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IMPLEMENTATION OF A BURN-UP CREDIT APPROACH FOR TRANSPORT AND STORAGE CASK

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ABSTRACT

TN International currently uses burn-up credit methodology for the design of casks dedicated to the transport of PWR uranium oxide spent fuel assemblies.

As long as the fuel enrichment of the PWR fuel assemblies was sufficiently low, a burn-up credit methodology based on the sole consideration of actinides and the use of a partial burn-up was satisfactory to cover the needs without necessity to design new casks.

Nevertheless, the continuous increase of the fuel enrichment during the last decade has led TN International to continue the investigations on the burn-up credit methodology in order to limit both the increase of the neutron poison content in the new basket designs and the burn-up constraints attached to the acceptability of the fuel assemblies for transport.

The strategy of TN International was then to take benefit of the large anti-reactivity reserves, which might be gained by the consideration of the fission products coming from the fuel irradiation.

A big step forward has recently been reached by TN International on this field with the definition of an advanced burn-up credit methodology based on the consideration of relevant fission products recommended by OECD.

In the meantime, TN International has taken the opportunity to use such burn-up credit approach in the design of the TN 24 E transport and storage cask developed for the German Nuclear Power Plants. The relevant task has been done according to the German standard DIN 25712 for burn-up credit application.

The paper will describe the basic principles of the burn-up credit methodology implemented by TN International such as:

- the current state of the art concerning the burn-up credit application in the criticality assessment,
- the basic approach used for the implementation of the advanced burn-up credit methodology (bounding axial burn-up profiles, fuel irradiation parameters, fission products, …),
- the area of validity of the TN International burn-up credit approach with fission products,
- example of application of the burn-up credit methodology for the design of the TN 24 E transport and storage cask under licensing in Germany,
- the perspectives of development of the burn-up credit methodology.

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1. INTRODUCTION

The current criticality safety demonstrations of transport and storage cask designs dedicated to full burned spent fuel assemblies of commercial reactors are based on the consideration of the "fresh-fuel" assumption and results in a significant conservative overestimation of the system reactivity. This methodology was applied by TN International as long as:

- the fuel enrichment remained quite low,
- cask design development was possible from technical and economical point of view.

In the 1980's, a burn-up credit methodology, based on actinides only, has been implemented by TN International in order to face the increase of the enrichments on the PWR fuel assemblies. This approach allows the reduction of conservatisms in the criticality demonstration of the transport and storage cask designs and still contains sufficient reactivity reserve due to omission of fission products. This methodology, currently applied by TN International for existing transport and storage cask designs, has been copiously approved by many competent authorities.

The permanent increase of the PWR fuel enrichment during the last decades has shown the necessity to continue the progress on burn-up credit application reducing conservatism in the approach and adopting assumptions more adequate with neutron-physical effects (modelling of the axial burn-up profiles, consideration of fission products…).

The large experimental program led in France in the 1990's by CEA, IRSN and AREAVA NC on validation of calculation systems for burn-up credit application enabled recently the implementation by TN International of an advanced burn-up credit approach based on actinides and fission products for criticality assessment of the transport and storage cask designs.

This big step forward has been dictated by:

- the needs to introduce higher enriched fuel in the existing transport and storage cask designs,
- increasing cask performances over current design performances in order to offer adequate cask for the Nuclear Power Plants,
- reducing conservative assumptions in the criticality assessment while still maintaining an acceptable safety margin,
- extended capacities offered by the computer code systems.

New identified needs of transport and storage cask developments for Germany in the 2000's were the adequate opportunity for TN International to implement the advanced burn-up credit methodology on a dual purpose transport and storage cask dedicated to PWR spent fuel assemblies.

The advantages obtained with this new burn-up credit methodology offer the possibility for TN International to increase the cask design performances, while keeping adequate safety margin.

From this experience, TN International expects to extend the methodology on mixed oxide fuel assemblies and more complex BWR fuel assembly designs. Investigations are on-going in co-operation with French partners (AREVA NC, EDF, CEA, and IRSN).

2. CURRENT BURN-UP CREDIT METHODOLOGY

Actinide burn-up credit methodology is currently used on the basis of the sole consideration of the profit due to decrease in reactivity lead by ^{235}U depletion and ^{238}U depletion with production of actinides (plutonium isotopes) during irradiation of the uranium oxide PWR fuel assemblies.

