SPENT NUCLEAR FUEL MANAGEMENT AFTER DRY STORAGE: FUEL INTEGRITY AND SAFE HANDLING DURING FUEL ENCAPSULATION

S. CARUSO, E. VLASSOPOULOS, P. GRÜNBERG, T. STEINBACH
National Cooperative for the Disposal of Radioactive Waste (Nagra)
Wettingen, Switzerland
Email: stefano.caruso@nagra.ch

D. PAPAIOANNOU, R. NASYROW, V. V. RONDINELLA
European Commission, Joint Research Centre (JRC), Directorate for Nuclear Safety and Security
Karlsruhe, Germany

G. NEUMANN, S. TITTELBACH
Wissenschaftlich-Technische Ingenieurberatung GmbH (WTI)
Jülich, Germany

Abstract

In Switzerland, the spent nuclear fuel assemblies arising from the operation of the five NPPs are currently stored in pools at the NPP sites and, after a cooling period, are transferred to transport/storage casks which are then transported and stored in centralized dry interim storage facilities. The National Cooperative for the Disposal of Radioactive Waste (Nagra) has proposed deep geological disposal as the solution for the management of all radioactive waste. Pre-disposal activities, in particular for the spent fuel encapsulation facility and related unloading/loading and handling operations from the transport/storage casks into the final disposal canisters, are safety-relevant operations. Nagra therefore initiated several studies and RD&D activities aimed at assessing spent fuel mechanical performance, but also at developing concepts for handling of consequence scenarios. Concerning the RD&D program, the main objective of the investigations is to assess the response of spent fuel rods to mechanical stresses corresponding to normal conditions and accident scenarios by means of experiments on PWR spent fuel rod segments. The experimental campaign is conducted at JRC Karlsruhe, with the focus on the effect of hydrogen load, hydride distribution and pellet/cladding interaction on the cladding integrity. Other studies are currently under development to investigate the deterioration of the cladding properties resulting from Delayed Hydride Cracking (with Paul Scherrer Institute), as well as the deterioration of the FA structural material for long-term dry storage conditions (with Framatome GmbH). 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. Key aspects and main achievements of these ongoing programs are presented here.

1. INTRODUCTION

In Switzerland, the spent nuclear fuel (SNF) assemblies arising from the operation of the five NPPs (3 PWRs, 2 BWRs) are currently stored in pools at the NPP sites and, after a cooling period, are transferred to transport/storage casks (T/S-C) which are then transported and stored in centralized dry interim storage facilities (ZWILAG, ZWIBEZ).

The National Cooperative for the Disposal of Radioactive Waste (Nagra) has proposed deep geological disposal as the solution for the management of all radioactive waste. In particular, a spent fuel encapsulation facility is foreseen for the packaging of the SNF from the T/S-C into the final disposal canisters (FDC). These will be emplaced in tunnels in a deep geological repository, with the canisters embedded in a bentonite backfill and surrounded by the Opalinus Clay host rock [1]. Pre-disposal activities such as SNF unloading/loading and handling operations, are safety-relevant operations. Nagra therefore initiated several studies and Research, Development and Demonstration (RD&D) activities [2] aimed at assessing spent fuel performance, but also at developing concepts for handling of consequence scenarios.

Concerning the RD&D program, the main objective of the investigations is to assess the response of spent fuel rods to mechanical stresses corresponding to normal conditions and accident scenarios by means of experiments on PWR spent fuel rod segments. The experimental campaign is currently conducted at the Joint Research Centre (JRC) Karlsruhe, with the focus on the effect of hydrogen load, hydride distribution and fuel/cladding interaction on the cladding integrity. Other experimental activities are under development with the

Paul Scherrer Institute (PSI) to investigate fuel cladding behavior, with the focus on the deterioration of the mechanical properties of the cladding material resulting from Delayed Hydride Cracking (DHC) for high burnup SNF. A new experimental campaign is also foreseen in collaboration with NPP Gösgen and Framatome GmbH aimed at investigating the mechanical properties of guide tubes under simulated long-term dry storage conditions.

