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Advanced methods for BWR transient and stability analysis

A. Schmidt, F. Wehle, S. Opel and R. Velten
AREVA, AREVA NP, Erlangen, Germany
andreas.schmidt@areva.com

Abstract

The design of advanced Boiling Water Reactor (BWR) fuel assemblies and cores is governed by the basic requirement of safe, reliable and flexible reactor operation with optimal fuel utilization. AREVA NP's comprehensive steady state and transient BWR methodology allows the designer to respond quickly and effectively to customer needs.

AREVA NP uses S-RELAP5/RAMONA as the appropriate methodology for the representation of the entire plant. The 3D neutron kinetics and thermal-hydraulics code has been developed for the prediction of system, fuel and core behavior and provides additional margins for normal operation and transients. Of major importance is the extensive validation of the methodology. The validation is based on measurements at AREVA NP's test facilities, and comparison of the predictions with a great wealth of measured data gathered from BWR plants during many years of operation.

Three of the main fields of interest are stability analysis, operational transients and reactivity initiated accidents (RIAs). The introduced 3D methodology for operational transients shows significant margin regarding the operational limit of critical power ratio, which has been approved by the German licensing authority. Regarding BWR stability a large number of measurements at different plants under various conditions have been performed and successfully post-calculated with RAMONA. This is the basis of reliable pre-calculations of the locations of regional and core-wide stability boundaries.

1. Introduction

The design of modern BWR fuel assemblies and reactor cores is governed by ongoing 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 burn-up, high Gadolinium loading in the fuel assemblies, part-length fuel rods, insertion of MOX fuel, power uprate, variation of fuel cycle length and challenging core loading strategies such as super-low leakage. These aspects 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 customers.

In [1] AREVA NP's comprehensive BWR code system for steady state and safety analyses COMPASS has been presented. This BWR fuel assembly and core design methodology 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 (fig. 1). The picture reflects AREVA NP's BWR methodology for European BWRs, but most of the codes shown are also used by AREVA NP for the analysis of BWRs in the USA and the Far East, and some are employed in AREVA NP's PWR methodology CASCADE-3D as well. This broadens the validity range in utilizing the large spectrum of different plants and cycles and the synergies between BWR and PWR application areas.

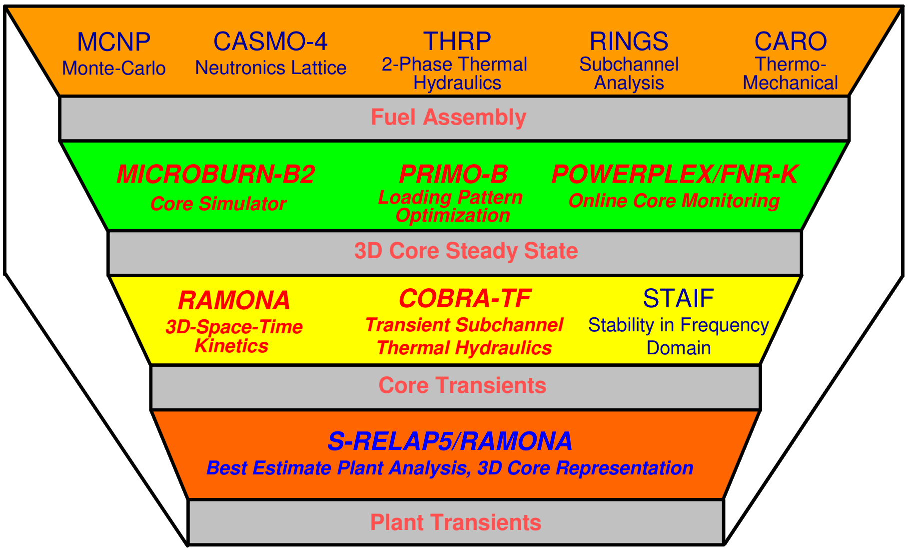

Figure 1: Overview of AREVA NP's Methodology COMPASS for European BWRs.

For transient applications the 3D neutronic/thermal-hydraulic code RAMONA is used. RAMONA was originally developed by Studsvik/Scandpower [2], but was extended by AREVA NP to satisfy all the special needs, to introduce fuel specific experiences and to fit in the code system COMPASS. RAMONA as a stand-alone code is used for stability analyses and for fast reactivity insertion accidents. To analyze operational transients a good representation of the entire plant is important. For this application the coupled code system S-RELAP5/RAMONA is used. In this coupled system RAMONA only represents the active core whereas the recirculation loop and all the other important reactor systems are modeled by S-RELAP5 [3]. This paper presents the experience of AREVA NP with this transient 3D code system obtained during the last years.