In other words, only actinides are considered as a result of the irradiation of the fuel assemblies within the core, without any consideration of important anti-reactivity reserves from the fission products.

TN International still implements such burn-up credit approach for the transport and storage cask development using 8 actinide isotopes among ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu.

This methodology use the conservative assumption of an uniform axial burn-up profile (see Figure 1) with a magnitude equal to the average burn-up guaranteed on the top and bottom 50 cm of the fuel active part extremities.

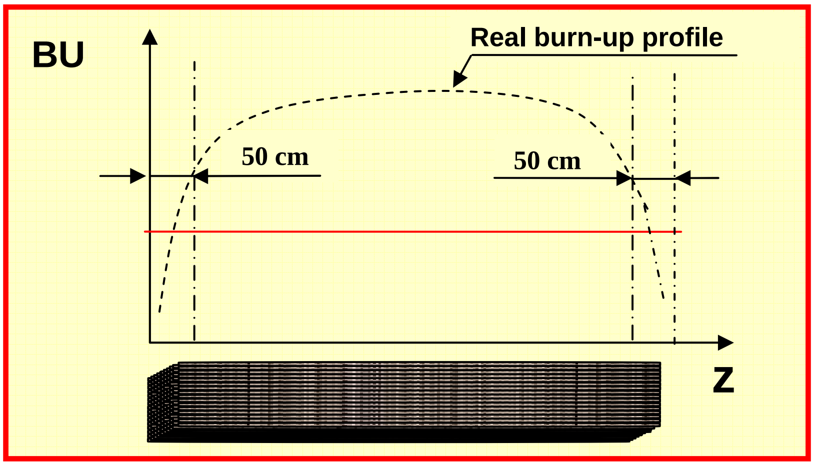

Figure 1: Uniform axial burn-up profile

On this basis, the burn-up verification procedure depends on the magnitude of burn-up (BU) guaranteed on the top and bottom 50 cm of the fuel active part as follow:

- BU ≤ 3 200 MWd/tU → Qualitative burn-up verification (irradiation check),
- BU > 3 200 MWd/tU → Quantitative burn-up verification (burn-up measurement).

This methodology has been used for more than 20 years by TN International on the TN12, TN13 and TN17 transport cask family used for the transportation of PWR uranium oxide spent fuel assemblies from the European reactors to the nuclear reprocessing plant of AREVA NC in La Hague (France).

Nevertheless, this methodology is quite limited today as it does not allow to take profit from the full anti-reactivity available with the fission products and the axial burn-up profile.

3. ADVANCED TN INTERNATIONAL BURN-UP CREDIT METHODOLOGY

The advanced actinides plus fission products burn-up credit methodology is based on the consideration of the irradiation of the uranium oxide PWR fuel assemblies and the associated change in the isotopic composition of the fuel, that means anti-reactivity profit led by:

- ^{235}U depletion (reduction of fissile isotopes quantity),
- ^{238}U depletion with production of actinides (even plutonium isotopes are high neutron absorbers for criticality),
- Production of high efficient neutron absorbers generated by fission of ^{235}U and plutonium (fission products).

According to the OECD recommendations [1], the actinide and fission product isotopes which can be selected for the definition of the isotopic composition of the fuel assemblies by consideration of the burn-up credit are:

- 9 actinides: ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am
- 15 fission products: ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{153}Eu, ^{155}Gd.

The selection of the fission products above obeys to the following rules:

- exclusion of gaseous and volatile fission products,
- selection of stable fission products,
- selection of variable fission products for which the daughter products have a higher absorption cross-section than the father product.

This methodology uses also a more realistic axial burn-up profile (see Figure 2) based on a bounding axial burn-up profile derived from sets of end of cycle real burn-up profiles from PWR uranium oxide fuel assemblies.

