Validation of CASCADE-3D for PWR MOX Core Design – An On-going Process

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ABSTRACT

At Framatome ANP, Germany, neutronic core design for pressurized water reactors (PWRs) is performed with the SAV95 part of the CASCADE-3D code system.

A major aspect of the development of SAV95 was an improved description of high plutonium content MOX fuel assemblies in recycling cores. MOX assemblies with a plutonium content as high as 7.6 w/o Pu_{tot} (5.1 w/o Pu_{fiss}) were designed and are used in cores with a MOX fuel assembly ratio of up to 35%. Reactor cycles with target exposures of ≈ 62.000 MWd/t for regular MOX fuel assembly reloads are currently in operation.

In a global market the neutron physical design codes have to provide the capability of predicting key safety parameters accurately for a wide range of application. Modern neutronic and mechanical design codes account for the relevant differences of the properties of uranium and MOX fuel and are capable to describe the behavior of the MOX fuel in the reactor with the same level of confidence as for uranium fuel. Apart from routinely performed comparisons with standard SAV95 design calculations, special measurements are used to verify the prediction of key safety parameters. Experiments in critical facilities are supplemented by special measurements in commercial power reactors. Control rod worth and γ–scans for determining the accuracy of the calculation of the power density distribution in MOX assemblies are part of the verification efforts for the Framatome ANP design codes. The Framatome ANP methods and methodologies are continuously developed and prove its applicability in the daily design work also for MOX fuel.

1. INTRODUCTION

The electricity generation with nuclear power is in competition with fossil energy sources. The decision on how to define the mix of energy sources in the individual countries is strongly driven by the accessibility of natural resources, but also by perceptions of the public opinion. Public acceptance is a prerequisite for a sustainable operation of nuclear facilities. A key role for forming public attitude plays confidence in the ability to operate nuclear power plants safely. High safety standards, reliability and flexibility are therefore main issues for the operation of nuclear power plants.

The optimum for an economically attractive operation of nuclear power plants is different in different operating and licensing environments. Thus, in a global market the neutron physical design codes have to provide the required accuracy for the prediction of key safety parameters for a wide range of application.

At Framatome ANP, Germany, neutronic core design for pressurized water reactors (PWRs) is performed with the SAV95 part of the CASCADE-3D (**C**ore **A**nalysis & **S**afety **C**odes for **A**dvanced **D**esign **E**valuation) [1, 2] code system (cf. Figure 1). SAV95 is a further development of the earlier standard design procedure versions SAV79[3] and SAV90[4].

In the past years, worldwide more than 400 cycles of 30 PWRs in 12 countries were designed with the SAV code systems, about 90 cycles with MOX assemblies. A continuous development of each version of the code system was necessary to meet the growing customer expectations. Extending the range of applicability had always been based on a high level of confidence in the prediction quality of the codes. Gaining the needed level of confidence requires continuous qualification efforts.

2. THE CASCADE-3D CODE SYSTEM

The CASCADE-3D coupled code system links code packages used at Framatome ANP for in-core fuel management and accident analysis for PWRs. The main components of the system are FOXS, PRISM/PINPOW, PANBOX/COBRA and RELAP5 (see Figure 1). Further elements of the CASCADE system are the automatic loading pattern optimisation tool PRIMO and the on-line core monitoring system POWERTRAX™.

In Fig. 1 the SAV95 part of the CASCADE–3D system is easily to be identified. Main components of SAV95 are the neutronic fuel assembly (FA) design tool FOXS, the reactor code PRISM, the evaluation tool PINPOWL, and PRIMO.

FOXS – with the 2-dimensional transport lattice code CASMO[5] as central module – is a versatile tool for neutronic fuel assembly design, including capabilities for the description of repaired or reconstituted FAs. The basic cross section library utilized in the system is well suited for present and future uranium and MOX fuel assemblies. FOXS provides data for FAs and reflectors to the reactor codes PRISM and PANBOX. These data comprise microscopic cross sections, assembly discontinuity factors and heterogeneous form functions; they result from single assembly multi-group transport calculations.

PRISM is a nodal 3-dimensional reactor burnup code based on a fast running steady-state flux solver. This state-of-the-art module combines the accuracy of advanced nodal methods with the efficiency gained by the application of multi-level acceleration techniques and the use of vectorization capabilities. It handles intra-nodal spatial variations of cross sections and uses discontinuity factors, which enables a more accurate treatment of FA asymmetries in a radial one node per assembly representation. PRISM allows for axially user defined non-equidistant node sizes.