2. Stability Analyses

Power oscillations are a well known phenomenon in Boiling Water Reactors and AREVA has worked in the area of analysis, detection, and suppression [4, 5, 6, 7] from the beginning of BWR operation. For high power/low flow operating conditions (fig.2, left) associated with unfavorable core power distributions, BWR operation requires attention with respect to potential power and flow oscillations. The most likely modes of BWR instabilities are the core-wide or global mode with in-phase oscillation and the regional mode, where the power in one half of the core oscillates out-of-phase with the power in the other half. In the latter mode, local flux signal oscillations tend to cancel out in the summation, so the local amplitudes may be significantly higher than the average flux signal measured by the power range monitors (fig.2, right).

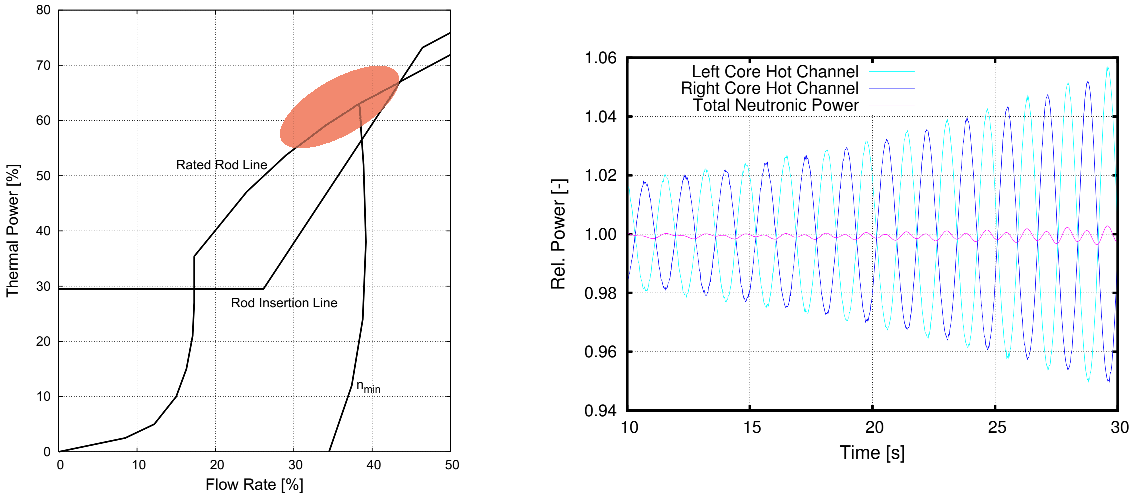

Figure 2: Left: Schematic representation of a BWR power/flow map and region (red marked) sensitive to power oscillations. Right: Power oscillations in opposite core regions and core average power signal.

In the development of advanced fuel assemblies, a major feature to be taken into consideration is the improvement of the stability performance of new reloads. The stability characteristics of a fuel assembly are known to depend on several design parameters and operating conditions. On the neutronics side, a smaller void-reactivity feedback is stabilizing, where e.g. the large water channel of the ATRIUM^{TM} design contributes to that effect. Flow perturbations and the resulting void fraction changes in the active fuel region have a smaller impact on reactivity due to the presence of moderator in the large water channel.

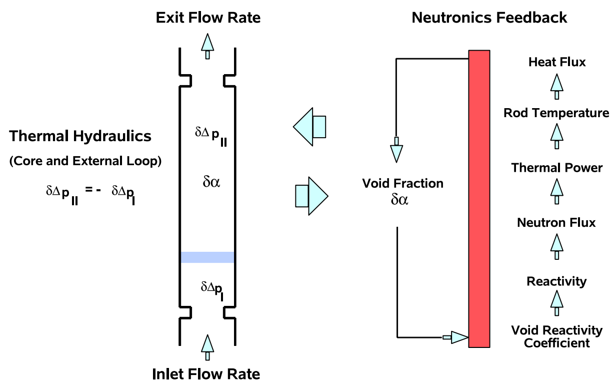

Figure 3: Feedback mechanisms of BWR stability.