Furthermore, a conceptual study has been developed to establish specific technical requirements for the encapsulation facility, focusing on fuel handling, retrieval and packaging operations. The main purpose is to ensure the safe management of any damaged and degraded spent fuel assemblies (FA) and to implement measures for the mitigation of accident scenarios.

Key aspects and main achievements of these ongoing programs are presented in the paper.

2. SPENT NUCLEAR FUEL MANAGEMENT IN SWITZERLAND

In Switzerland, five light water reactors are currently in operation with a target operating lifetime of 60 years, except for the BWR Mühleberg, which will be shut down after 47 years of operation. A total of approximately 12 300 spent fuel assemblies are expected to be discharged. These fuels come from a wide range of LWR fuel types (UO_2 and MOX) produced by several fuel vendors and e.g. cladding materials, initial enrichment and operational histories vary. Much of the SNF is characterized by a very high burnup at discharge (60–70 GWd/t).

Switzerland reprocessed 771 tHM of SNF up to 2006. The SNF was sent for reprocessing in France and the UK and a total of 634 vitrified residue canisters (180-l flasks) have already been returned and are stored at ZWILAG in T/S-C as high-level waste (HLW) that will also undergo deep geological disposal.

Nagra plans to submit a general license application for a high-level waste geological disposal site by 2024. The SNF/HLW repository is expected to become operational by 2060, with emplacement up to 2075. An illustration of Nagra's disposal concept for HLW and SNF is given in Fig. 1. Further details regarding the Swiss waste management concept are provided in [3].

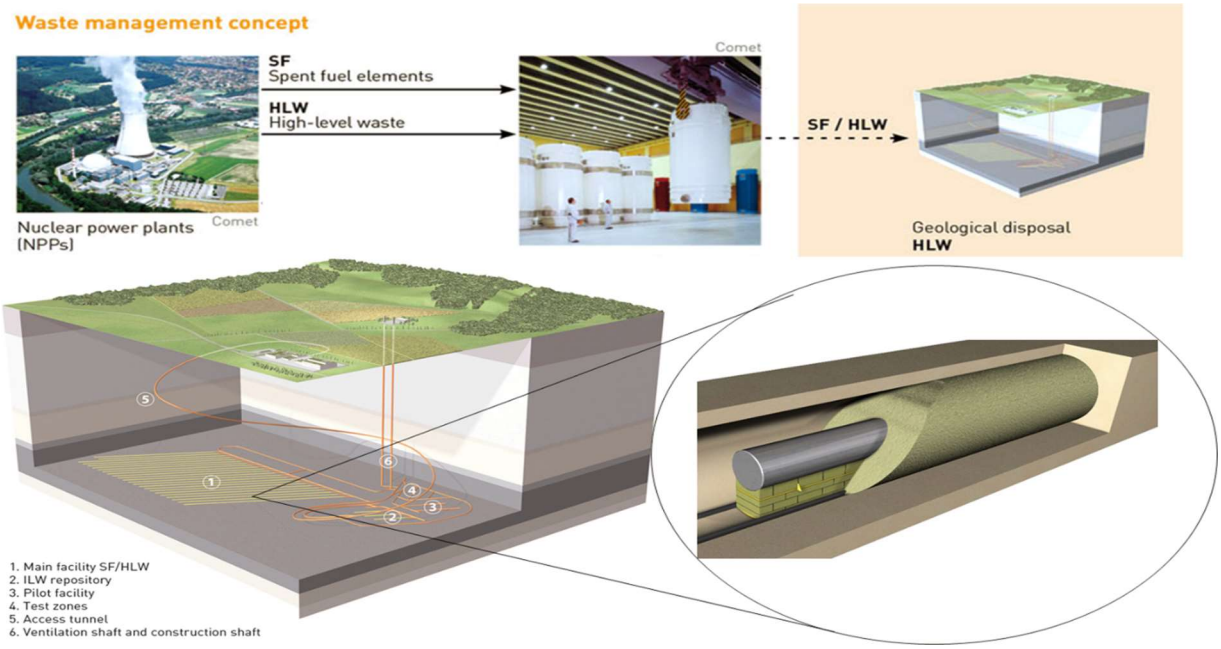

FIG. 1. Nagra's disposal concept for SNF.

