THE COMPREHENSIVE METHODOLOGY FOR CHALLENGING BWR FUEL ASSEMBLY AND CORE DESIGN USED AT FRAMATOME ANP

St. Misu, H.D. Kiehlmann, H. Spierling, F. Wehle
Framatome ANP GmbH
P. O. 3220
D 91050 Erlangen, Germany
stefan.misu@framatome-anp.de

H. Moon, R.G. Grummer
Framatome ANP Inc.
2101 Horn Rapids Road
Richland, WA 99352-0130, USA

ABSTRACT

Safe, reliable and flexible reactor operation with optimal fuel utilization is an invariable utility requirement. Current market demands such as further increase of enrichment and burnup, flexible cycle length, power uprate, MOX and reprocessed uranium fuel (ERU) assembly insertion imply demanding fuel design conditions and, as a consequence, great challenges to the fuel assembly and core design methods. In this paper the Framatome ANP's BWR methodology COMPASS (**com**prehensive BWR **p**rogram **a**ssembly for **s**teady state and **s**afety analysis) is presented together with results of recently performed validation and verification work. The paper demonstrates that COMPASS meets the challenges in all areas of steady state and transient fuel assembly and core analysis: neutronic, thermal hydraulic and mechanical fuel assembly analysis, in-core fuel management analysis and core monitoring, core transient analysis including stability, and plant transient analysis. This is achieved by employing advanced physical models and by extensive validation of the methodology. The validation is based on experimental programs, measurements at Framatome ANP test facilities, and most important, comparison of the predictions with a great wealth of measured data gathered from BWR plants during many years of operation.

1. INTRODUCTION

The design of modern BWR fuel assemblies and reload cores is governed by permanent basic requirements: safe and reliable performance, optimal fuel utilization, and a high degree of flexibility in core operation. While these general requirements do not change from plant to plant, they are accompanied by current design and operational trends, which may be plant specific: increased enrichment and discharge burnup, high Gadolinium loading in the fuel assemblies, part-length fuel rods, MOX insertion, power uprate, strongly varying operating cycle length (from ½ year up to two years mainly in the U.S.), and challenging core loading strategies such as super-low leakage.

These circumstances determine the challenges to be met by the design methodology: excellent predictive capability for all relevant steady state and transient conditions, accuracy in the demonstration of margins with respect to operational and safety limits; a high level of harmonization in the modules of the overall code system, which ensures consistency of all analyses; and a degree of code system automation which allows the designer to respond quickly and effectively to the needs of the customer.

This paper reviews the main codes used in COMPASS, Framatome ANP's **com**prehensive BWR **p**rogram **a**ssembly for **s**teady state and **s**afety analysis. Emphasis is placed on European BWR fuel assembly and core design and on methods for which very detailed validation and verification work has recently been performed.

2. COMPASS OVERVIEW

The BWR fuel assembly and core design methodology COMPASS can be visualized as a four step cascade comprising fuel assembly design, in-core fuel management analysis and monitoring, core transient and stability analysis, and plant transient and accident analysis (see Fig. 1). This picture reflects Framatome ANP's BWR methodology for European BWRs, but most of the codes shown are also used by Framatome ANP for the analysis of BWRs in the USA and the Far East, and some (e. g. MCNP, CASMO-4, CARO, S-RELAP, PRIMO) are employed in Framatome ANP's PWR methodology CASCADE-3D as well.