Thermal-hydraulic stability can be qualitatively characterized by the boiling- (Δp_{II}) to non-boiling (Δp_I) pressure drop ratio (fig.3). A perturbation in the void fraction results in perturbations of both pressure drops, where the boiling pressure drop lags the non-boiling pressure response (density wave). At a certain frequency, this time lag will result in a pressure drop response that will reinforce the void fraction perturbation. This destabilizing trend due to positive feedback increases as the boiling- to non-boiling pressure drop increases. Generally, it can be stated that any measure that increases the non-boiling pressure drop has a stabilizing effect, while an increase of the boiling pressure drop is destabilizing. The good thermal-hydraulic stability behavior of the ATRIUM^{TM} design mainly results from the introduction of part length fuel rods which gives an optimized two-phase/single-phase pressure drop ratio.

For stability analyses in the time domain AREVA NP has applied RAMONA for many years to nearly all European plants [4, 5, 6, 7]. The standard application for RAMONA in this field is the investigation of the stability performance of a loading pattern. At a certain burn-up state and operating point the response in reactor power or other interesting parameters after a perturbation, for example with a control rod movement, is analysed. For determination of the decay ratios (ratio of two consecutive maxima) two groups of homologous hot channels are selected. These channels should have preferably a large and equal distance to the center of the core and should be shifted azimuthally by about 45°. Based on the transient data of the hot channel powers, the decay ratios are determined.

To maintain the high level of availability for BWRs, RAMONA is extensively qualified based on data from the thermal-hydraulic test facility KATHY in Karlstein and on plant stability measurements. These measurements are performed in many plants at different burn-up states. Specified operating conditions are set and the neutron flux is detected with power range monitors for a few minutes. Based on noise analysis codes (e.g. ANNA), the decay ratios are determined.

European BWR's are faced different regulatory requirements. Based on this, different precautionary measures against instabilities are taken: For example in some plants the power-flow map is divided in three regions: the normal operating region, the surveillance region, where APRM and LPRM signals are monitored and the exclusion region, where a scram would be initiated. The validity of these regions is verified by measurements or calculations.

AREVA NP has gained a tremendous experience in the area of BWR stability This is demonstrated in figure 4, which shows a comparison of reactor measurement data and calculated decay ratios. The measured decay ratios mentioned in this figure are obtained for different European power plants and different core loadings under various operating conditions recorded in the recent years. The deviations are always lower than 0.1. The high predictive quality of RAMONA even sometimes allows replacement of stability measurements.

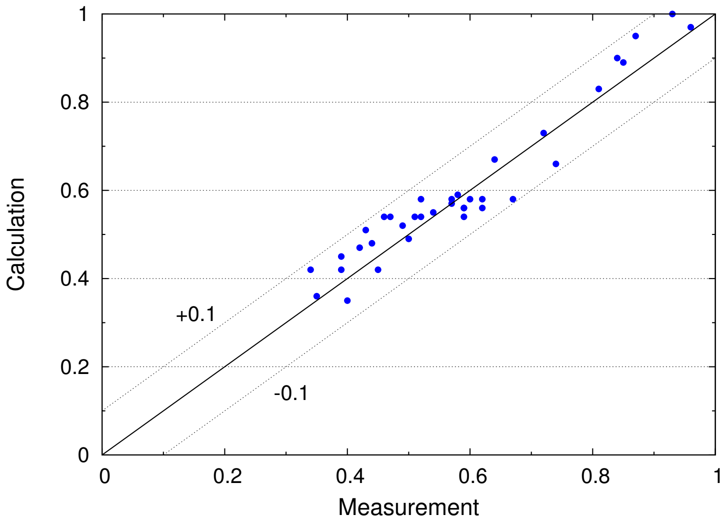

Figure 4: Comparison of decay ratios from measurements and from RAMONA analyses.

With a time domain code like RAMONA not only the determination of decay ratios is possible, also the real time behavior of the reactor power, other important parameters and the influence of the whole reactor system can be investigated. So for example the efficiency of countermeasures against instabilities can be evaluated.

An important part of todays stability analysis is also the question if the core tends to oscillate in a global or a regional mode. As mentioned before a regional oscillating neutron flux may not be detected with average power range monitors. For the investigation of regional modes an out-of-phase perturbation is applied, or like in the examples shown in fig. 5 only in one half of the core control rods are moved. In both cases the oscillations of two hot channels in the different core halves are starting in-phase. But in the right case the phase shifts and the oscillations remain out-of-phase till they are completely disappeared. What theoretically may happen for conditions far away from the stability boundary is shown in fig. 6. The neutron fluxes in the different core halves are first oscillating in-phase in spite of an out-of-phase perturbation. But after 15 seconds the phase shifts and the two fluxes are oscillating completely out-of-phase. On the right side of the figure the total neutronic power is shown. After the oscillation seems to decay (with a decay ratio slightly lower than 1), it starts to grow again, with a two times higher frequency, which is an indication of regional oscillations. The rising issue is, that the amplitude of the oscillations in a hot channel could be much higher than of the average flux, detected by average power range monitors. It has been further observed that even for an in-phase perturbation after some time an out-of-phase oscillation establishes if this is the dominating oscillation mode.