2.1. Nagra's surface encapsulation facility

The spent fuel assembly unloading/loading operations will be carried out in the hot cell facility of the surface facility of the repository. Several docking stations dedicated to T/S-Cs and FDCs are planned to operate simultaneously in a hot-cell environment. Nagra has to ensure the safety of the handling operations during the transfer and encapsulation of the fuel assemblies into the FDC. Indeed, a prerequisite for the licensing of a HLW repository is the development of the scientific basis to ensure, within specified safety margins, the safety of all transfer operations with SNF. At the time of encapsulation, specific issues such as opening the T/S-C, handling of spent fuel in the surface facility and repackaging of the fuel into the disposal canisters must be properly addressed. Several safety-relevant issues need to be considered, from material aging to possible release of radioactive substances from the T/S-C/SNF after interim storage.

3. SPENT FUEL INTEGRITY AND SAFE HANDLING DURING FUEL ENCAPSULATION

The section is subdivided into two main parts: the first is devoted to the description of Nagra's RD&D program on spent fuel performance during pre-disposal activities, while the second part consists of a brief overview of the studies carried out on safety-relevant aspects of the spent fuel handling, retrieval and packaging operations in the surface facility, as well as the analysis of concepts for the safe handling of defective FAs and measures for the mitigation of accident scenarios.

3.1. Nagra's RD&D program on pre-disposal activities

Several mechanisms have been identified that could affect the mechanical properties of fuel and cladding, especially for high burnup fuel and during a prolonged dry storage period [4–5]. Therefore, confirmatory data and further analyses are needed to validate and develop a better understanding of the SNF degradation mechanisms [6–7]. The objectives of Nagra's research program can be summarized in four points, namely:

(a) to assess the response of SNF rods to mechanical loads corresponding to normal conditions of transport and accident scenarios by means of mechanical testing on commercial spent fuel rod segments, to be followed by further characterization (metallography, fuel release and fuel fragment characterization, etc.);
(b) to contribute to the investigation of potential degradation mechanisms and the behavior of SNF under conditions relevant for long-term interim dry storage, also by means of induced heat treatment of the fuel rod segments. High burnup SNF and MOX are technical gaps that will be filled;
(c) to assess the possible bounding contamination of the T/S-C and hot-cell equipment in the surface encapsulation facility;
(d) to perform a conceptual study on T/S-C and the encapsulation facility covering accident mitigation, consequence analysis and mitigation activities, decontamination and decommissioning issues.

In order to achieve these objectives, an experimental campaign on SNF rod segments irradiated at NPP Gösgen was initiated and is currently conducted at the JRC- Karlsruhe [8], with the focus on the effects of hydrogen load, hydride distribution and pellet/cladding interaction on the cladding integrity. The agreement between Nagra and JRC-Karlsruhe is intended to build on partnerships with many other organizations, e.g. participation in the US/EURATOM INERI[1] program "Assessing the Integrity of High Burnup Spent Nuclear Fuel in Long Term Storage and Transportation" as well as the IAEA Coordinated Research Project "Spent Fuel Performance Assessment and Research (SPAR IV)" [9].

Other experimental activities to investigate fuel cladding behavior and FA structural materials are currently under development, devoted to long-term dry storage aspects and focused on high burnup SNF.

[1] International Nuclear Energy Research Initiative

3.1.1. Nagra-JRC Karlsruhe collaboration

A collaborative research arrangement between Nagra and JRC-Karlsruhe was signed in 2016, initiating a multiyear experimental campaign to assess the properties and behavior of spent nuclear fuel after discharge from the nuclear reactor, covering in particular extended storage, transport, retrieval thereafter and disposal in a deep geological repository. Previous experimental investigations were performed at JRC-Karlsruhe and reported in [10–12]. More recently, two new devices for mechanical testing on fuelled and pressurized spent nuclear fuel rod segments have been developed for gravitational impact and three-point bending tests [13]. The main objectives of this program are the determination of rod mechanical response in bending loads (quasi-static and dynamic), the study of rod failure progress and the characterization of fuel release in the case of rod fracture. Additionally, the hydride behavior in the cladding is assessed with a series of post-test examinations investigating and associating the Zr hydride content and morphology to the rod's response. An overview of the experimental campaign is given in Fig. 2. The campaign consists of two phases, namely the development and optimization of the mechanical testing devices in cold laboratories and their installation and application for testing irradiated spent fuel in hot cells.