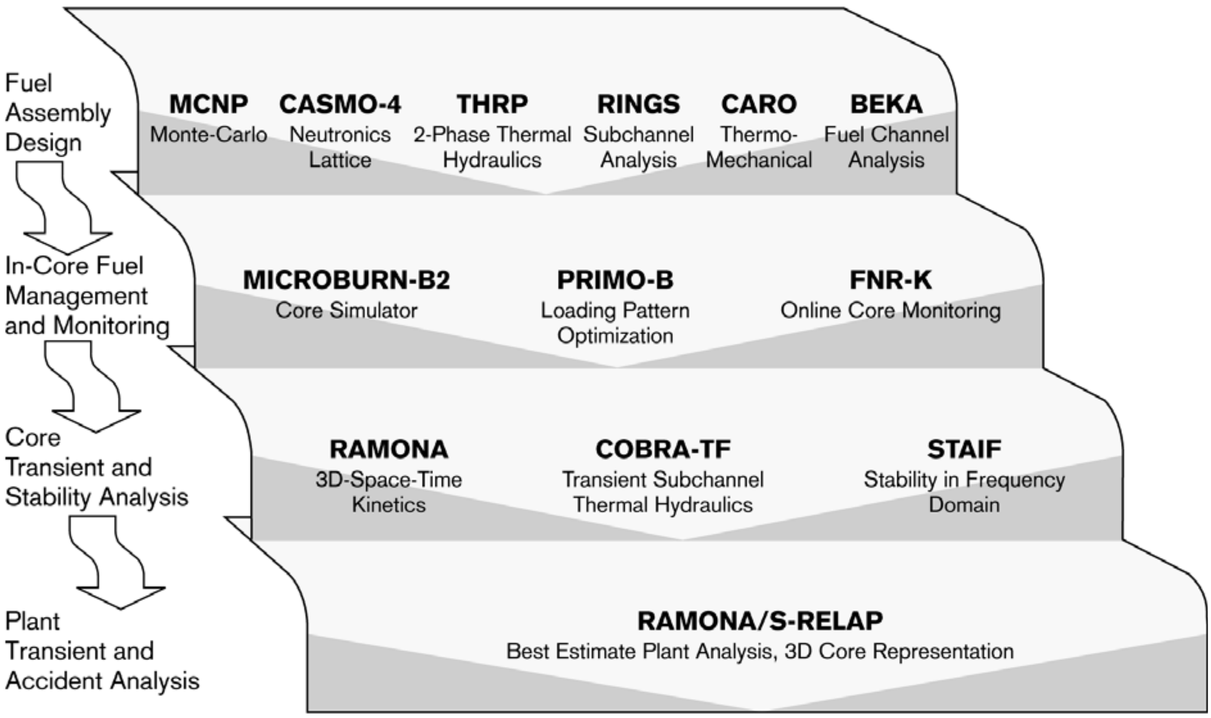

Figure 1. Overview of Framatome ANP's BWR Methodology COMPASS for European BWRs

3. FUEL ASSEMBLY DESIGN

The fuel assembly design code package comprises the Monte Carlo code MCNP, used for reference neutronic calculations, the 2D lattice code CASMO-4, furthermore the two-phase thermal hydraulic design code THRP, and RINGS, used for steady state thermal hydraulic subchannel analysis. The thermo-mechanical design analysis of fuel rods is performed with the code CARO, structural analysis especially for the fuel channel is done with BEKA.

3.1 NEUTRONIC FUEL ASSEMBLY DESIGN

The lattice code CASMO-4 developed by Studsvik/Scandpower AB, is used by Framatome ANP for BWR neutronic fuel assembly analysis and design /1/. This code provides nuclear data for all further steady state and transient codes of the COMPASS system. The validation available from the software supplier was significantly extended through Framatome ANP own validation efforts. For example, the k-infinity over exposure and the nuclide density predictions up to high burnup MOX and UO_2 fuel were compared with Monte Carlo depletion calculations carried out with OCTOPUS /2/. This code system consists of the Monte Carlo program MCNP-4C coupled with the burnup code ORIGEN-S. For both UO_2 and MOX fuel assemblies, the CASMO-4 results for reactivity (see Fig. 2) and nuclide inventory are in close agreement with OCTOPUS up to high burnup.