Another important feature of PRISM is the continuous representation of the cross sections with respect to thermal-hydraulic parameters. This allows for only one cross section library for the description of steady-state reactor conditions from cold, zero power to full power operation.

For an accurate and efficient determination of pin powers, exposure, fluxes and detector signals, the full 3-dimensional interpolation and modulation scheme is directly integrated into the core simulator.

Based on the 3-dimensional pinwise solution provided by PRISM, the evaluation code PINPOWL determines key safety parameters, such as local departure from nucleate boiling ratios (DNBR) and pinwise waterside cladding corrosion data. Other central functionalities of PINPOWL are the generation of process computer data for various reactor types and the support of automated report generation procedures.

The automated loading pattern optimization tool PRIMO is based on evolutionary algorithms which have proved to be well suited and robust for complex engineering tasks. PRIMO is coupled to PRISM, the standard SAV95 design code. The PRISM calculations are the most time consuming part of the entire optimization process, but that way one avoids the main disadvantage of optimization methods using simplified models for the core simulations. The derived results are of the same quality as of standard design calculations. Reload expert knowledge in form of heuristic rules were implemented into PRIMO to significantly increase the search speed of the algorithms.

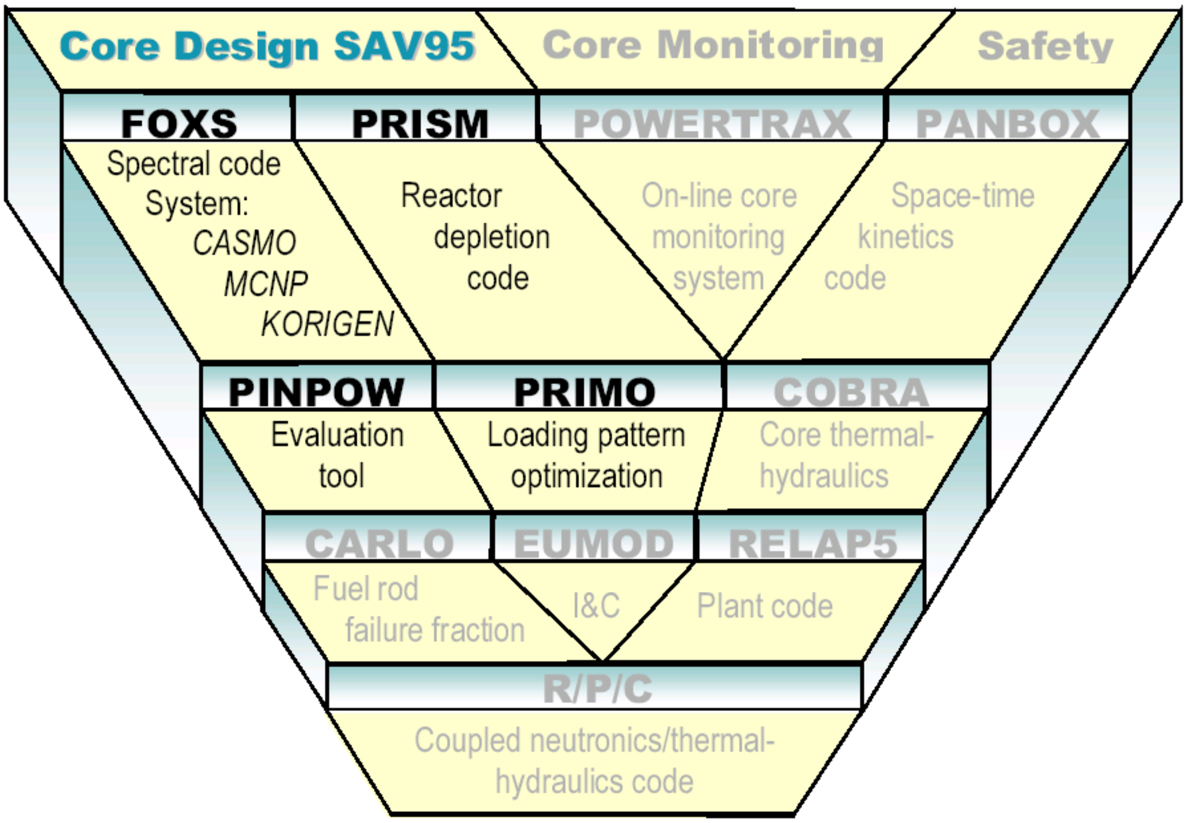

Figure 1. The SAV95 Part of the CASCADE-3D Code System

The consistency between in-core fuel management and safety analysis is achieved on one hand by using the same steady-state flux solver and pin power reconstruction modules in both reactor codes PRISM and PANBOX. On the other hand, both codes are directly provided by the same input information in terms of cross section libraries, fuel assembly form functions, as well as reactor data, geometry and fuel nuclide number densities.