Analogue studies are of great importance for developing and optimizing the mechanical devices, as well studying the mechanical behavior of hydrogenated Zry-4 under bending stresses [14]. The highly inhomogeneous material properties and geometrical configuration of the SNF prevents a direct investigation of individual phenomena (i.e. PCMI, Zr hydride orientation and/or concentration, etc.). On the contrary, the cold test campaign is conducted with standard reference materials, allowing the elimination of a large number of material-introduced uncertainties. The results were used for numerous device developments and as indications to be considered for potential degradation mechanisms (i.e. crack initiation and growth) in the experimental campaign on irradiated SNF rods [15–16].

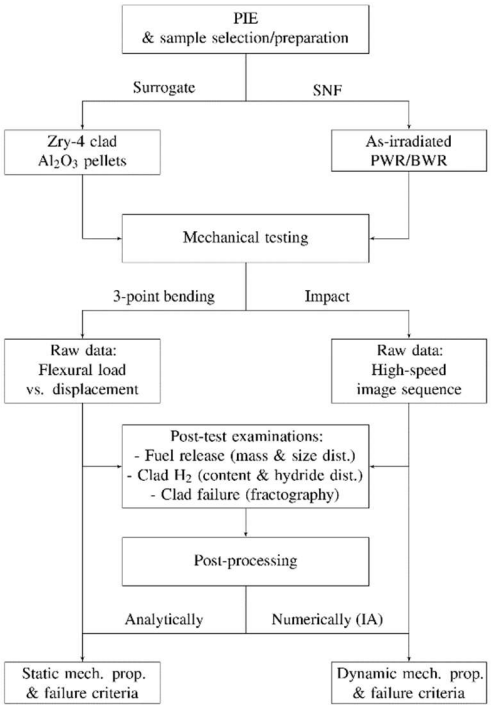

FIG. 2. Overview of the experimental campaign conducted at hot-cell facilities at JRC-Karlsruhe.

Following the analogue studies, both devices were installed in the hot-cell facilities to conduct the same type of experiments with commercial LWR SNF rod segments from low to very high burnup. A series of post-irradiation examinations were used as the basis for selecting and preparing the segments for the mechanical testing [15]. The objective of the hot-test campaign is to select, analyze and combine information from the mechanical tests and post-test examinations to characterize the mechanical response of the composite fuel/cladding system.

The goal is to derive empirical formulae to describe the flexural properties of the rods as a function of their properties (i.e. burnup, H_2 concentration and/or hydride orientation, etc.). The method of analysis and the post-processing of the experimental results differ because the acquisition capabilities of the test apparatuses are very different. Static mechanical properties and failure criteria were derived analytically from the 3-point bending test results, while an Image Analysis (IA) methodology [16–17] was developed to characterize the mechanical response of the SNF rods during the impact tests.

The post-test examinations gave important insights into the possible bounding contamination of the T/S-Cs and/or hot-cell facility in the case of cladding integrity loss. The impact test and the fragments of a low burnup fuel after the impact test are shown in Fig. 3. It can be noted that only the volume of the fuel that was directly affected by the former has been released [18–19]. This has been observed in every experiment including a wide range of burnup and fuel types, as well as in the previous experimental campaign [10–12] where different loading configurations were used. The fuel/cladding bonding (for high burnup) and/or the cladding plastic deformation (for lower burnup samples) acted against an excess release of fuel. The total mass released upon fracture is in the range of ~1–2 g/fracture. This value represents the weight difference between the intact rodlet and the sum of the segments resulting from the fracture. Although a correlation has been observed between burnup and total mass release, it is of great importance to note that all results are within the same order of magnitude. Preliminary results on the characterization of the heavy fuel fragments were reported in [14]. Around 2% of the total released mass consisted of aerosols and fine particulates which were deposited on the inner walls of the testing chamber or were collected on the filters coupled to the chamber. An aspiration system was used to "vacuum" the testing chamber before, during and after the impact test, to collect the aerosols on a sequence of filters with different mesh sizes. The filters (see Fig. 4) were examined with SEM-EDS (Scanning Electron Microscopy with Energy Dispersive Spectroscopy) and showed that the aerosols consisted of particles sizes ranging from sub-micron to > 15 micron [8]. Further results have recently been reported in [20].