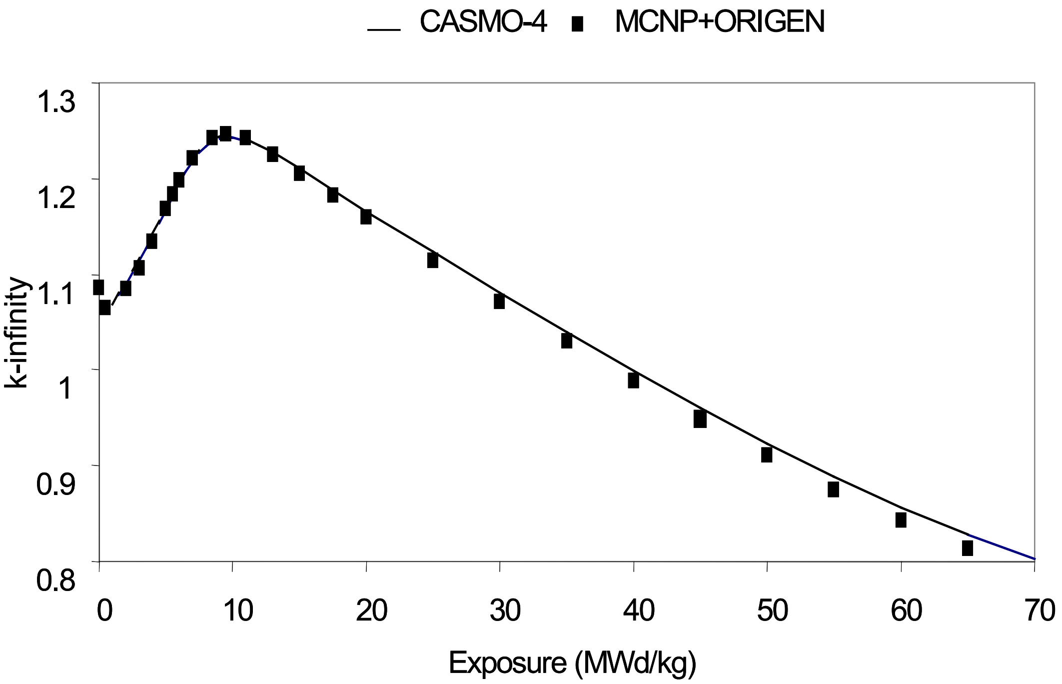

Fig. 2. Comparison of k-infinity calculated with CASMO-4 and OCTOPUS

Furthermore, CASMO-4 results were compared with nuclide concentrations measured in the ARIANE program for high burnup fuel /3/. Results of the comparison of nuclide concentrations are presented in Fig. 3. For both UO_2 and MOX fuel, the CASMO-4 results are in close agreement with the experimental data. The agreement of CASMO-4 results with this experimental data is equally good as for comparisons with MCNP-4C coupled with the burnup code ORIGEN-S.

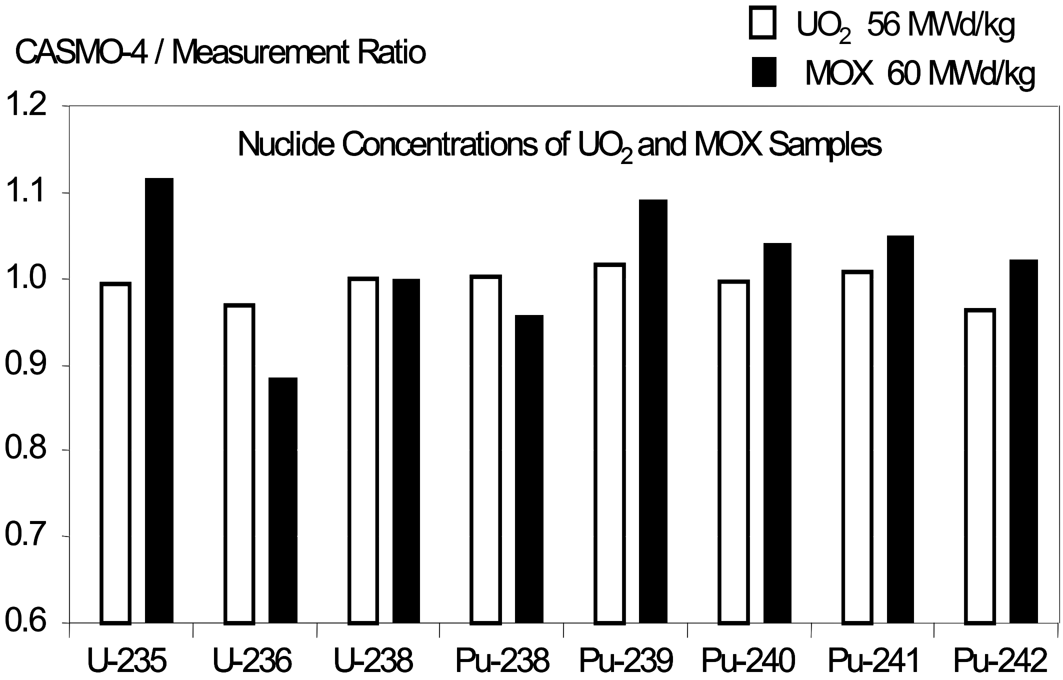

Fig. 3. Comparison of nuclide concentrations calculated with CASMO-4 and ARIANE Measurements

An overview of CASMO-4 qualification work is given in Table I. As can be seen, the differences between CASMO-4 results and the individual validation data are very small in general, which demonstrates the excellent accuracy achieved with CASMO-4 in BWR lattice analysis.

