International Conference
Nuclear Energy for New Europe

Uncertainty of the Fuel Assembly Burnup in Siemens/KWU PWRs and its Usage in Safety Analyses

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ABSTRACT

Even though fuel assembly burnup itself is not a safety relevant parameter, it is strongly correlated to several other parameters that are safety relevant. Accordingly, fuel burnup can be used to describe a licensing condition and is suitable for defining design limits. The fuel assembly burnup values serve as state descriptor for a comprehensive characterization of the properties of a particular fuel type (e.g. Uranium Oxide/ Mixed Oxide) and enrichment, e.g. in terms of reactivity, validity of safety analyses, or as a starting point for decay heat or source term calculations. For TÜV NORD EnSys GmbH & Co. KG as a technical expert organization that monitors the safe operation of nuclear facilities as a government contractor, knowledge of the burnup monitoring method and its uncertainties is therefore an important issue. This knowledge is needed to assess whether the safety margins derived from the safety analyses are still sufficient. Best Estimate Plus Uncertainty (BEPU) approaches becoming more commonly used in the recent years instead of conservative approaches to improve the economical utilization while still complying with all safety requirements.

An important factor for the determination of the fuel burnup, and thus for its uncertainties, is the power density distribution. This presentation gives an overview of the method for determining the power density distribution in the 1300 MW Siemens/KWU built PWRs based on the results of the Aeroball Measurement System, and discusses the determination of uncertainties by comparing measured and calculated data. Additionally, a possible BEPU approach for verifying safety related values of a transport and storage cask loading based on the average fuel assembly burnup values is outlined. A statistical analysis shows that the safety-related parameters correlating with the fuel assembly burnup comply with the maximum and minimum values specified in the licensing procedure.

1 INTRODUCTION

TÜV NORD EnSys GmbH & Co. KG (TNE) is a technical expert organization that monitors the safe operation of nuclear installations as a state contractor. This includes the assessment of operator requests regarding fuel handling, transportation, and storage as well as the execution of the corresponding actions. From a safety point of view, it must be shown in the application documents that all safety parameters are fully met even with the assumption of conservative fuel properties. Even though fuel assembly burnup (FA) itself is not a safety relevant parameter, it is, together with the irradiation and the decay times, a key parameter for the determination of safety-relevant properties for a fuel of a given type and enrichment. Accordingly, the knowledge of the method of the burnup monitoring and its uncertainty is essential for the assessment of the safety parameters.

The fuel burnup and thus its uncertainty is defined by the heavy metal mass in the fuel assembly and the integral over its local power density during its lifetime. Secion2 of this paper gives a brief description of the method used to determine the power density distribution based on Aeroball Measurement System (AMS) measurement results used in Siemens/KWU built PWRs. Section 3 describes the derivation of the uncertainties by comparing the measurement results and the calculated values. Section 4 presents a possible BEPU approach for the consideration of the fuel burnup uncertainty in cask loadings based on a statistical random sampling method.

2 POWER DENSITY MONITORING IN SIEMENS/KWU BUILT PWR BASED ON AEROBALL MEASUREMENT RESULTS

In Siemens/KWU built PWRs, the neutron flux distribution is measured by the incore Aeroball Measurement System. This system comprises probes which hold the aeroballs, and the detector table. The probes consist of inner and outer tubes, which are inserted from above into a control rod guide tube of an FA that holds no control rod assembly cluster. The inner and outer tubes are connected by a gas passage which is permeable for the carrier gas (N_2), but not for the aeroballs. The 1300 MW_{el} PWRs hold 28 aeroball probes for a core of 193 FAs.

The flux mapping with the AMS is performed only on demand. Between the measurements, the aeroballs, steel balls of 1.7 mm (≈ 0.067 inch) diameter, are positioned at a resting position outside the core. When an aeroball measurement is performed, all aeroball stacks are blown into the AMS probes in the reactor core simultaneously, where the Vanadium of the steel is activated by the thermal neutron flux. The aeroball stack in the core reaches from a few centimetres below to a few centimetres above the active core height. After an activation time of normally three minutes, the aeroballs are then blown out of the core and onto the AMS detector table, which is located within the containment. There, the gamma activity of the Vanadium decay is measured. Figure 1 shows a schematic view of the AMS and its main components and figure 2 shows the distribution of the aeroball measurement probes in a 1300 MW_{el} Siemens/KWU built PWR.