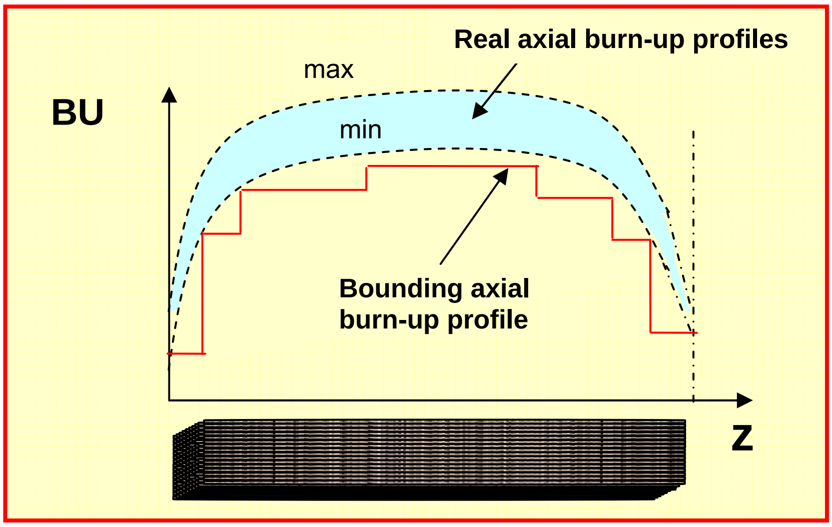

Figure 2: Bounding axial burn-up profile

As this methodology takes credit for a more realistic axial burn-up profile compared to the above mentioned current burn-up credit approach, fuel irradiation parameters need to be managed with precautions and must be justified so that to always ensure the conservatism of the approach.

4. APPLICATION TO THE TN 24 E TRANSPORT AND STORAGE CASK

The new advanced burn-up credit methodology has been implemented recently by TN International on the TN 24 E transport and storage cask dedicated to PWR uranium oxide (UO_2) and mixed oxide (MOX) fuel assemblies coming from the German Nuclear Power Plants. This application constitutes a world premiere for transport and storage cask.

Design performances of the TN 24 E transport and storage cask

The TN 24 E transport and storage cask design has been developed by TN International for the needs of the German Nuclear Power Plants.

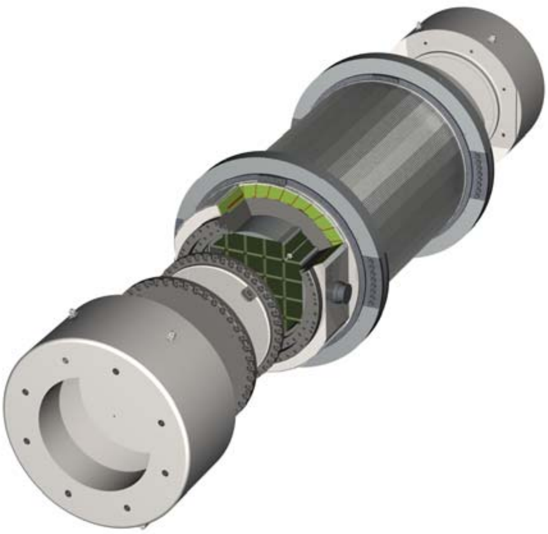

Figure 3: TN 24 E transport and storage cask design

It allows the loading of up to 21 PWR uranium oxide (UO_2) and mixed oxide (MOX) fuel assemblies in a dedicated basket using boron carbide compound technology to ensure the content sub-criticality.

The TN 24 E package model is designed to receive up to 21 UO_2 PWR fuel assemblies with maximum initial ^{235}U enrichment of up to 4.65 wt-%.

The advanced burn-up credit methodology has been implemented only for UO_2 PWR fuel assemblies whereas the conservative "fresh-fuel" assumption has been still used for MOX fuel assemblies.

Safety standards

The criticality safety demonstration for the TN 24 E cask design is based on:

- the IAEA regulations [8] for the safe transport of radioactive material associated to type B(U) packages containing fissile material,
- the criticality safety related German standard DIN 25712 [9] relative to application of burn-up credit in the criticality safety analysis of *transport and dry storage* of spent nuclear fuel,

Calculation procedure and code validation

The calculation procedure (see Figure 4) used by TN International for the TN 24 E burn-up credit criticality calculations is based on:

- DARWIN 2.0 [2] depletion code developed by CEA and EDF in cooperation with AREVA NP,
- CRISTAL V1.0 [3] criticality calculation code developed by CEA and IRSN in cooperation with AREVA NC, AREVA NP and EDF.