Studsvik	MCNP-4A	k-eff	1.5-3.0‰
		Fission rate distrib.	1-1.5%
	Isotope measurements*	Nuclide concentr.	3-7%
	Crit. experiments	k-eff	2-3.5‰
		Fission rate distrib.	1%
Others (Jap.)	Crit. experiments	k-eff	< 3.5‰
		Fission rate distrib.	1%
Framatome ANP	MCNP-4B	k-eff	< 4.5‰
		Fission rate distrib.	1-2%
	Isotope measurements** (ARIANE)	Nuclide concentr..	<5-10%
	MCNP/ORIGEN	k-eff	< 4.5‰
		Fission rate distrib.	1-1.5%
		Nuclide concentr.	< 10%
*Actinides, exposure 35000 MWd/t		**Actinides, exposure 60000 MWd/t	

Table I. Overview of CASMO-4 qualification

3.2 THERMAL HYDRAULIC FUEL ASSEMBLY DESIGN

The code THRP is applied for the detailed analysis and design of the thermal hydraulic fuel assembly conditions /4/. Pressure drop, critical power, and void distribution are key quantities calculated with THRP. For the determination of the fuel assembly critical power and the void fraction vs. quality, the code employs dryout correlations and void correlations, respectively. The calculated data is qualified by comparisons with test results from Framatome ANP's own test facility KATHY. Extensive void measurements were performed recently in the KATHY Loop. Typical results are shown in Fig. 4. The void measurements have confirmed that the correlation used in thermal hydraulic fuel assembly and core analyses is applicable with very good accuracy to the ATRIUM^{TM} 10 fuel assembly design at all axial or void levels.

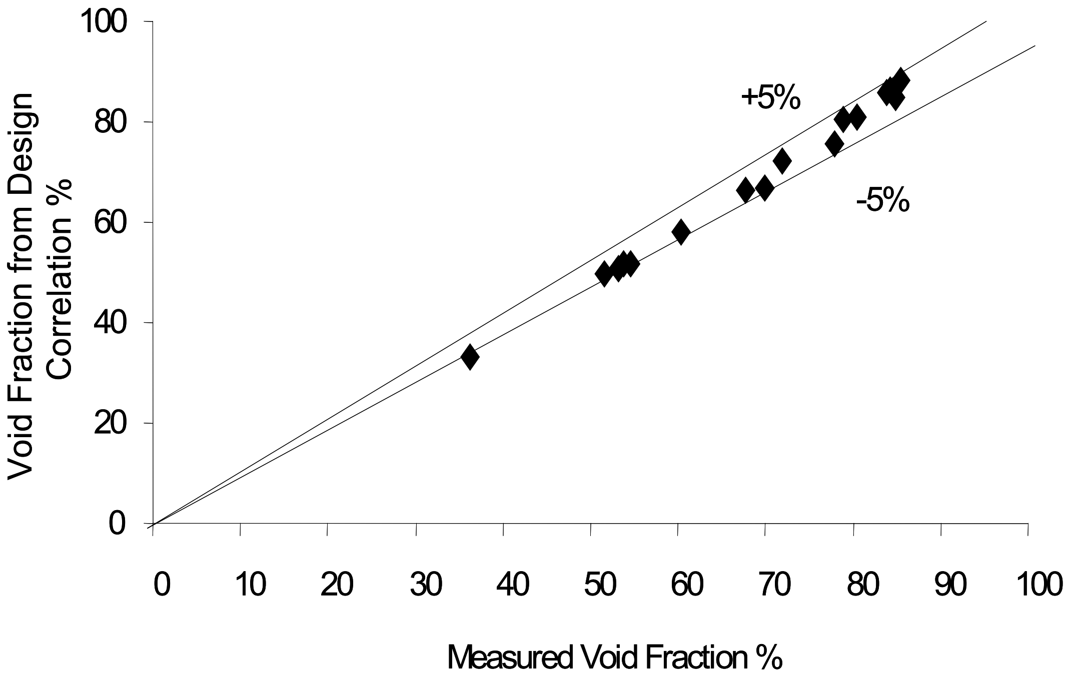

Fig.4. Comparison of design correlation with measured void fractions

3.3 SUBCHANNEL ANALYSIS

The subchannel analysis is performed with RINGS, an annular flow code which predicts dryout power and dryout location by calculating the conditions at which the liquid film flow rate is reduced to zero. Evaporation, droplet entrainment and droplet deposition are modelled in detail. Special emphasis is put on the modelling of spacer effects. Comparison with experimental data of 3x3 and 4x4 tests demonstrate the capability of RINGS to predict the flow quality and mass flux in subchannels under typical BWR operating conditions. With RINGS, experimental critical data for various fuel assembly types up to 10x10 were successfully post calculated. Therefore, the extent of the complex and costly full-scale tests carried out to determine the thermal-hydraulic characteristic of new assembly and spacer designs can be significantly reduced.