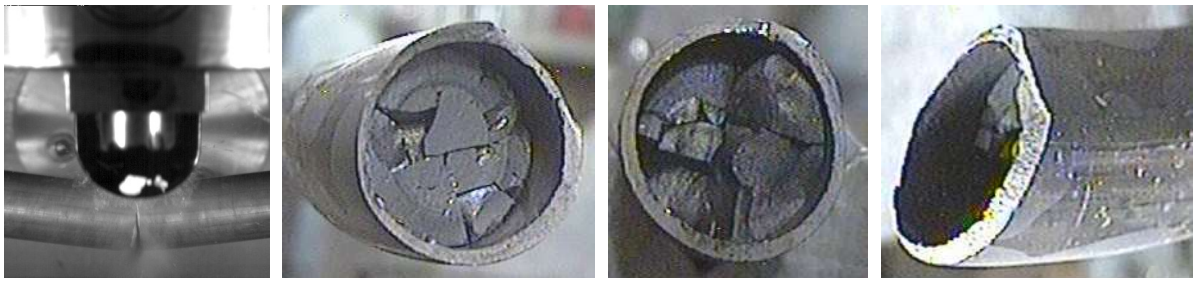

FIG. 3. Impact test (left) and fracture segments (middle and right) of low burnup PWR fuel rodlet after the impact test; fracture surfaces are formed directly under the hammer.

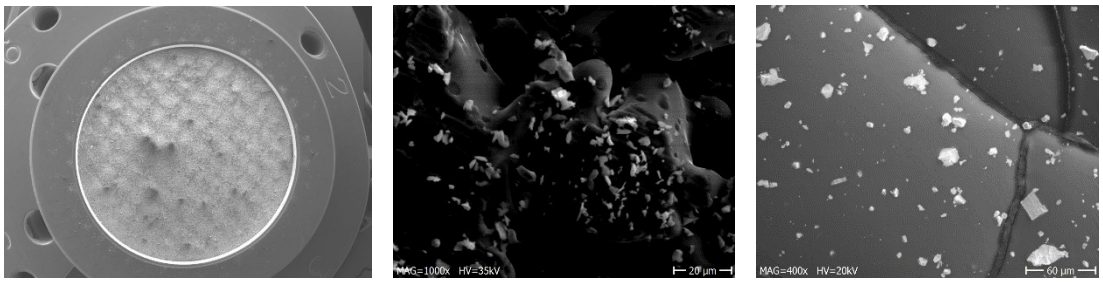

FIG. 4. Fuel release analysis: particle filter(left) and SEM micrographs (right)

3.1.2. Nagra-PSI collaboration

Complementary experimental activities are under development with the Hot Laboratory and the Laboratory for Nuclear Materials at PSI, which are performing comprehensive post-irradiation examinations (PIE) and research on fuel rods, comprising characterization of SNF up to very high burnup using standard and advanced analytical tools [21–23]. The aim of these investigations is to study the deterioration of the cladding properties resulting from Delayed Hydride Cracking as well as the effect of the pellet-cladding contact pressure on the

cladding mechanical behavior. In this respect, two main directions are considered: a) the analysis of the cladding integrity and b) the investigation of the impact of the fuel pellet on the cladding. Fuels with different burnups from BWRs and PWRs are investigated. In particular, the DHC behavior of zircaloy and its dependence on irradiation-induced embrittlement will be investigated using mechanical tests and microscopic investigations of the cladding structure. Analyses of the fuel/cladding interface, using mainly metallography and/or FIB for sample preparation, are intended to reveal residual stresses of the material influencing the mechanical behavior at the cladding interface and potentially attracting hydrogen towards the adjacent cladding. These activities, together with the study of the role of a liner for DHC (inner cladding liner for BWR or outer cladding liner for PWR) and the fatigue behavior of the cladding, have the goal of defining the boundary conditions for hydrogen diffusion and the related stress levels for build-up and reorientation of hydrides.