3. VERFICATION AND VALIDATION OF CASCADE-3D

One major aspect of the development of SAV95 was an improved description of high plutonium content MOX fuel assemblies and recycling cores using those assemblies. MOX assemblies with an average plutonium content as high as 7.6 w/o Pu_{tot} (5.1 w/o Pu_{fiss}) were designed and are used in cores with a MOX fuel assembly ratio of up to 35%. Reactor cycles with target exposures of ≈ 62.000 MWd/t for regular MOX fuel assembly reloads are currently in operation[6].

Utilizing the potential of those advanced MOX fuel assemblies in hyper low-leakage core designs is highly demanding for material and design codes [7]. To meet the customer expectations, the tools have to provide the required flexibility, robustness and reliability. The SAV95 system is applied for 6-months up to 24-months cycles. The resulting diversity of core designs in different markets is a challenge for the capabilities of the design tools.

The SAV95 system is qualified by theoretical benchmarks and by comparison of the calculated results with measurements. Theoretical benchmarks give a picture of the quality of the applied mathematical models and its implementation, while comparisons to measurements demonstrate the 'real' prediction quality for the operational behavior of the plants.

Routinely performed startup and core follow measurements are continuously extending the database for the qualification of SAV95. As an example, Figures 2 and 3 show the comparison of measured and calculated detector signals only for MOX fuel assemblies and of boron equivalents of the control rod worth for cores with and without MOX assemblies.

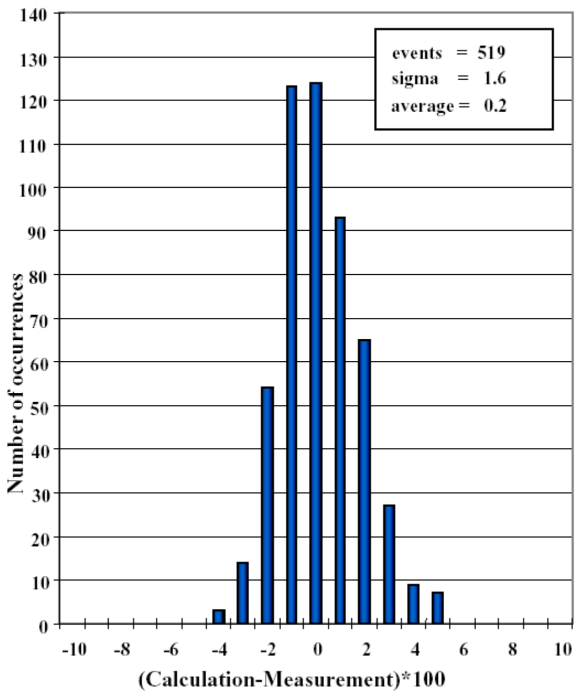

Fig. 2 Comparison of Calculated and Measured Detector Signals in MOX Fuel Assemblies

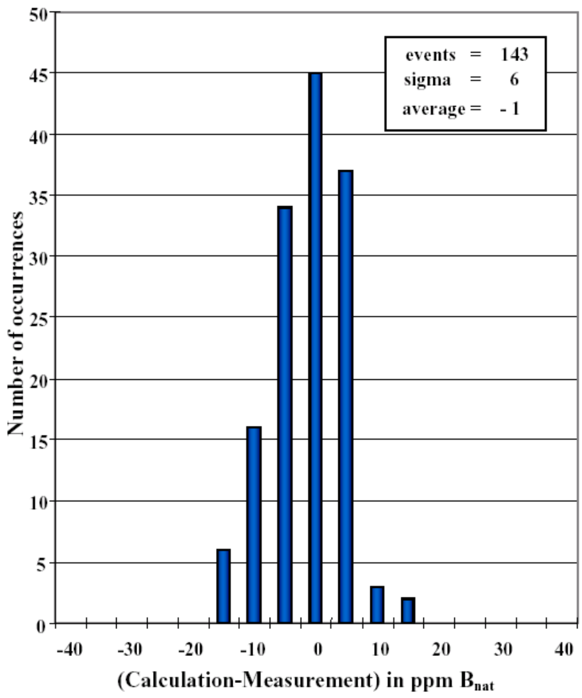

Fig. 3 Comparison of Calculated and Measured Control Rod Worth

Apart from those routinely performed comparisons with results from standard design calculations, special measurements have been used for the verification of specific aspects.