3.4 THERMO-MECHANICAL AND MECHANICAL FUEL ASSEMBLY ANALYSIS

CARO, the thermo-mechanical design code, is the tool for fuel rod analysis based on statistical methods /5/. The course of the probabilistic analysis with input and output is illustrated in Fig. 5. The detailed physical models of CARO have undergone an extensive validation, the data base covering a large burnup range up to about 100 MWd/kgU. This design analysis method is employed to establish a Thermal Mechanical Operating Limit (TMOL; maximum local power vs. burnup) that can be used for both core monitoring and the optimisation of fuel management.

The fuel channel behaviour and free control rod motion over the fuel channel lifetime is verified with Framatome ANP's code BEKA. This analysis is based primarily on realistic time histories of differential pressure and neutron flux. For the structural analysis of accident conditions, the non-linear response of the fuel assembly and fuel channel to the excitation is analysed using an explicit time integration scheme. Thus, the stresses in the fuel channel, fuel structure and the impact forces of the spacer grids can be computed.

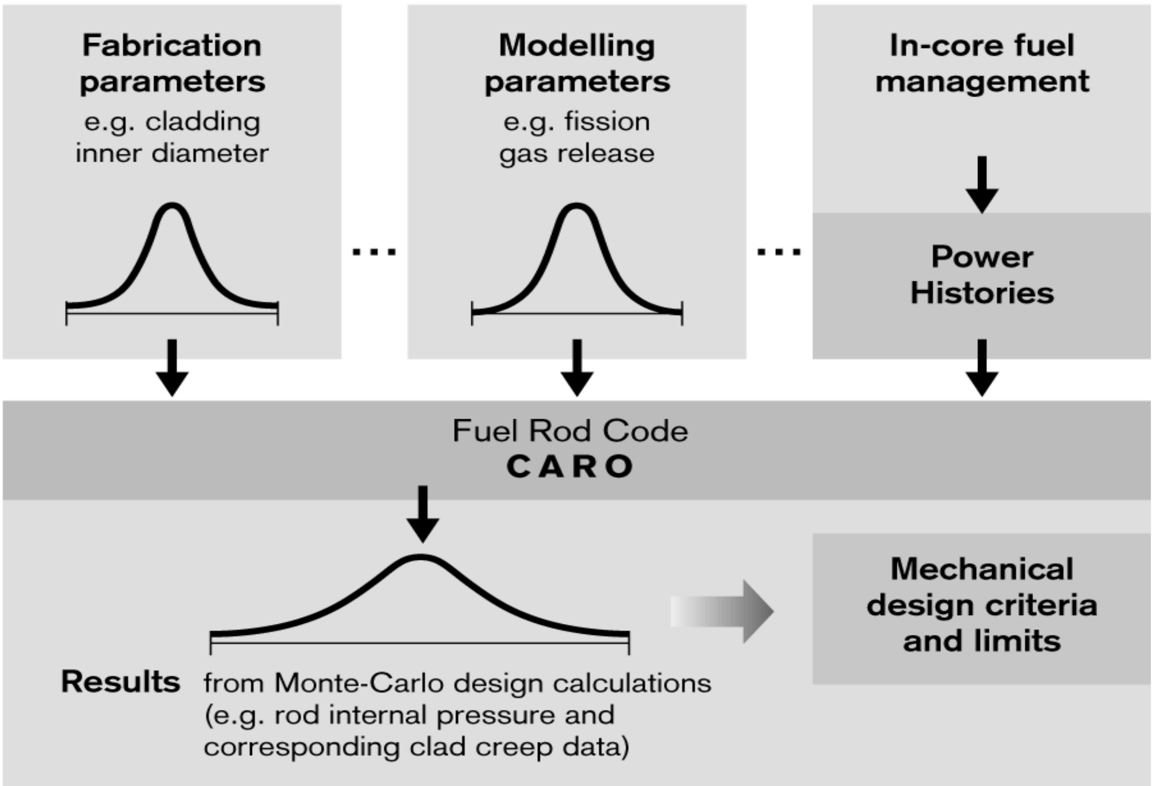

Fig. 5. Simplified schematic flow diagram of statistical fuel rod design

4. IN-CORE FUEL MANAGEMENT AND MONITORING

The 3D core steady state design and optimisation is performed with MICROBURN-B2, whereas FNR-K (in USA the code system POWERPLEX is used instead) is the online core monitoring software package (see Fig. 1).

