that is gathering data on high burnup rods that are similar to the rods in the assemblies that have been used for long term storage in the DOE demonstration project. These experiments are directly addressing gaps related to rod performance under long term storage and transportation loading conditions. The work is being performed at Pacific Northwest National Laboratory and Oak Ridge National Laboratory. The two reports are: "Initial Results of Destructive Examination of Ten Sister Rods at PNNL" [5] and "Sister Rod Destructive Examinations" [6]. One important general result is the End of Life RIP measurements for all the rods at both laboratories measured < 5.5 MPa. An internal RIP of 5.5 MPa relates roughly (recognizing thermal conditions affect RIP directly) to a hoop stress of ~50 MPa, well below the 80 MPa hoop stress limit determined in [3,4] that affects ductility of the cladding material.

Thermal Profiles

Three reports have been generated this year that relate to analysis and testing of thermal environments that spent fuel is subjected to during transfer from wet to dry storage, long term dry storage, and transportation. Thermal conditions directly affect rod characteristics associated with RIP, hydride effects, and hoop stress. These characteristics, in turn, affect how the rods will respond to thermal and mechanical loads in storage and transportation operations.

The first report, "Thermal Modeling of TN-32B Cask for High Burnup Spent Fuel Data Project" [7] detailed the computational fluid dynamics modeling of the DOE spent fuel cask demonstration. For this demonstration, high burnup fuel was selected for dry storage in a TN-323B cask to obtain thermocouple (TC) data related to the peak cladding temperature (PCT) of the rods during the drying process. The fuel was selected to drive the temperatures up as high as possible to approach the PCT limit of 400° C, as set by the Nuclear Regulatory Commission (NRC). This data was then used to verify and validate the thermal modeling codes. Surprisingly, the TC readings were well below the 400° C limit. The maximum measured temperature was 229° C and calculated estimate was 238° C. These results provide solid V&V for the CFD modeling as well as demonstration that PCTs for this dry storage system are well below the NRC thermal limit.

The second report, "Modeling Validation Exercises Using the Dry Cask Simulator" [8], resulted in the CFD modeling verification of experimental data obtained from a separate thermal experiment conducted at Sandia National Laboratories. Modeling and analyses were performed by four institution; the US Nuclear Regulatory Commission, Pacific Northwest National Laboratory, Centro de Investigaciones Energeticas, Medioambientales y Techologicas (CIEMAT), and Empresa Nacional del Uranio, SAS (ENUSA). This validation exercise resulted in all models estimating the PCT within 5% RMS error.

The results of the experimental work and the analyses demonstrate that PCTs can be accurately predicted through modeling and analyses and that they are much lower in realistic operational conditions than previously thought.

Stress Profiles

The main effort under this topic is the Normal Conditions of Transport (NCT) tests conducted in 2017 to obtain strain and acceleration data on surrogate spent fuel rods. This data was then reduced and analyzed, culminating in the report; "Data Analysis of the ENSA/DOE Rail Cask Test" [9].

Importantly, the data and subsequent analyses demonstrate that strains and accelerations resulting from the NCT testing were very small. These resultant low strains and accelerations, combined with the mechanical characteristics discussed earlier, provide substantial margin for the integrity of spent fuel during NCT operations.

Conclusions for the DOE work

This past year resulted in considerable progress through experimental tests as well as modeling and analysis. In all this work, there were no unexpected behavior and results are leading to the development of a strong database that will help justify the position that spent fuel will maintain its integrity during extended dry storage as well as transportation. If fact, it appears there will be substantial margins of safety under normal storage and transport operational conditions. Further work is planned to build on this database and to address new issues as operational and policy conditions change.

<u>Collaborating Organizations Contributions</u>

BAM

In the context of the safety assessment about the integrity of spent fuel assemblies under transport conditions, numerical simulations are performed. The fuel element segment modeled represents the part of a generalized BWR fuel assembly between two spacers. Dynamic and quasi-static finite element calculations are performed to simulate the spent fuel behavior under regulatory defined accident conditions of transport. Beam elements are used for the modeling of the fuel rods representing the compound consisting of claddings and fuel pellets. The dynamic load applied is gathered from an experimental drop test with a spent fuel cask performed at BAM. A hot cell bending test performed at JRC Karlsruhe is the basis for obtaining the material behavior of the fuel rods. The material properties are determined by simulating the test setup of JRC and optimizing the results to fit the experimental load deflection curve. The simulations of the fuel assembly segment are used to get a better understanding about the loads on fuel rods under accident conditions of transport, see [10]. Other studies at BAM are concerned with the fracture mechanics based failure analysis of embrittled unirradiated cladding tubes under Ring Compression Test conditions [11] and the spent fuel characterization and evolution until disposal within the European Joint Programme on Radioactive Waste Management (EURAD) project.

