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Assessment of spent nuclear fuel during dry storage in casks with fuel rod performance codes - TÜV NORD EnSys approach for fuel rod performance calculation for an extended dry storage period

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ABSTRACT

In Germany, spent nuclear fuel is stored in casks under dry conditions in independent interim storage facilities. The most common cask design is the CASTOR® type cask. According to the licenses of the casks and the facilities, the interim storage time is limited to 40 years from first cask storage. With the Repository Site Selection Act, the discussion concerning a final repository in Germany started in 2013 again from scratch. According to this act, the new site selection procedure should be finished in 2031. Due to the time needed for erection and commissioning, the start of operation of a final repository in Germany will be after 2050. Consequently, the limited licenses for the casks and the interim storage facilities, whose operation started before 2010, will have to be extended for another period of 40 years or even longer.

The fuel cladding is a barrier for the retention of fission products and is important to keep the fuel geometry intact during dry interim storage, transportation and handling for preparing the final storage in a repository. Since the most proofs on criticality safety, heat removal and radiological loads are based on a known fuel geometry in the casks, it is necessary to preserve the fuel geometry during the entire storage time. Therefore, systematic cladding failures during the dry interim storage and subsequent transportation and handling have to be excluded. To prevent systematic cladding failures during the licensed dry storage period of 40 years the German rules contain a stress and a strain criterion. Further, the cladding temperature is limited. This is due to experimental limits to determine the annealing of irradiation damages in the cladding material. The temperature's limit is not fixed by the rules and standards and is understood as a technical limitation during cask storage in the German licensing process. In a prolonged interim storage period, other cladding criteria than the stress and the strain state of the cladding may become more important. Moreover, the cladding criteria may be affected by long-term effects not considered today.

Our approach on assessing the fuel performance under dry storage conditions considers the complete fuel life cycle from the in-pile irradiation over the wet storage in the pool as well as the cask drying procedure up to the end of the storage time. We plan to consider a storage time of about 100 years in our calculations. We have already implemented the in-pile and pool storage performance of the fuel rods. For this stage, the fuel manufacturing data and the operational data of the reactor are needed. After the wet storage time in the pool, the spent fuel is loaded into a cask. At this stage, the thermal boundary conditions change from the thermo-hydraulic model of the code to a prescribed outer cladding temperature during dry storage and its development over time. In the first step, we implemented the calculation workflow and made first calculations to comprehend the existing strain and stress criteria for the 40 years storage time. In the second step, we want to implement probabilistic methods to analyse the fuel performance in an entire cask. The third step will include the development and implementation of fuel and cladding models for long-term dry storage periods up to 100 years. In our implementation, we used the TRANSURANUS (2008) fuel performance code and our fuel database system TITANIA (2005).

INTRODUCTION

Usually during the times between the irradiation cycles and after the final in-pile use, irradiated nuclear fuel is stored in wet conditions in the pool inside the reactor building. After a certain time of wet storage, the decay heat falls below certain values, which allow to remove the fuel from the reactor site for transportation and dry storage of the spent fuel in casks. This becomes an important issue when the spent fuel pool is already filled to its full capacity or when the decommissioning of the power plant requires the emptying of the pool. The spent fuel can be transported for either reprocessing, interim storage or final disposal. In Germany, the reprocessing of spent fuel from German power reactors is excluded by the amendment to the German Atomic Energy Act of 2002. With this act, the fuel cycle has been changed from a closed cycle to an open cycle at the back end. The final repository site Gorleben was selected in 1977. However, with the German Repository Site Selection Act in 2013, the discussion concerning a final repository for nuclear and other radioactive waste started from scratch again (Spykman 2018). According to this act, the site selection procedure should be finished in 2031. Based on the experience so far, one can assume a time delay of more than a decade for the licensing of the repository and another decade for commissioning. Therefore, 2050 or even later may be a more likely date for realizing and operating a final repository for spent nuclear fuel (SNF) and high active waste (HAW) in Germany.