The relevant reactions for the AMS activation and measurement are

$$^{51}V \xrightarrow{(n,\gamma)} {}^{52}V \xrightarrow[\text{}]{\overset{(\beta^-,\gamma)}{Half-life\ 3.7\ Min}} {}^{52}Cr \tag{1}$$

In most plants, the measurement is performed with 32 semiconductor detectors, which are positioned equally spaced over the ball stack length and thus represent 32 axial layers of the corresponding fuel assemblies. The measured activation values are automatically corrected for effects as in-scattering from neighbouring segments, decay during the measurement, residual activity of the aeroballs, impurity activities etc. The resulting activation values can then be directly compared to calculated ones. There are three calculation paths whose activation values are compared to the measured values: The first one is the ‘core design path’ which is calculated with predicted core states and rather wide calculation steps of several EFPD. The second is the ‘core follow path’ calculated by the local online core monitoring system using the respective actual plant parameters and small burnup steps of typically ≈ 1 EFPH.

These comparisons allow a direct control of the calculation quality and the impact of the deviation of predicted operational parameters to the actual ones. A colour scheme of such a comparison is illustrated in Figure 2.

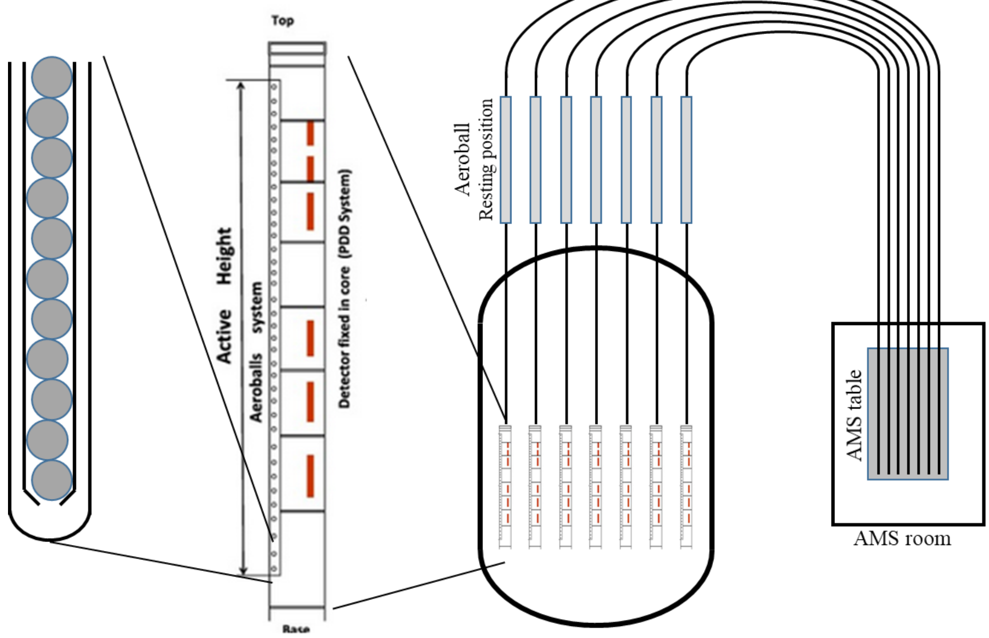

Figure 1: Schematic view of the AMS components. Left: Aeroball guide tube with ball stop (permeable to carrier gas). Center: Schematic view of a fuel assembly with AMS probe. Right: Total view of the AMS