3.1.3. Further studies on SNF structural materials

The focus of the research work cited above is mainly directed at the spent fuel rod integrity and mechanical behavior. However, during the fuel handling operations foreseen in the surface facility, such as unloading from T/S-C as well as loading into the FDC, the role of the structural material of the FA is very important. The top nozzle of a FA is connected through guides tubes (for PWR) to the FA body, therefore, these components are crucial for guaranteeing the handling integrity of the fuel. The structure of a FA is designed to support at least two times the handling load, however it can be important to evaluate the handling integrity after the dry storage period. For this reason, a new experimental campaign is foreseen, aiming to investigate guide tube mechanical properties under long-term dry storage conditions. The work would consist of selecting guide tube samples from very high burnup fuels with different material compositions and performing a dedicated heat treatment on selected samples. Afterwards, a series of tensile tests will be performed on treated and untreated samples, followed by metallography. The analysis results of these tests should serve as supporting data for the design of Nagra's packaging facility.

3.1.4. EURAD - Spent Fuel Characterization and evolution until disposal (SFC)

For the first wave of EURAD (COFUND-EJP[2], 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)", 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.

Nagra took part in the coordination for the development of the WP proposal and, together with JRC and PSI, will contribute to the program in terms of task leadership, experiments and research activities. It is worth mentioning that this program (this WP in particular) is a unique opportunity to establish an integrated approach to the backend of the fuel cycle through the synergic and goal-oriented involvement of several research institutes and stakeholders.

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[2] European Joint Program

3.2. Conceptual study for fuel handling, retrieval and packaging operations

The study starts with a structured analysis of all FA components and their technical function with regard to the handling of the FA in the encapsulation facility and an analysis of loads during reactor operation, loading into T/S-C, long-term dry storage and subsequent transport. The evaluation of the above-mentioned loads indicates a lack of reliable information regarding a possible change in the mechanical properties of the fuel cladding during long-term dry storage and the resulting behavior of FA during transport in a T/S-C after long-term dry storage. As mentioned above, Nagra's R&D projects are intended to provide the required information.

For the analysis of FA handling operations in the encapsulation facility, damage of the FA as a result of the above-mentioned loads is assumed. Possible accident scenarios due to the damage of the FA are identified and categorized with regard to their impact on the routine operation of the encapsulation plant.

Most of these accident scenarios can be avoided by early detection of the damage, for instance

— detection of fission gas in the cask cavity;
— detection of FA handling damage by visual inspection of the FA inside the T/S-C;
— detection of structural FA damage and prevention of further damage by controlling the lifting force during lifting of the FA and special treatment of the FA afterwards.

After detection of FA damage, the FA remains in the cask, but the remaining FAs will be unloaded and the cask will be closed again. The FA receives special treatment at the end of the loading campaign. The hot cell of the encapsulation plant has to be equipped accordingly for the special treatment. After the detection of geometrical changes in the FA by visual inspection of the FA outside the T/S-C, the FA can be stored in the hot cell until a special FDC with extra large FA positions is provided. Very efficient measures to avoid dropping the FA and the contamination of the hot cell by CRUD or fuel fragments are

— fail-safe and redundant design of the crane system of the hot cell;
— use of a transfer flask after unloading the FA from the T/S-C (see Fig. 5).