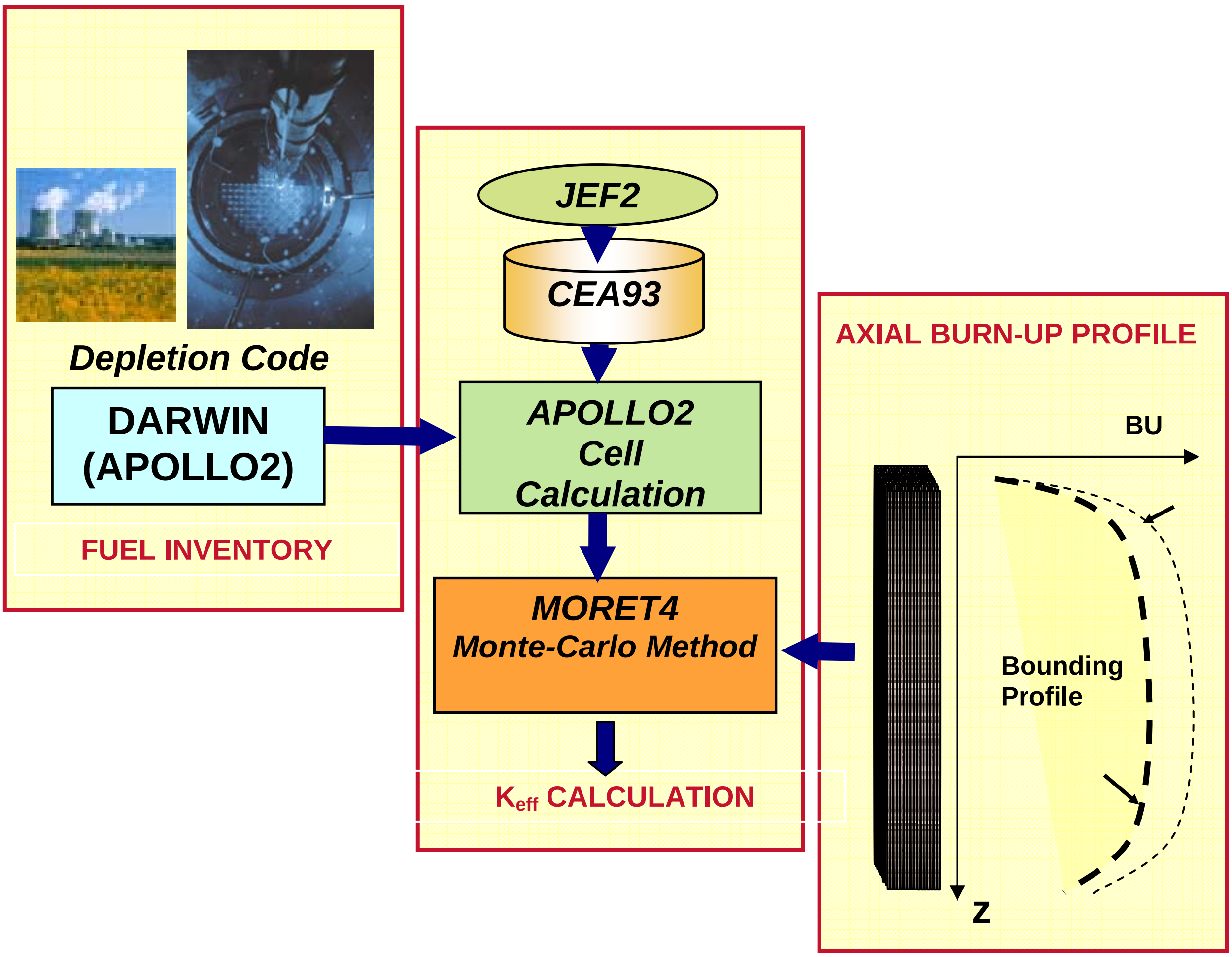

Figure 4: TN 24 E calculation process

DARWIN depletion code

In order to validate the DARWIN system for spent fuel inventory calculations, a large experimental program based on spent fuel has been performed in France. The experimental data are based on chemical analysis measurements performed on fuel rods irradiated in French PWR reactors and on fuel assembly dissolutions at the AREVA NC/La Hague nuclear reprocessing plant. Uranium, plutonium, americium and fission product isotopes were analysed in PWR samples.