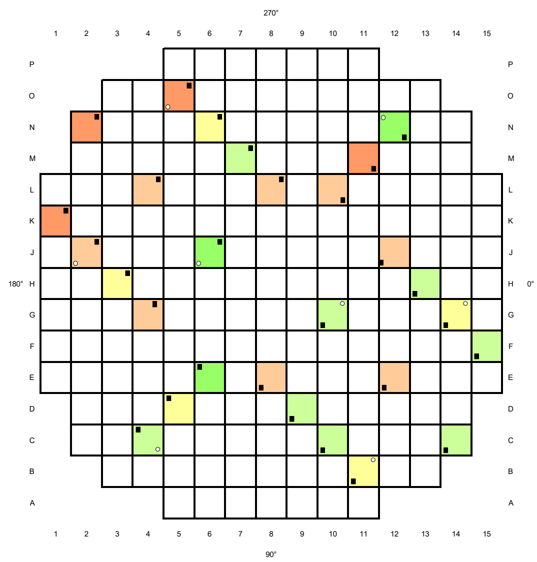

Figure 2: Distribution of the AMS probes in a Siemens/KWU built 1300 MW PWR

The distribution of the normalized activation values is used to generate an adapted core-wide flux solution and adaptation factors that are used to transform the calculated flux distribution to the measured one [1]. These adaptation factors can then be applied to the subsequent core follow calculations in the third calculation path. In this path, the calculations are performed with short time steps of ≈ 1 EFPH taking into account the actual plant parameters. The adaptation factors are used to ensure that the deviations between the measurement and the calculation are carried over to the following calculations. This calculation path is named 'measured path' and the resulting burnup values are called 'measured values', even though they are not actually measured.

3 DETERMINATION OF BURNUP UNCERTAINTIES BY COMPARISON OF MEASURED AND CALCULATED VALUES

The uncertainty of the fuel assembly burnup is determined by the uncertainty of the heavy metal mass per FA, which is not discussed in this paper, and the uncertainty of the released energy of the fuel assembly, which is the integral of the local power over time.

The uncertainty of the fuel assembly burnup in the Siemens/KWU concept is determined by the comparison of the burnup values based on the aeroball measurement results from the 'measurement path' to the pure calculation values from 'core design path' (calculation). This concept is based on the fact that both the core design and the power density reconstruction of the core monitoring system on site are calculated with verified and validated code systems. Accordingly, the true burnup value will endup being in most cases between the measured and the calculated values. The advantage of this statistical approach is that it delivers the total uncertainty of the power density calculation that includes the uncertainties of the measurement with the AMS, the uncertainties of the input parameters and the uncertainties of the code systems. Figure 3 shows the distributions of the differences (calculation minus measurement) for different fuel types and burnup intervals. As can be seen, the shapes of these distributions are in good accordance with a Gaussian distribution. The corresponding uncertainty is then

$$U_{PDD} = |\mu| + k^*\sigma \quad (2)$$

Here μ is the average deviation between measurement and calculation, σ is its standard deviation and k is a factor for the required confidence level depending on the process in which the burnup value is used. For example, the German regulations from the Nuclear Safety Standards Commission (KTA), KTA 3301 [2] and KTA 3303 [3], demand k=2 for accident analyses and k=1 for all other cases.

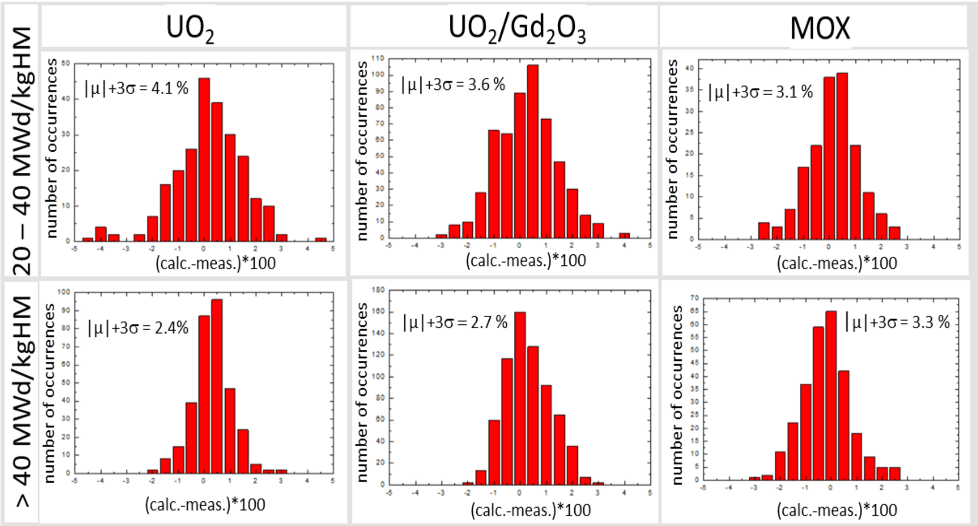

Figure3: Distribution of the differences calculation–measurement of the fuel burnups for different fuel types and burnup intervals

For the assessment of safety issues, it is often advantageous or necessary to derive individual burnup uncertainties for different burnup intervals, fuel types (U/U-Gd/MOX) or initial enrichments. A prerequisite for this is that the defined subgroups contain a sufficient quantity of data points (calculation minus measurement) for this statistical approach.