3.1. COMPARISON OF THE CONTROL ROD WORTH IN URANIUM AND MOX FUEL ASSEMBLIES

From early on, measurements were performed for the determination of the control rod worth in uranium and MOX fuel assemblies. An important aspect of these measurements was the dependence of the control rod worth on the coolant temperature. The experiments were performed at coolant temperatures from 20 °C to 245 °C. These measurements demonstrated a reduction of the control rod worth in fresh MOX assemblies compared to uranium assemblies by about 20% at high coolant temperatures. This difference is continuously reduced with decreasing moderator temperature. Both effects are accurately reproduced by the design methods.

Additional measurements were carried out in commercial power reactors for burnt fuel. One measurement campaign confirmed a reduction to 10 % after one irradiation cycle for the difference of the control rod worth in MOX and uranium fuel. Table I shows the result of a comparison for different control rod configurations at beginning of cycle, hot zero power conditions of a 2nd measurement campaign. The DI bank represents a group of 12 fully inserted control rods (out of 61) with a total boron equivalent of about 160 ppm B_{nat}. Those measurements are routinely performed during reactor start-up and its statistical evaluation is shown in Figure 3. The accuracy of the prediction of the control rod worth in uranium and MOX assemblies has to be seen in comparison to these results. For the special measurement a core design was chosen with 8 MOX assemblies placed on control rod positions. In parallel to that MOX bank, the control rod worth of a configuration with 8 uranium assemblies was measured. Each of the control rod groups had a boron equivalent of about 90 ppm B_{nat}. The comparison shows deviations well within the statistical uncertainty depicted in Figure 3 and the uncertainty of approximately ±10 ppm B_{nat} to be considered for this type of measurement. No difference in the prediction quality of the control rod worth for MOX and uranium fuel becomes obvious from these comparisons.

TABLE I. Comparison of Calculated and Measured Control Rod Worth (C - M) in Uranium and MOX Fuel Assemblies (Boron Equivalent at Beginning of Cycle, hot, zero power, Xe = 0)

Δc_B(DI Bank) in ppm B_{nat}	Δc_B(MOX Bank) in ppm B_{nat}	Δc_B(Uranium Bank) in ppm B_{nat}
3	1	-6

3.2. POST IRRADIATION EXAMINATIONS WITH ISOTOPIC MEASUREMENTS

The calculation of the burnup dependent isotopic composition of the fuel is verified by comparisons with results of isotopic analyses performed in the frame of post irradiation examinations. Those measurements were performed on uranium and modern MOX fuel up to exposures of ≈50 MWd/kg. Figure 4 shows the results of a comparison for MOX fuel with 2nd generation plutonium.

The initial fissile plutonium content of the investigated fuel was 3.2 w/o (≈ 6.7 w/o Pu_{tot}) in a natural uranium matrix. The plutonium quality at beginning of life was 58 w/o (Pu_{fiss}/Pu_{tot}). In the 4 irradiation periods the cycle averaged power density at the probe level was 220 – 260 W/cm. The measurement uncertainty stated for the main isotopes ^{235}U, ^{236}U and ^{239}Pu – ^{242}Pu was 3 % (relative) and for ^{238}Pu it was 10 % (relative). With the exception of $^{238}Pu/^{239}Pu$, for which the almost constant deviation with burnup lets the initial content provided by the laboratory appear questionable, the comparison shows an overall good agreement between calculation and measurement.