MICROBURN-B2 /6/ was developed by Framatome ANP. Several years ago, the code system CASMO-4/MICROBURN-B2 obtained the NRC approval and is thus used by Framatome ANP worldwide for BWR core analysis and in-core fuel management optimisation. MICROBURN-B2 has advanced neutronic models, such as two-group advanced nodal expansion, pin power reconstruction, and microscopic depletion analysis for all important fuel nuclides. The thermal-hydraulic module of MICROBURN-B2 is consistent with the design thermal hydraulic program THRP. The neutronic lattice data is gained from CASMO-4 calculations.

MICROBURN-B2 has undergone comprehensive validation. Gamma scans (integral assembly and pin-by-pin) and measured data from reactor operating cycles provide the main measurement support for the qualification of calculated power density distributions. Special emphasis was put on the validation of pin powers calculated with the code's pin power reconstruction model. In 1998, a pin-by-pin Gamma scan of one ATRIUM^{TM} 10 UO_2 fuel assembly was carried out by Framatome ANP at the Gundremmingen BWR /7/. This scan was done after the first irradiation period of the fuel assembly. Since the assembly had been positioned in the central region of the core, the measurement provided local power density data for the maximum power achieved by the assembly during its insertion history. During the same period, additional gamma scans were made at Gundremmingen within the frame of the international GERONIMO Program /8/. These pin-by-pin measurements were made on one 9x9 MOX fuel assembly and two adjacent 9x9 UO_2 fuel assemblies.

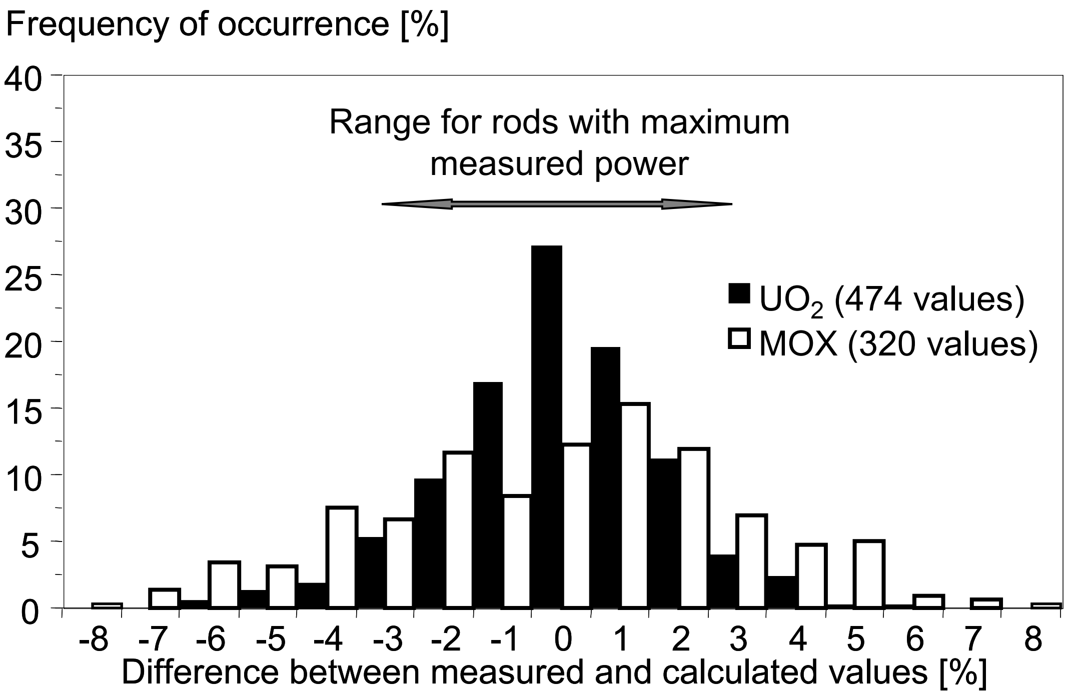

Fig. 6. Frequency distribution of all local deviations for MOX and U fuel assemblies between pin-by-pin Gamma-Scan measurements and MICROBURN-B2 calculations.

The scanned MOX and UO_2 assemblies were positioned in a core region with a rather steep radial power density gradient. Fig. 6 shows the frequency distribution of all local deviations (ratio of calculated and measured Barium-140 distributions) between measured and calculated pin power densities for all fuel assemblies scanned. In spite of aggravating conditions, such as MOX/UO_2 spectral interaction and the significant radial power gradient, the agreement between measured and calculated pin powers is very good

In Europe alone about 170 reactor operating cycles were analysed with MICROBURN-B2 and the analytical results were compared with measured data. In addition, comparisons with results from codes with more detailed physical modelling, such as CASMO-4 four-bundle calculations, yield analytical support. In all of this validation work, MICROBURN-B2 has consistently demonstrated a high level of accuracy in the prediction of reactivity and nodal power distributions for all kinds of loadings, including cores with mixed UO_2 and MOX fuel. A summary of the uncertainty margins is given in Table II. Comparisons with previous models clearly demonstrate the improvement in the simulation of 3D BWR cores achieved during the last several years.