The validation program [12] has shown the capability of the DARWIN package to simulate the fuel inventory versus burn-up and covers a large range of UO_2 fuel assemblies with various ^{235}U enrichments lower than 5 wt-% associated with burn-up from 10 to 60 GWd/tU.

According to the validation of the DARWIN system on French experiments for the fission products capture cross-section validation [13], corrections factors are considered for the fuel inventory calculations by maximising the fissile isotopes content and minimising the high capturing isotopes production. For the case of absorbing fission products for which the concentration is underestimated by the DARWIN code, no correction factor is considered on the relevant nuclides in a conservative way (the factor is considered equal to 1).

The influence of the correction factors on the package reactivity has been evaluated on each TN 24 E criticality relevant configurations and deducted from the k_{eff} acceptance criterion.

The DARWIN system used for fuel inventory of the UO_2 fuel assemblies complies with the requirements of the KTA 3101.2 standard [14] as required by the DIN 25712 standard.

The validation program of the DARWIN system covers a large range of UO_2 fuels with various ^{235}U enrichments up to 4.5 wt-% associated with burn-up up from 10 GWd/tU to 60 GWd/tU and, therefore, covers the area of application of the TN 24 E cask.

CRISTAL V1 code package

The criticality safety calculations are performed with the “standard route” (see Figure 4) of the CRISTAL V1 code package based on the JEF2.2 nuclear data [4].

The “standard route” deals with the 172-group nuclear data library CEA93 (derived from JEF2.2 evaluation), the APOLLO2 cell code [5] used for self-shielding, flux calculations, cross-section calculations, and homogenisation, and the MORET4 Monte Carlo code [6].

The calculation procedure used for the criticality assessment contains the following stages:

- definition of the cross-sections of the fissile medium representative of the fuel assembly and of structural components of the packaging with the APOLLO2 code,
- calculation of k_{eff} of the package with the Monte-Carlo MORET4 code on the basis of the cross-sections above,
- comparison of the calculated k_{eff} with the k_{eff} acceptance criterion.

The CRISTAL V1 system has been validated on a large database of benchmark experiments (about 1900) [7], which includes all the different kinds of configurations encountered in the industrial nuclear fuel cycle and burn-up credit configurations.

The qualification of the fission products cross-sections is based on (see [10]):

- Oscillation experiments performed by CEA in the MINERVE reactor (Cadarache – France),
- Fission products criticality experiments performed by IRSN (Valduc – France).

The approach for establishing sub-criticality in the criticality safety analysis uses the definition of a statistical “Upper Sub-critical Limit” (USL) as described in NUREG/CR-6361 [11] and is in agreement with the German standard DIN 25712. USL calculations have been performed against benchmark experiments representative of the TN 24 E package.

Definition of the axial bounding burn-up profiles for UO_2 fuel assemblies

The bounding axial burn-up profiles of UO_2 fuel assemblies implemented for the TN 24 E application are defined according to:

- the methodology presented at the ICNC conference [15],
- the observations [16] and [17] of phase II benchmark experiments conducted by the Expert Group on Burn-up Credit Criticality Safety,
- a compilation of a large reactor database containing axial burn-up profiles for all the population of fuel assemblies from the German Nuclear Power Plants,
- the requirements of the German standard DIN 25712.

The methodology described in [15] to [17] enables to generate a bounding axial burn-up profile (see Figure 5) resulting in the highest “end effect” of reactivity as a continuous function of the average burn-up of the UO_2 fuel assemblies.