4 SAFETY-RELATED VERIFICATION OF TRANSPORT AND STORAGE CASK LOADING

The safety-related framework for acceptable cask loadings has been defined in the safety-related requirements (acceptance criteria) in the Technical Acceptance Conditions (TA), in the Implementation Regulations for the Technical Acceptance Conditions (IRTA) and in the approval certificate for the transport and storage cask. Figure 4 shows the derivation of the limiting values. For each cask loading, it must be demonstrated that these acceptance criteria are fulfilled.

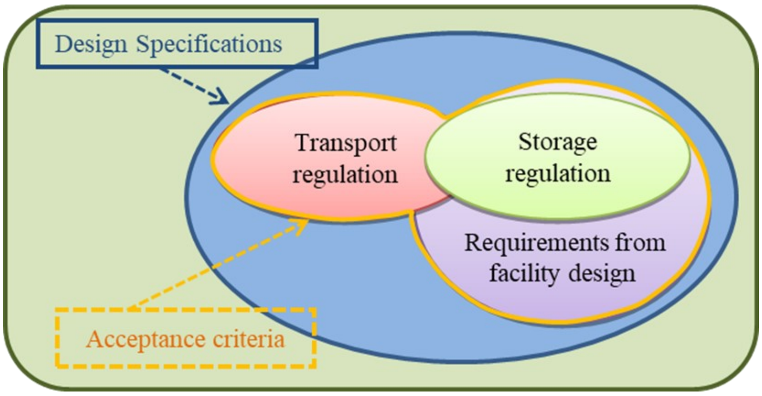

Figure 4: Derivation of the acceptance criteria from the regulation values.

In terms of thermal and shielding design, safety-relevant parameters are decay heat (DH), the activity inventories and the dose rates integral values for the whole cask loading (cask-related sum forms) or position-related individual values. These parameters are described by the factors DH_i for the FA-wise decay heat and the source term factors S_n^i and S_γ^i that are the FA-wise neutron- (NSS) and gamma source strengths (GSS) of the fuel relative to the reference source strengths for a given cask position i. With the totals $S_n = \sum_i S_n^i$ and $S_\gamma = \sum_i S_\gamma^i$ the corresponding acceptance criteria are:

- Compliance with the decay heat
  $DH_i \le P_{slot,\,i}$ for all FA and
  $P_{min} \le \Sigma_i(DHi) \le P_{max}$
  where P_{slot} is a loading position dependent allowed power level and P_{min}, P_{max} are decay heat limits for the complete cask.
- Compliance with design target dose rates at the cask surface with the key parameters
  Neutron Source Strength $S_n < \text{limit}_n$ and
  Total Source Strength
  $S_1 = k_\gamma * S_\gamma + k_n * S_n + k_4 * S_4 \le \text{limit}_s$
  where $S_4 = \sum_i S_4^i$ is the additional γ source strength of the activated FA structure materials.

Additionally, burnup-dependent parameters exist for the assessment of the long-term fuel integrity, but they will not be addressed in this paper.

The transport may only be performed if the acceptable limits are not exceeded, and the requirements are met.

It must be assessed whether all allowed values of the FA burnup dependent safety-relevant parameters that are specified in the licensing procedure are fulfilled. For this, the average FA burnups obtained using "best-estimate" methods and their uncertainties must be determined and considered in the verifications.

The average FA burnup is used as an input variable for the calculations of the neutron and gamma source strengths and the decay heat.

The uncertainty of the average FA burnup has a decisive influence on the uncertainty of the calculated values of DH, NSS and GSS.

For the analysis of the safety parameters for a cask loading, the uncertainties of the power density and the FA burnup must be determined on a plant-specific level. The data basis for these analyses in Siemens/KWU built PWRs are the aeroball measurements, which are regularly performed for the determination of the power density distribution and calibration of the incore instrumentation. The derived uncertainties can be differentiated by FA-types (UOX/U-Gd/MOX) and FA burnup intervals, as appropriate. This has been described in section 3 of this paper.

For the safety analysis, regulatory requirements demand that a one-sided tolerance limit *k* (see eq. (2)) must be determined for the uncertainty to be applied. The tolerance limit must be chosen so that the verification criterion is met with a probability of at least 95 % with a statistical confidence level of at least 95 %.