Uncertainty of k-effective (hot and cold)	< 2‰-3‰ (3.5‰)
Uncertainty of pin power (RMS 1σ)	< 2% (2.7%) UO_2
	< 3% (3.6%) MOX
Uncertainty (RMS 1σ) of simulated 2D Gamma TIP distributions.	2%-3% (4%)
Uncertainty (RMS 1σ) of simulated 3D Gamma TIP distributions.	3%-5% (7%)
The maximum values are given in parentheses	

Table II. MICROBURN-B2 method uncertainty.

Framatome ANP's on-line monitoring of the BWR cores in Europe is based on the FNR-K system /9/. This simulator provides detailed information on significant reactor physics parameters and margins with respect to safety-related limits. Apart from providing the operator with instantaneous graphical output of essential results, other attractive features are the archiving and reconstruction of reactor state points as well as the capability of performing predictive calculations. FNR-K physical model includes an adaptation software, which corrects the predicted powers according to the difference to measured gamma sensitive travelling in core probes (TIP). The model is qualified by comparisons with gamma-scan measurements, and by comparisons of predicted TIPs before adaptation with measurements. The monitoring system is well established and runs on different hardware platforms in several German BWR plants.

5. CORE TRANSIENT AND STABILITY ANALYSIS

5.1 DESIGN ANALYSIS

Within Framatome ANP's BWR methodology COMPASS, the codes used in the area of core transient and stability analysis are: RAMONA for 3D space-time kinetics in post- and predictive calculations, including stability analysis, and STAIF for stability analysis in the frequency domain. The input data for both codes is generated starting from calculations with the assembly and core design codes CASMO-4, THRP and MICROBURN-B2 (see Fig. 1). Framatome ANP has vast experience in the field of stability as well as in-core transients in

BWR plants. This includes detailed analysis of in-phase and out-of-phase oscillations as well as reactivity-initiated transients. The accurate demonstration of sufficient margins during transients contributes to plant availability. Here too, extensive validation is the key to success.

RAMONA, originally developed by Studsvik/Scandpower /10/, has been extended to calculate two-phase thermal-hydraulics in accordance with the steady-state design code THRP. The code has undergone validation beyond that of the software supplier, through comparisons with stability measurements at BWR plants and the KATHY loop, as well as through international benchmarks. RAMONA gets its reactor physics input from CASMO-4 and MICROBURN-B2. In generating this data, quantities like void fraction, void fraction history, pellet temperature, coolant temperature, burnup and control state are varied to permit not only the description of the initial conditions of the core, but of the subsequent transient conditions as well.

One focus of the RAMONA application is on the 3D investigation of fast reactivity insertion accidents, such as control rod drop. In these cases, there is no need to model in detail systems in the reactor periphery. The accuracy of the calculated pellet enthalpy rise for some reactivity insertion accidents (RIA) can be significantly improved with 3D analysis compared to investigations where a simplified geometric core model is used /11/.

The NRC-approved frequency domain design code STAIF /11/ has been benchmarked against stability measurements at BWR plants and with measurements in the KATHY loop /11/ as well as with the Ringhals-1 OECD/NEA data. It incorporates a linearized, Laplace transformed model of the reactor core and the recirculation loop. The differential equations and correlations describing each part of the system are consistent with those used in Framatome ANP's steady state BWR design codes. One-dimensional neutron kinetics permits axially variable void and Doppler feedback and takes six groups of delayed neutrons into account. The main features of the thermal hydraulics model are: multiple channels with independent geometry and axial power distribution; two mass, one energy and one momentum equation; thermodynamic non-equilibrium and non-equal phase velocities; empirical correlations for slip and friction; boiling model and heat transfer. Based on transfer functions that define the linear dynamic behaviour, the code estimates the decay ratio for the fundamental (core-wide) and the first azimuthal (out-of-phase) modes. The latter covers the effect of radial and axial power distribution on the probable oscillation mode.

In Fig. 7 the comparison of measured decay ratios for various cycles of different BWRs worldwide and calculated results with STAIF are presented, showing very close agreement for the whole range of decay ratios. From the participation in the Ringhals-1 stability benchmark organized by OECD/NEA, Framatome ANP also obtained valuable evidence of the excellent qualification of STAIF /11/. The stability analysis in the frequency domain performed by this code is a reliable and an accurate way to compare the stability merits of individual fuel assembly designs and of different core loadings.