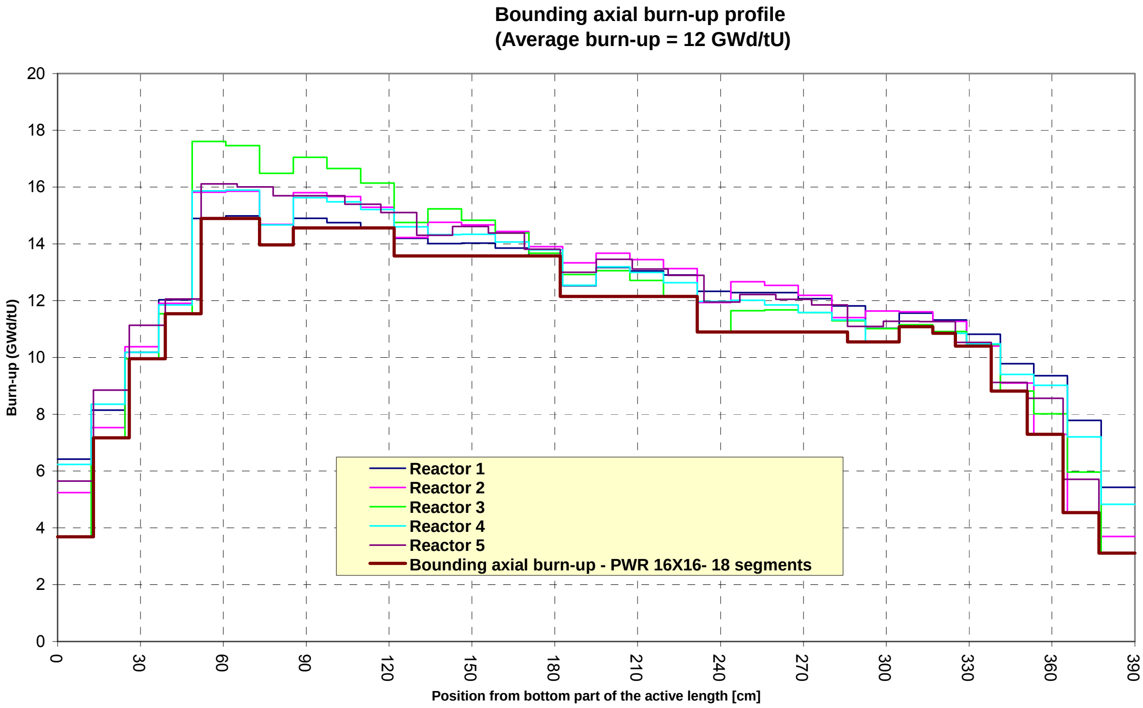

Figure 5: PWR bounding axial burn-up profile

It can be pointed out that the methodology used for the definition of the bounding axial burn-up profiles from Figure 5 is reduced from 32 to 18 burn-up steps without any significant impact on the reactivity.

Fuel assembly parameters for depletion calculations

The parameters relevant for depletion calculation are: specific power, fuel temperature, moderator temperature, moderator density, boron concentration in the moderator, irradiation history, fuel environment during irradiation, solid absorbers in the assembly grids, integral burnable absorbers (part of the fresh-fuel) and cooling time.

All fuel parameters retained for depletion calculations of PWR UO_2 fuel assemblies are selected in order to generate neutron spectrum hardening, by comparison with the nominal reactor parameter, and then lead to a conservative contribution for the criticality analysis.

Criticality calculation model and results

Figure 6 gives an overview of the longitudinal section of the TN 24 E criticality calculation model resulting from the CRISTAL V1.0 package.