At present, the interim dry storage time for SNF and HAW is limited to 40 years by cask licenses starting with the loading date. The same time span applies to the interim storage facility starting with the first stored cask. Expecting a date later than 2050 for a final repository, the interim storage period has to be extended to more than 60 years. Currently, a period of 100 years is under discussion. Consequently, licenses have to be renewed for both the casks and the storage facilities (Spykman 2018).

During dry storage, the cladding integrity in terms of excluding a systematic cladding tube failure and the preservation of the fuel geometry is essential during the storage time span, since most of the proofs for heat removal, criticality safety and radiological dose on the casks surface are based on this. Therefore, cladding criteria for dry storage up to 40 years storage time have been established in Germany. These are a stress and a strain limit, a maximum cladding temperature limit and the limitation of ongoing cladding corrosion during storage time (Spykman 2013, 2015 and 2018).

Since the cask license limits the dry storage time of SNF to 40 years and a final repository will not be available in the next decades, the discussion about possible new verifications of the existing criteria and the identification of new criteria started for a prolonged dry storage time (Spykman 2018). Within this discussion, we already addressed the efforts on needed data and knowledge, calculation methods, modelling and tools. For the assessment of the in-pile performance with different boundary conditions, we implemented full core analysis methods using either best estimate, conservative or probabilistically varied data and model parameters.

Based on our knowledge and experience on assessing the fuel rod performance during the in-pile irradiation in normal as well as design-basis accident conditions, e.g. LOCA, which we use as a technical support organisation in nuclear issues, we developed an approach for the assessment of the fuel rod performance for a longer interim dry storage period. This approach includes the in-pile irradiation history, the wet storages times in the on-site pool and the dry storage in the cask up to 100 years. Each of this three life stages has its own boundary conditions and different modelling requirements. In this paper, we will show the first implementation of the approach described above.

MODELLING AND DATA FLOW

In the modelling of the needed data, there are four main areas of interest. The first one is the fuel rod design data and the fuel assembly design data, which are essential for the modelling. This includes among other parameters e.g. the geometry of all parts of the different fuel rod types, the enrichment, porosity and density of the fuel pellets and void volumes inside the fuel rod. We obtained this data from the fuel assembly design and manufacturing documentation. At TÜV NORD EnSys, we compile all this data in our TITANIA (2005) database that we developed for the assessment of the performance of all fuel rods in a core loading for a specific reactor cycle (Spykman et. al 2005). To be able to obtain best estimate, conservative or probabilistic based results each dataset is stored with its nominal, minimal and maximal value. Fuel data, which is not stored in the TITANIA database, will be added on demand. A visualisation of the full core assessment is shown on the left side of figure 1.