JRC

The mechanical stability of a PWR fuel rod with burn up 67 GWd/tHM has been investigated. A pressurized rod segment has been subjected to impact and bending experiment with devices installed in a hot cell. The specimen impacting has been recorded by a high speed camera, the rupturing load and overall displacement could be measured during the bending test. The data were completed with post-test examinations to characterize the released mass upon rupturing. The final goal of these investigations is to determine criteria and conditions governing the response of spent fuel rods to accidental external mechanical load.

In the conducted impact and bending experiments on the 67 GWd/tHM UO_2 segment, the total mass released did not exceed the mass of a single fuel pellet. The majority was dispersed in form of fragments, but a small fraction of around 2 % was released as fine particles. Under the frame conditions of the bending test, 3 kN force is required to rupture the fuel rod.

Although the overall fracturing and dispersal behavior is similar to that observed during previous test campaigns, further tests to single out individual parameters that might affect the integrity of a SNF rod are needed. Mechanical tests on MOX SNF rods are of highest priority in the program. Isolating cladding hydride effects in the overall mechanical behavior of irradiated fuel rods is also foreseen. However, the number of experiments that can be performed is limited and there is an acute need to model them and extend the results gained at the JRC KARLSRUHE beyond the conditions that have been tested. For this purpose, theoretical approaches are developed by BAM (Germany) [12]and NAGRA (Switzerland) [13-15] in the frame of our fruitful collaboration.

NAGRA

Nagra initiated several studies and RD&D activities aimed at assessing spent fuel mechanical performance (response of spent fuel rods to mechanical stresses), but also at developing concepts for handling of consequence scenarios. The main experimental campaign is conducted at JRC Karlsruhe, using Gösgen NPP spent fuel rods segments with different burnup levels (from low to ultra-high burnup), with the focus on the effect of hydrogen load, hydride distribution and pellet/cladding interaction on the cladding integrity. Impact and bending tests on four samples have been already performed [13-18] whilst post irradiation examination are currently under development and/or in a post processing phase. It is worth to note that neither for the high burnup case nor for the low burnup case the total mass of fuel released upon fracture exceed than one pellet per fracture. On the base of these results, and supported by fracture surfaces analysis, the unzipping of fuel material from cladding tubes of SNF during handling accident scenarios (fuel crash) can be with very high probability excluded.

Furthermore, Finite Element Analysis (FEA) are used to simulate the rod's response based on Static and Transient Structural models in ANSYS Mechanical: rod specific simulations are intended to derive rod failure criteria and to examine the influence of various phenomena as burnup, gap size, pellet interaction and strain rate. In a second stage the development of macroscale models would be foreseen to tackle entire fuel assembly including the structural parts, to investigate its mechanical integrity in various incidental scenarios (transportation shocks/vibration, handling, etc.).

Other studies are currently under development to investigate the deterioration of the cladding properties resulting from Delayed Hydride Cracking, with Paul Scherrer Institute. First results are close to be released [19]. Furthermore, a conceptual study is under development to establish specific technical requirements for the encapsulation facility, focusing on fuel handling, retrieval and packaging operations. The main scope is to ensure the safe management of any damaged and degraded fuel and to implement measures for the mitigation of accident scenarios.

For the first wave of EURAD (COFUND-EJP, Horizon 2020), an RD&D Work Package (WP) on spent nuclear fuel has been established collaboratively. A WP titled "Spent Fuel characterization and evolution until disposal (SFC)" [20], aimed at reducing uncertainties in spent fuel properties with the main focus on the pre-disposal phase, was developed together with waste management organizations, research institutes and technical support organizations. The main target of the WP is to understand the performance of SNF during a possible prolonged storage prior period to its transport, but also during transport and encapsulation and final emplacement in a deep geological repository. The purpose of the program is to build the capability, on a European scale, for ensuring the safety of all safety-relevant operations, and to understand the behavior of fuel, cladding, pellet-cladding interaction and ageing effects under normal and postulated accident scenarios, in order to identify relevant or typical bounding cases at the time of re-conditioning and pre-disposal activities.

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