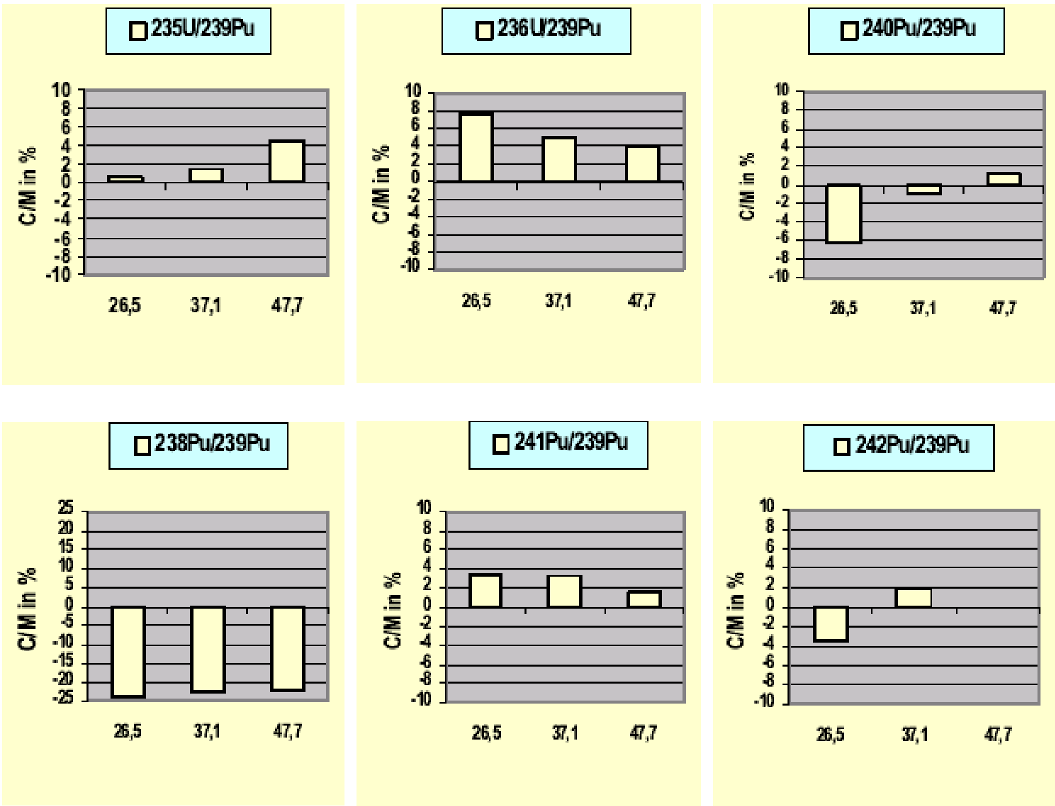

Figure 4. Comparison of Calculated and Measured Isotopic Inventory of MOX Fuel with 2nd Generation Plutonium in Dependence of Burnup (Relative Deviation)

Comparisons to other experiments, e.g. the ARIANE program [8], confirm the good agreement of the calculated and measured built up of neutron physically relevant nuclides. In the frame of that international program PWR MOX samples were measured up to an exposure of ≈ 59 MWd/kg.

Another aspect of such measurements, besides the verification of the isotopic built up, is the burnup achieved during irradiation. The comparison of the measured burnup, based on the determination of the ^{148}Nd concentration, is an indication also for the prediction quality of local powers. The exposures of the probes determined based on the neodymium concentration agree with the calculated values within the ± 3% uncertainty limits.

3.3. POWER DENSITY DISTRIBUTION MEASUREMENTS

Starting in the late 60s and early 70s, critical experiments were performed to verify the calculation method of fuel rod powers in MOX assemblies embedded in a uranium assembly environment.

Experiments in the KRITZ and VENUS critical facilities are used to qualify the SAV design procedures. For handling and safeguards reasons, these measurements were performed on fresh fuel.