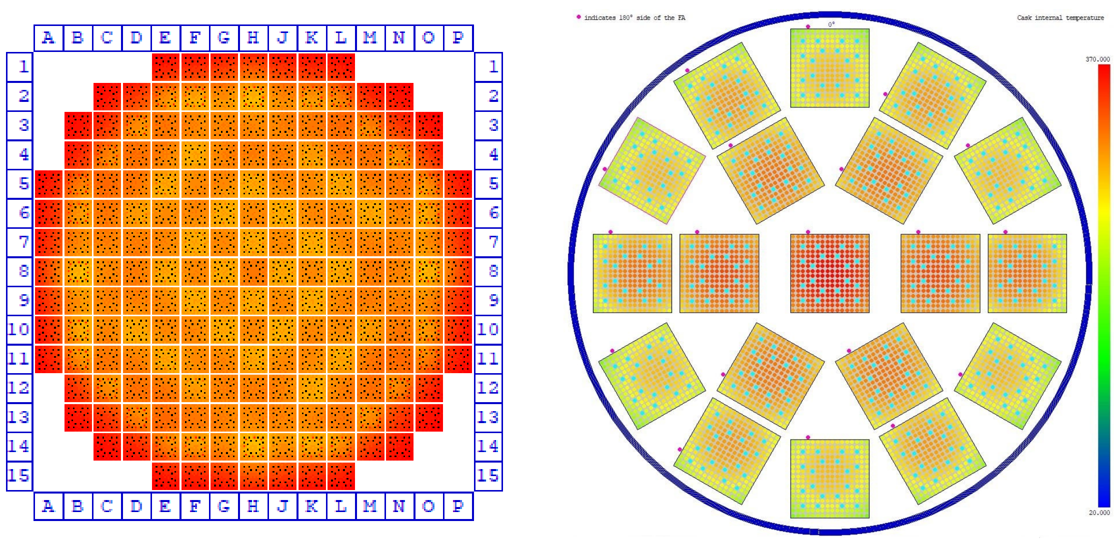

Figure 1. From full core analysis (left, colour from red to yellow indicates higher rod power) to full cask analysis (right, colour indicates cladding temperature after cask loading and drying).

The second part of the needed data for the modelling are the data of the reactor operation, which will be compiled for each fuel assembly in the cask. For the fuel performance calculation with TRANSURANUS (2008) the inlet temperature, the coolant flow rate, the coolant pressure, the fuel rod pitch as well as the axial power and flux distribution as a function of time are needed. We use in the full core analysis the so-called “equivalent full power days” power histories of the core design. These cover the loads on the fuel rods of the real core power that might be for some time lower due to load following core-operation. The datasets of the full power days power histories do not contain the time of the outages nor the time of the cycles in which the fuel was not loaded into the core but stored in the on-site pool under wet conditions. In our approach we decided to consider these times in the timeline that we use in the full cask calculations in order to be able to model all stages of the fuel rods life. An example of a single fuel rod timeline data is shown on the right side of figure 2.

The third part of the needed data is the decay heat after the fuel has been irradiated. This information is used to calculate the fuel performance during times when the fuel is stored in the pool under wet conditions as well as in the times of dry storage in a cask. This data is not used in the full core calculation and is introduced as a new kind of datasets. In our prototype of the data management system, we implemented a simple correlation by Way and Wigner (1946) to have a dataset within our timeline data. In the next chapter, we will discuss with which data this can be substituted. In the current implementation, we calculate the decay power following each irradiation cycle and add it up subsequently to get appropriate post decay power values compared to the values calculated for the loading time point. This method is adapted from the DIN 25463-1 (2014).

The fourth part of data modelling consist of determining the temperature data of the fuel rod cladding during dry storage. In the licensing process, the cask specific loading pattern with high power at loading time is evaluated with regard to the maximum cladding temperature. For a specific loading pattern, an upper limit in decay power is defined for each position in the cask. The right side picture in figure 1 shows the fuel cladding temperature of a homogenous loading pattern for a CASTOR V/19® type where each position has the same upper limit of the decay power. For the calculation of the cladding temperature we implemented a specific model that based on an ANSYS (2016) calculation carried out by TÜV NORD EnSys. Within the homogenous loading pattern the cladding temperature reaches the highest value in the middle of fuel assembly in the central cask position. All other cask positions follow a parabolic function to the lowest values at the edge of the cask. A similar parabolic function with different parameters is used to model the temperature within the fuel assembly. The cask temperature function and fuel temperature functions were superposed to get the cladding temperature. Compared to the ANSYS (2016) calculation the parabolic approach agrees with the data very accurately at the cask loading time. In addition, we calculated the axial temperature profile of the hottest rod in our ANSYS analysis. The middle picture in figure 2 shows an example of the axial temperature profile. The time course of the cladding axial temperature distribution follows the decay power course in time. In our implementation, we developed a set of formula to calculate the temperature for each fuel rod as a function of time. Starting from the individual decay power of the rod and the fact that the temperature will fall to ambient values with a flat axial profile after an infinity storage time, we calculated the time course of the cladding temperature of the fuel rods including the plenum temperatures. The blue curve in the right picture of figure 2 shows the calculated cladding temperature as a function of time over a period of 100 years of dry storage.