5.2 STABILITY MONITORING

Framatome ANP provides two online stability monitoring systems: ANNA™, based on decay ratio evaluation with autoregressive noise analysis, identifies the margin to instabilities and the onset of regional oscillations. CSM detects global and regional oscillations in an adequate time frame and can activate different staggered countermeasures, depending on frequency, growth rate and height of the signal amplitude.

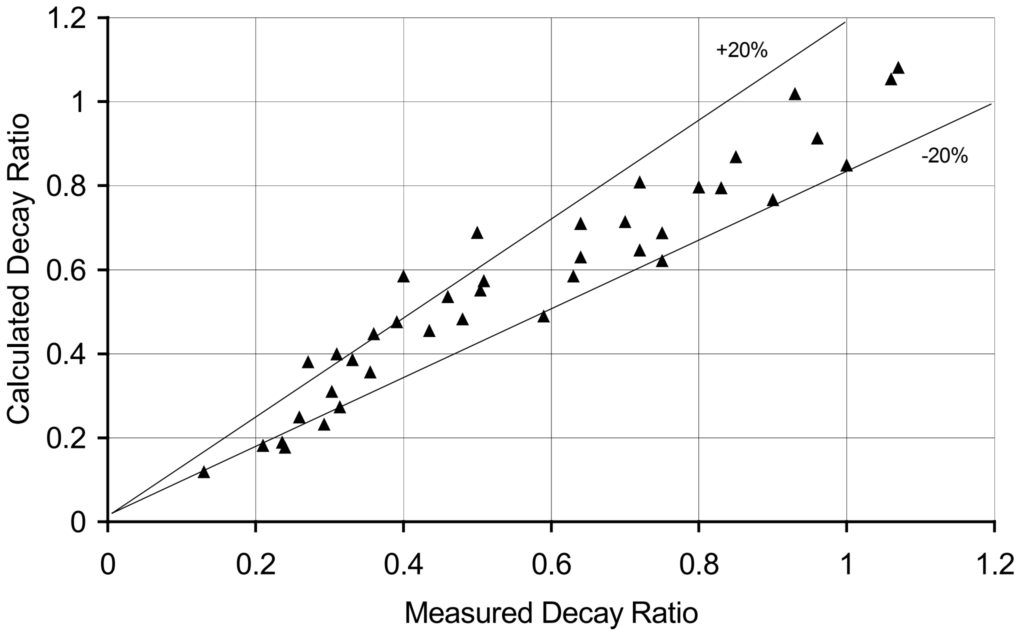

Fig. 7. Comparison of measured decay ratios in various cycles of different BWRs and calculated decay ratios with STAIF.

6. PLANT TRANSIENT AND ACCIDENT ANALYSIS

RAMONA has also been used by Framatome ANP to analyse operational transients, e. g. loss of the main heat sink. In these cases, reactor systems such as the reactor vessel with internals, the recirculation system, the steam and feed water system with the important components and the reactor control devices can be modelled. More recently, for European BWRs, the coupled code system RAMONA/S-RELAP has successfully been applied for plant transient analysis, for example for the OECD/NRC BWR turbine trip benchmark. The coupled system represents an advanced method for simulation and analysis of operational transients and LOCA. The results support optimal reactor operation by improving safety margins. For several years Framatome ANP has used S-RELAP5 for both PWR and BWR analysis. Decades of in-house experience, and access to extensive plant data for both reactor types, lead to an excellent „best estimate“ system which is well suited for evaluating complex transients.

7. SUMMARY AND CONCLUSIONS

COMPASS, Framatome ANP's comprehensive methodology for BWR fuel and core design, features a high degree of harmonization in the methodology and advanced code system automation. This allows the designer to respond quickly and effectively to the needs of the customer. The validation basis has been greatly extended through comparisons with recent measurements, by means of comparisons with higher order methods and by international benchmarks. The constant high quality of the validation and verification work guaranties optimal results for the customer when challenging BWR fuel assembly and core design is used. The strength of the code validation base is largely due to Framatome ANP's proximity to the BWR plants in reload fuel and plant service activities. On-going development work, for example in the areas of transient subchannel analysis, automatic core loading optimisation, and more refined RAMONA/S-RELAP plant transient analysis will further improve the overall capability of the methodology and the accuracy of results. The goal is to be even more responsive to the customer needs now and in the future as well.

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