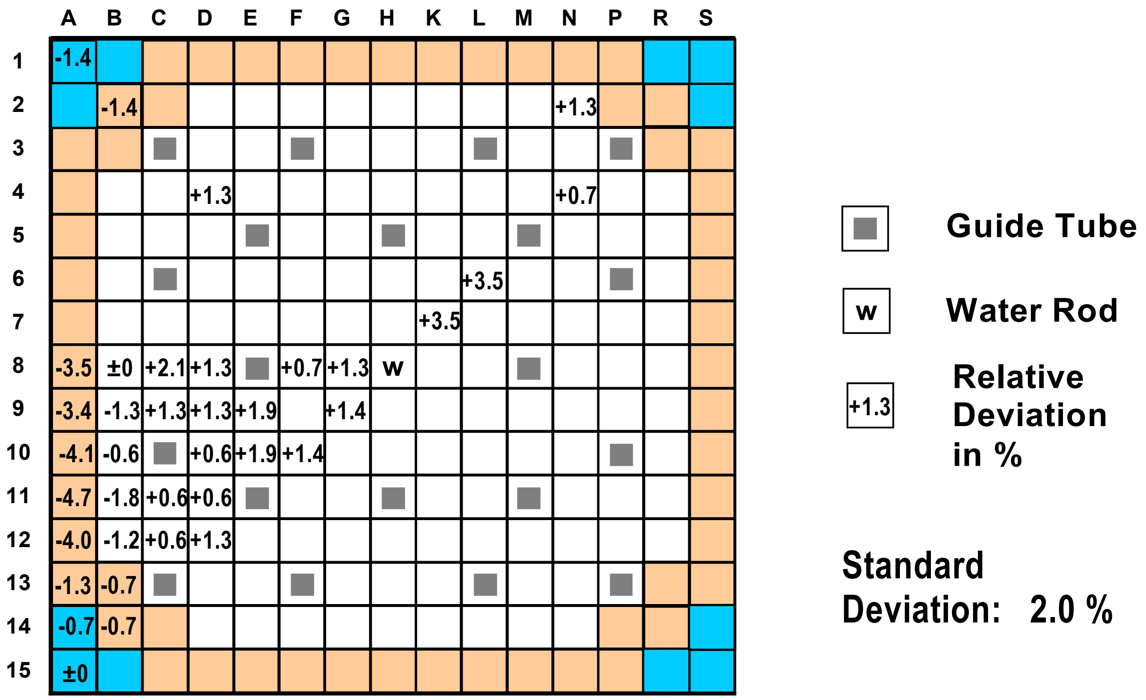

Figure 5. **γ-Scan on Irradiated MOX Fuel Rods**
Comparison of Calculated and Measured Power Distribution

For the verification of the prediction quality for fuel rod powers in burnt high plutonium content MOX assemblies γ-scan measurements have been conducted on irradiated MOX fuel assemblies during a regular refueling outage of the Goesgen power plant (KKGg). The fuel assembly average plutonium concentration was 7.1 w/o Pu (4.8 w/o Pu_{fiss}), i.e. the plutonium quality was about 67.5 w/o. Measured was the axial and radial power profile in the MOX assembly. The high burnup of the selected fuel assembly of 19.1 MWd/kg achieved at the end of an annual cycle indicates the high power level of the assembly during its 1st insertion period.

After an outage length of about 30 days, the fuel assembly was planned to be reinserted in the next cycle for its 2nd irradiation period. Therefore, the achievable decay time was rather short. This resulted in a high γ-background for the measurement. A lead absorber of 5 cm thickness had to be used as screening device for the collector to reduce the γ-intensity. This caused an overproportional reduction of the low energy photons and made the direct measurement of ^{140}Ba (γ-line at 537 keV, half life 12.7 d) with the required accuracy impossible. Under that circumstances only the ^{140}La γ-line at 1596 keV (half life ^{140}La: 40.3h, precursor isotope ^{140}Ba) was usable for the evaluation of the pin power distribution. After a decay time of about 20 days ^{140}Ba and ^{140}La are in equilibrium and the γ-line of ^{140}La is then direct proportional to the pin power. The measurement accuracy was determined to 1.5 % for the reproduction of the measurement and the overall uncertainty is estimated to ± 3% for the fuel rod average power. The γ-lines best suited for burnup equivalence (e.g. ^{144}Ce, ^{144}Pr) are low energetic and the resulting measurement uncertainty was too high to be evaluated. The interference with other nuclides prevented the use of the higher energetic lines.

Figure 5 demonstrates the good agreement of calculated and measured axially integrated power densities. The standard deviation is 2%. The calculated data are derived from standard SAV95 reactor calculations for the end of the cycle. The last part of the cycle was operated with a coast down period of 54 days down to 87 % rated power. This has only limited impact on the radial power distribution but results in a significant change of the axial power shape and has to be considered for the comparison of axial distributions. Each fuel rod was measured in 5 axial positions. The maximum standard deviation in a single measurement layer is 2.6 %.

4. CONCLUSIONS

The conclusion from these comparisons is that CASCADE-3D is capable to predict key safety parameters for actual recycling cores at the same level of accuracy as for uranium cores. The design methodologies used at Framatome ANP were developed to provide the margins required for a safe and flexible operation of the reactors independent of the type of fuel that is used. The methods and methodologies are continuously developed and prove its applicability in the actual design work also for MOX fuel. The high standard for predicting key safety parameters described here has to be maintained or even further improved for new developments in core design.

5. REFERENCES

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