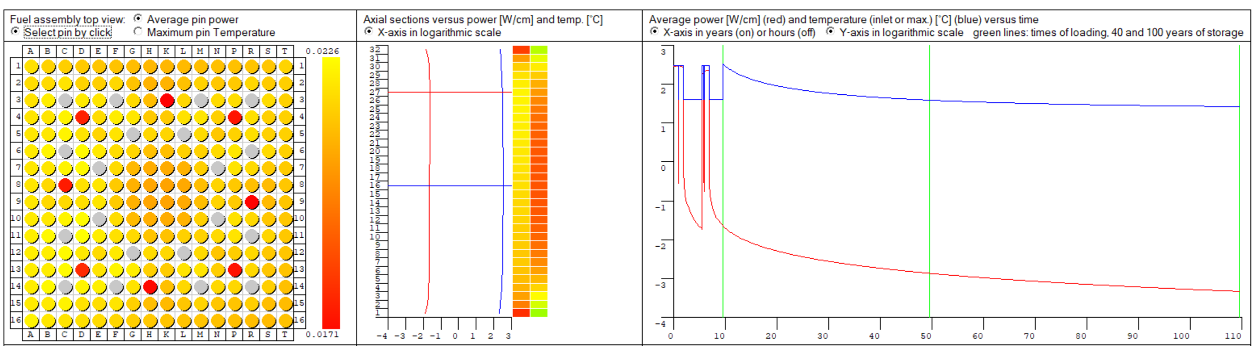

Figure 2. Modelled life of a fuel assembly from first reactor cycle, pool storage under wet conditions and cask dry storage:

left: fuel assembly top view, colour indicates average rod power at cask loading time; red dots are gadolinium doped rods with lower power, grey dots symbolise the guide tubes

middle: axial power (red) and axial temperature (blue) distributions (log scale) of a selected rod at cask loading time

right: timeline of a selected fuel rod from first in-pile operation until 100 years dry storage; green lines indicate the time of loading, 40 years and 100 years of storage

DATA SOURCES

As described in the chapter above, we implemented many simplifications and interpolations of decay power and temperatures at this stage in our prototype. We have chosen this way to "run" the system with data within realistic boundary conditions. In the following, we will describe how these "not so sophisticated" assumptions and interpolations can be replaced with data that are more accurate which are provided by validated codes or from other trustworthy sources. We plan to install import functions for all datatypes used in the calculations of fuel rod performance during dry storage. This is already fulfilled for the power data of in-pile irradiation, the plant specific data and the fuel data from the TITANIA database. This data is usually obtained from the fuel vendors and the utilities. Furthermore, it is assessed by the authorities and its technical support organisations (TSO) and is used in the surveillance of the power plant.

The decay power during the wet storage in the on-site pool and dry storage in cask is not available for this kind of calculations in the required level of detail. Currently the maximum and /or covering data for the assessment of safety aspects is determined for one time point of interest. To validate our approach, calculations with more detailed and time-resolved data are required. This calculation can be carried out by using the fuel vendors or the utilities code system, like CASCADE-3D (1999) and DINUM (2010) or the code systems CASMO-5 (2019), SIMULATE-3 (2018) in combination with the SCALE (2020) code system used by TÜV NORD EnSys or the KENOREST code developed by the GRS (2016).

Alternatively, the calculation of the post irradiation power is implemented in a more sophisticated manner in our application, e.g. by implementing the commonly accepted procedure described in the DIN 25463-1 (2014-2) and other standards. Although this involves more effort, it also offers a certain independence from costly code licences.