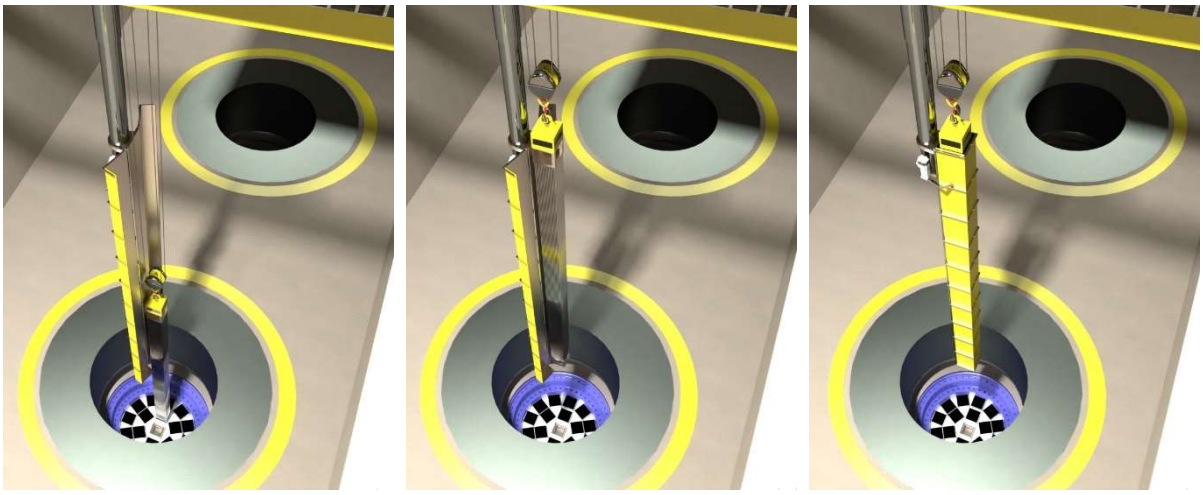

FIG. 5. Unloading the FA from T/S-C with employment of a transfer flask.

Nevertheless, it cannot be excluded that a FA may get caught during lifting out of the T/S-C or during lowering into the FDC. Both worst-case scenarios lead to stopping of the loading campaign. Although the scenarios can be handled with well-planned countermeasures such as

— cutting the T/S-C basket structures;
— disassembling the FA.

a blockage of the hot cell for a long time is expected. To reduce the risk for the loading campaign, at least two redundant hot cells in parallel operation are foreseen. Since the probability of these worst-case scenarios is regarded as very low, the hot cell does not have to be equipped with all the required handling equipment for these countermeasures but should be designed such that the equipment could be installed if required.

Based on the analysis of the accident scenarios, the requirements for the technical equipment and the design of the hot cell are derived, as well as additional requirements for the FDC, i.e. larger fuel positions for FA with deformations and the possibility to load damaged fuel rods or fuel pellets in the FDC.

Considering that the encapsulation facility will remain in operation long after the last T/S-C is loaded, the knowhow transfer regarding the inventory of the T/S-C, the operation and handling of the casks as well as the availability of cask handling equipment and cask gaskets has to be assured. The knowhow transfer should start as soon as possible with the documentation (for instance in a database) of all required information on the cask loadings, possible mishaps during cask loading and deviation of loaded FA from nominal properties.

The results of international and Nagra RD&D projects dealing with the effects of long-term dry storage on the fuel cladding properties have to be evaluated with regard to the assumptions and results of the conceptual study presented here.

4. CONCLUSIONS

Pre-disposal activities, in particular for the spent fuel encapsulation facility and related unloading/loading and handling operations from the transport/storage casks into the final disposal canisters, are safety-relevant operations. Nagra therefore initiated several studies and RD&D projects aimed at the assessment of spent fuel performance, but also at the development of concepts for handling of consequence scenarios, also in line with the decision by the Swiss Federal Council on the 28.03.2013 and the ENSI Guideline G20 [24].

The results achieved so far form a first technical basis for the development of safety and operational requirements with respect to the planning and construction of Nagra's surface encapsulation facility for spent fuel and HLW.

ACKNOWLEDGEMENTS

The Nagra/JRC experimental program is being conducted by JRC-Karlsruhe with the support of the Swiss nuclear power plant Gösgen and Framatome GmbH (former AREVA GmbH). For this reason, the authors are particularly grateful to F. Jatuff, R. Sarrafian, B. Zimmermann (NPP Gösgen) and E. Schweitzer, K. Nissen (Framatome GmbH) for their strong support of the experiments.

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