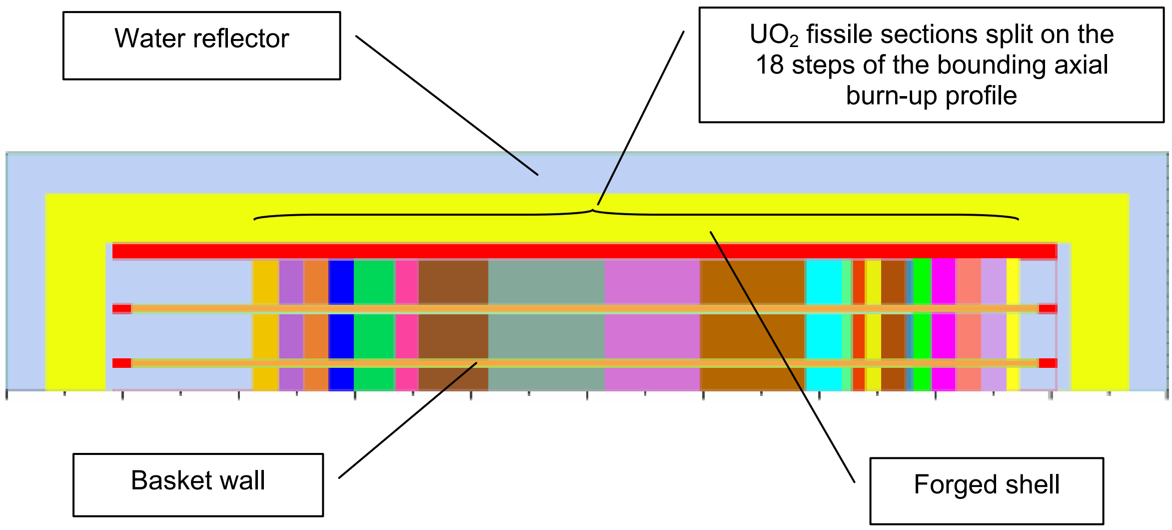

Figure 6: TN 24 E criticality calculation model

Uncertainties due to the manufacturing tolerances (cask, basket and fuel assemblies) are incorporated in the calculation model in order to find the maximum k_{eff} value.

The containment system of the TN 24 E package is closely reflected by 20 cm of water and all void spaces of the package cavity are filled with water.

Table 1 point out two advantages:

– the reactivity gain obtained respectively by the current burn-up credit methodology (with actinides only) and the advanced burn-up credit methodology (with actinides + fission products), using identical burn-up (BU) requirement, by reference to the "fresh-fuel" assumption,

– the optimised minimum burn-up requirements, associated to the burn-up credit methodologies above, which allow the respect of the applicable keff acceptance criterion.

Loading constituted of 21 UO_2 PWR (^{235}U = 4.65 wt-%)	**Current burn-up credit approach (Actinides only)**	**Advanced burn-up credit approach (Actinides + 15 FP)**
Reactivity gain compared to "fresh fuel" [%]	≈ 3% (BU = 12 000 MWd/tU)	3% + 3% ≈ 6% (BU = 12 000 MWd/tU)
Minimum burn-up requirement [MWd/tU]	≈ 23 000	≈ 12 000

Table 1: Comparison of burn-up credit approaches

In addition, it can be pointed out that only six fission products (including ^{149}Sm) provide 75 % of the reactivity gain led by the fifteen fission products.

The advanced burn-up credit methodology is a big step forward in comparison to the progress made in 1980’s by TN International using current burn-up credit methodology instead of “fresh-fuel” assumption for transport cask development.

In other words, the advanced burn-up credit methodology is an appropriate alternative for compensating PWR fuel enrichment increase and limiting the poison content inside the basket materials in the most equilibrated economical and safety environment.

5. CONCLUSIONS

The new advanced burn-up credit methodology implemented by TN International, based on actinides and fission products consideration as well as on bounding axial burn-up profiles, allows to extend burn-up credit advantages to new transport and storage cask design developments dedicated to PWR fuel assemblies.

The new methodology allows an adequate level of criticality safety of the transport and storage cask designs whereas performances of the cask designs are continuously increased with high enriched PWR fuel assemblies.

This methodology, implemented for the TN 24 E transport and storage cask design dedicated to German nuclear power plants, gives evidence that extended burn-up credit methodology based on fission products is adequate for high enriched PWR uranium oxide fuel assemblies.

The calculation codes DARWIN 2.0 and CRISTAL V1.0 associated to the advanced burn-up credit methodology have been validated for burn-up credit application on a large experimental program led in France. The validation of the code system has been implemented against post-irradiation examination (PIE) data and critical benchmark experiments representative of the transport and storage cask configurations.

Taking profit of the feedback received by investigations led on different burn-up credit approaches, TN International is expecting new perspectives for the transport and cask design developments by:

- Extension of the advanced burn-up credit methodology to selected transport and storage package designs dedicated to UO_2 PWR spent fuel assemblies with the advantages of:
  - ✓ reduction of burn-up requirements,
  - ✓ reduction of basket poison content,
  - ✓ simplification of the burn-up verification procedure.
- Extension of the application area of the advanced burn-up credit methodology to:
  - ✓ MOX PWR fuel assembly,
  - ✓ UOX BWR fuel assembly.

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