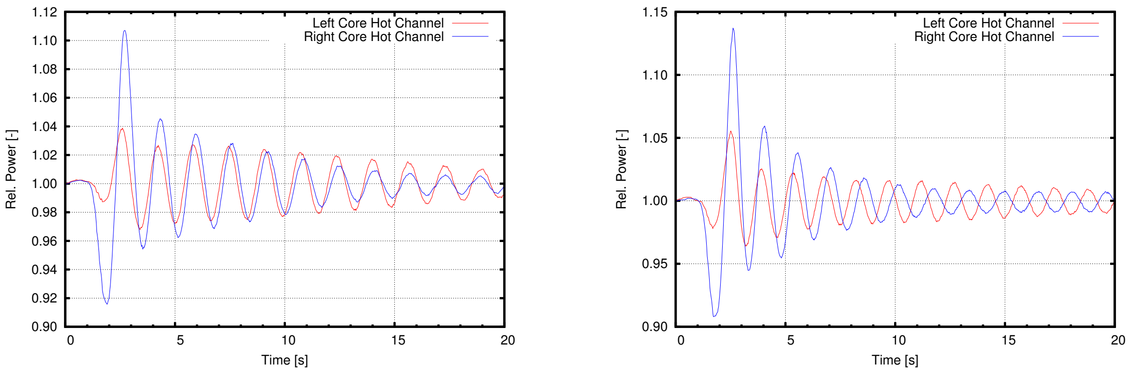

Figure 5: Power oscillations in opposite core regions after an out-of-phase perturbation for two different cases.

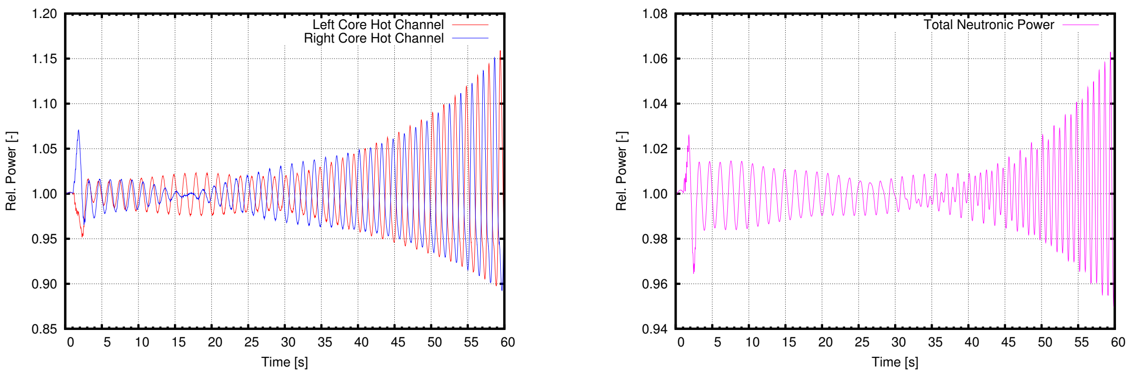

Figure 6: Power oscillations in opposite core regions and core average power signal for a regional instable situation.

Under such extreme conditions a transient dry-out could occur in the hot channels. Large oscillations in the mass flow rate could prevent the heater rods from being sufficiently cooled, and lead to the dynamic processes of dry-out and re-wetting of the heat-exchange surfaces [8]. Fig.7 shows the measured (KATHY) transient behavior of the cladding temperature and the inlet flow rate in the unstable region during a continuously increase of heat input. Also this demonstrates the need for highly qualified and reliable stability methodology with special attention to be capable to predict also out-of-phase oscillations.