The most important dataset is the temperature distribution of the fuel claddings in the casks as a function of time. We have already described the procedure for calculating the fuel rod temperature vs. time using an ANSYS (2016) calculation for the period after cask loading and storage at the interim storage site. Considering the needed degree of detail, ANSYS (2016) calculation will be very time consuming and resources demanding. An alternative codes system is COBRA-SFS described in GRS (2020). A comparison between the codes COBRA-SFS, ANSYS-CFX and COCOSYS can also be found in GRS (2020). Not considering the big efforts and cost in using such code systems, we think that a set of such calculations to calibrate the temperature calculations routines implemented in our prototype may also be purposeful.

To summarize the brief overview of possible data sources with high resolution needed to calculate the fuel rod performance during dry storage in casks, we have to face a big data problem. Big in the sense of size but also big in the sense of the demands on the data sources. However, we are confident that these problems can be solved and we have already gained some experiences in both, managing big data amounts and finding appropriate sources. In addition to the modelling activities, we will continue documenting this issue.

STATUS OF IMPLEMENTATION AND FIRST RESULTS

We have implemented the automatic creation of the TRANSURANUS input files for a single fuel rod as well as for all fuel rods of a selected fuel assembly of a selected cask loading for calculations with best estimate fuel data and model parameter. All data not automatically generated from the TITANIA database can be modified manually with our TRANSURANUS input editor, which is also implemented in this prototype.

We added an output scheme to the TRANSURANUS fuel performance code that is adapted to the requirements for assessing the fuel performance under dry storage conditions. At present, we create output files for each calculated rod containing values for inner and outer cladding temperature as well as the average cladding temperature in time for all axial sections. Further, the output files contain in the same manner the hoop stress, the sum of creep and plastic strain, the oxide layer thickness and the content of hydrogen content in the cladding material. The inner rod pressure and the free volume inside the rod development in time are also included in the output as integrated values. This output scheme generates roundabout four gigabytes of data for one fuel assembly. Therefore, there was a need for a limitation of the information in the output file. In the future, we will extend the output scheme to more parameters, which will be selectable by the user. If the output of more or more detailed results is required, the user can change the TRANSURANUS input files manually before starting the calculation. However, even if this is possible for a few number of calculations, it is challenging for all rods of a fuel assembly or a cask loading. Nevertheless, the output scheme within the code has to be adapted if new aspects are in focus or new models are implemented in the TRANSURANUS code.

As mentioned before the output of the TRANSURANUS calculation can result in a huge amount of data if the complete evolution of a parameter in time for each axial section of the rod is of interest. Therefore, in addition to the pure data we have also implemented processed data in a way that displays only show the maximum values both axially and in time. The user can select this option, among others.

Figure 3 shows the maximum hoop stress and the sum of plastic and creep strain (hoop strain) as a function of time for the fuel rod at position D 10 of the fuel assembly on the left side of figure 3, which was calculated with the TRANSURANUS fuel rod performance code. Each displayed result has its own ordinate scaling in the same colour as the displayed timeline. In this example, we have limited ourselves to two results of the output scheme. Since the axial maximum values are displayed in time, the display of the axial distribution is in this case meaningless.