To date, the impact of the uncertainty of the fuel assembly burnup of the FA of a cask load is considered by a conservative approach where the uncertainties of the safety parameters due to the burnup uncertainties are simply added together. With increasing computing power and more refined methods, conservative approaches are more and more replaced by BEPU approaches to reduce the uncertainties for economic optimization. For TNE it is consequently necessary to be able to evaluate such advanced approaches. In the following, a possible approach using a statistical sampling method is outlined. For this example, the focus will be on the safety parameter decay heat of a cask loading.

A Monte Carlo method is used to realistically evaluate the decay heat of a cask loading. As the FA burnup uncertainty is assumed to have normal distribution, a Gaussian random number method was used.

In this two-step procedure, first Gaussian random numbers are generated and then normality tests are performed. The normality tests are based on two hypotheses: the null hypothesis, where the random numbers follow the normal distribution, and the alternative hypothesis, where the random numbers do not follow the normal distribution. If the null hypothesis is accepted at the 5% significance level through the normality tests, it is concluded that the random numbers are normally distributed. Otherwise, the random numbers are not normally distributed at the 5% significance level. The normality test techniques used are χ^2 test, Kolmogorov-Smirnov test, Shapiro-Wilk test, and D'Agostino-Pearson test.

The decay heat analyses are conducted based on the core operating parameters which are randomized using the presented random sampling method. Then the decay heat distribution is obtained. This decay heat distribution considers the effect of the operating parameters only. Finally, the uncertainties of the cross sections and the nuclide inventory code ORIGEN [5] are also considered to obtain the total decay heat distribution.

Since the total heat distribution is derived from finite samples, it has a sample mean $(\overline{\mathrm{M}_{TS}})$ and standard deviation (S_{TS}). To determine the decay heat limit, the population mean μ_T and the standard deviation (σ_T) with a (1-α) confidence level are estimated by

$$\mu_T = \overline{\mathrm{M}_{TS}} + t_{f,1-\alpha}\frac{S_{TS}}{\sqrt{N}} \tag{3}$$

$$\sigma_T^2 = \frac{f * S_{TS}^2}{\chi^2_{f,1-\alpha}} \tag{4}$$

where N is the number of random numbers and f is the degree of freedom (N-1) and $t_{f,1-\alpha}$ and $\chi^2_{f,1-\alpha}$ are the t-distribution and Chi-square distribution, respectively.

Taking into account the 95/95 criterion, the decay heat of a cask loading (DH_{limit}) is obtained by

$$DH_{limit} = \mu_T + k * \sigma_T \tag{5}$$

This approach enables TNE to assess the safety analyses of transport and storage cask loadings that are performed even with possible future Best Estimate Plus Uncertainty methods instead of the currently used conservative approaches.

5 CONCLUSIONS

The fuel assembly burnup is, though not directly limited from a reactor physics point of view, a main parameter for the determination of the properties of spent fuel of a given fuel type and enrichment. The burnup is thus a basic parameter for the calculation of safety-relevant fuel parameters such as decay heat, neutron- and gamma source strength, internal pressure, reactivity etc. and their uncertainties. The uncertainty of the FA burnup thus has a major impact on the uncertainties of these parameters, which must be considered for a conservative approach in safety studies or applications for fuel usage, handling or storage.

Consequently, the knowledge of the methods for burnup monitoring and the derivation of its uncertainty is mandatory for TÜV NORD EnSys GmbH & Co. KG as a technical expert organization that monitors the safe operation of nuclear facilities as a state contractor.

The main contribution to the FA burnup uncertainty comes from the uncertainty of the FA power density, which defines the released energy of the FA. This paper describes the concept of the Siemens/KWU built PWRs. In these plants, the power density distribution is calculated both by the core design codes with nominal conditions and by an online core monitoring. This considers the actual plant parameters, the deviations between the predicted core-wide neutron flux distribution, and the flux map measured by the AMS. The FA burnup uncertainty is then derived by comparing the values from the 'core design path' with those from the 'measured path'.

A possible way for the consideration of the error propagation to key safety parameters by a random sampling approach to reduce the uncertainties and thus improve the quality of the safety analyses is outlined. This allows TÜV NORD EnSys GmbH & Co. KG to keep pace with future developments in the methods used for safety analyses and verifications, retaining and further improving its ability to assess the safety of such advanced verification processes.

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