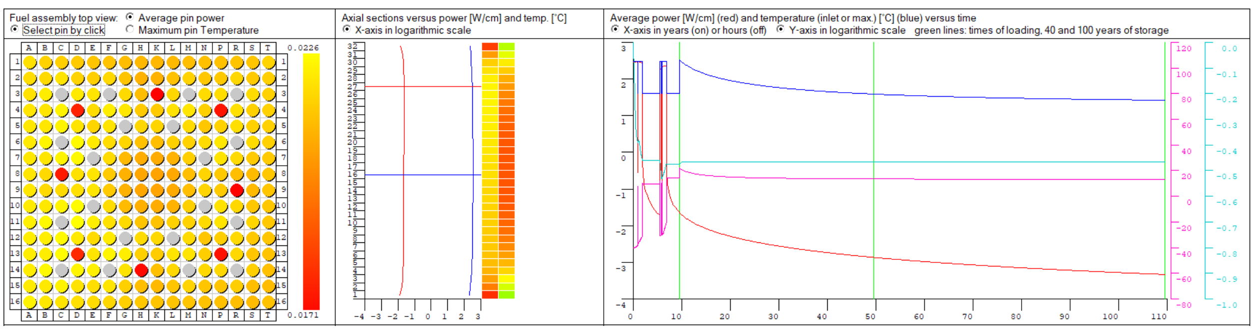

Figure 3. Modelled life and calculation results of a fuel assembly from first reactor cycle, wet storage in the on-site pool and cask dry storage: The pictures in the left and in the middle are identical with figure 2. In the right picture, the hoop stress (pink) and hoop strain (blue turquoise) vs. time shows the result for one rod of a complete calculation of all rods in the fuel assembly together with the input data temperature (blue) and input data power (red).

One may have noticed that we use the same graphical interface to display the input data and the calculated output data of our prototype. This way, on the one side, it is easier to find errors in the input data, and on the other side, it is easier for the user to see effects of the input on the output. Figure 2 and figure 3 show only the graphical output part of the user interface, where the user can make his selections, start the input generation and read the result files for displaying and further processing.

To summarise our initial results, in all our calculations for one cask we selected for our prototype implementation, we did not find any results, which exceed the maximum values for the tangential stress and tangential strain as limited by the license.

FURTHER ACTIVITIES

Following our successful implementation of this prototype, we plan some further activities to develop it to an useful tool for assessing future cask loadings and the parameter to be assessed for a prolonged storage time. This includes the implementation of data exchange formats to use temperature data and decay power from other codes as described in the DATA SOURCES chapter. The user interface will be extended to a more comprehensive and user friendly selection of the output data from the TRANSURANUS calculation. Beside the possibility to select the output in a wider range in this application, the TRANSURANUS input file of has to be extended to transfer this information, and the TRANSURANUS output scheme of has to be adapted accordingly. This includes also the type of results such as only maximum, average or/and minimum values with the coordinates of time and axial section. Furthermore, a set of selected input parameters for a “conservative calculation” is planned for the implementation. This corresponds to the supervisory approach in real cask loadings, which includes the implementation of the creep correlation used in the current licensing processes for dry storage. It will also help to limit the effort as warranted by the safety significance.

The TRANSURANUS code has an option for a probabilistically calculation. This feature may be added to the code input generation. Alternatively, we can also implement our method for probabilistic assessments that we used for the evaluation of the core design. This decision is still open.

Another extension of the TRANSURANUS code is the further development of our model for the solving, dissolving and reorientation of hydrides in the cladding. Finally, we want to implement appropriate colour schemes for the displayed data to achieve the best visibility of the displayed graphs. Last but not least, we have to write a manual for the future code user.

CONCLUSION

Within this paper we have provided a brief description of our prototype development for assessing fuel performance during dry storage in casks. We have addressed the main work areas, the established workflow, the data sources and their alternatives. A main point of our work, besides the overall data management, is the visualisation of the data and the creation of an interactive tool for the user. We already followed this approach in the development of the full core LOCA analysis. It was a real facilitation in data management, creation of input files and post processing of the results for later analysis.

We use the TRANSURANUS fuel performance code, which we are well familiar with. Nevertheless, other fuel performance codes may be useful in the same way. We use the data, which we have collected in our database TITANIA. This database is somehow unique due to the comprehensive data included in it. However, the stored data there is still available in the fuel documentation. Therefore, others can use the data if they have access to this documentation.

The paper briefly shows what we have done so far and addresses what still needs to be done to make the system fully operational in the further activities. When starting with the development of this prototype one goal was to establish a workflow to assess the fuel performance under dry conditions in casks. We think this became all the more important since the knowledge in nuclear issues is not growing in times of the phase-out in Germany. Especially the extension of the storage time from today 40 years up to 100 years will require generations of scientists and engineers. Perhaps this work will contribute to transfer some of our know-why and how we have come to our results today.

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