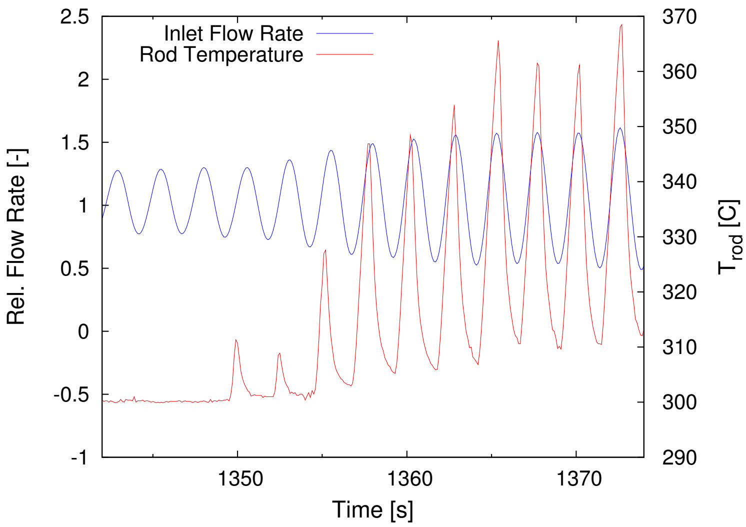

Figure 7: KATHY measured inlet flow rate and cladding temperature.

3. Reactivity Initiated Accidents

The reactivity initiated accident (RIA) is one of the design basis accidents in the safety analysis of boiling water reactors. Postulated events are the control rod drop or the continuous control rod withdrawal. A detailed space-time reactor simulation including thermal hydraulic effects is required to properly account for the feedback effects that follows the control rod movement.

Studsvik/Scandpower proved the applicability of RAMONA for a realistic examination of such events [9]. AREVA NP started with 3D RIA analyses already in 1993, results obtained with RAMONA can be found in [10, 11].

Postulating a rod drop accident with significant enthalpy rise in a German BWR means to assume the highly unlikely coincidence of several events:

- the core is near criticality
- a control rod gets stuck during withdrawal, thereby disconnecting from its drive
- the weight sensing device at the control rod drive fails
- the drive is further withdrawn
- the stuck rod gets loose again and falls out of the core, a system of clamps and notches limits the drop to a maximum of 21 cm

- in order to insert significant reactivity, the rod has to get stuck at a radial and axial position where the differential rod worth is high enough to cause a prompt supercritical transient
- the neutron flux excursion is very localized so that high relative peaking factors occur

Especially the last two conditions are typically fulfilled assuming the dense rod patterns and the essentially un-voided conditions during reactor start-up. Since the normal rod withdrawal sequence is designed such that the worth of the withdrawn rods is minimized, the withdrawal of an out-of-sequence rod, i.e. an erroneous reactor operation has to be assumed additionally to the above mentioned hardware failures. It can be shown that in this case the differential rod worth is very high in the uppermost part of the core.

In the following the results of such a postulated RIA event analyzed with RAMONA are presented. The examined core loading is an equilibrium cycle with ATRIUM^{TM} 10XP fuel assemblies with an averaged U235 enrichment of 4.77 w/o. This core loading covers fuel assembly averaged burn-up values up to 69 MWd/kg. It is assumed that the considered control rod drops in the axial position with the maximum differential rod worth (dropping length 21 cm) and has a constant speed of 25 cm/s. The initial operating conditions of the reactor are conservatively chosen as cold zero power, which means in detail 20°C moderator temperature, atmospheric pressure and 10^{-8} of rated power. The core is modeled in full core symmetry to describe the highly asymmetric power excursion during the transient. In the neutronic and in the hydraulic model all fuel bundles of the core are explicitly represented. The two group macroscopic cross sections and diffusion parameters, which depend on the material composition, are generated with a special emphasis on the parametric dependence on fuel temperature in order to take into account the Doppler effect. Kinetic parameters are also depending on the material composition. In the fuel pin model, the constitutive relations for the fuel, gap and cladding are functions of the corresponding fuel temperature and exposure. The transient calculations of the rod drop are performed at cycle burn-up states at the beginning and at the end of the cycle, and several different positions of the dropping rod are taken into account. So, the surrounding of the dropping rod is changed and therefore the influence of different fuel assembly burn-up values can be studied. The reactivity worth of the dropping rod has been adjusted to a certain value by modifying the thermal absorption cross-section. Doing this, all different calculations varying the location of the dropping control rod and the cycle burn-up state became comparable. Calculations for prompt reactivity insertions of $\Delta\rho_p = 0.25$ %, 0.50 % and 0.75 % have been performed. Here the prompt reactivity insertion $\Delta\rho_p$ is defined as the total reactivity insertions due to the dropping rod $\Delta\rho$ minus the fraction of delayed neutrons β_{eff}.

As the RIA fuel failure experiments show that there is a correspondence between fuel enthalpy increase and fuel burn-up, today the fuel failure threshold is normally expressed as a function of fuel pin enthalpy rise vs. fuel pin burn-up. For the above mentioned calculations all corresponding pairs of fuel rod local enthalpy rise and local burn-up are plotted in figure 9 (right). As it can be seen in this figure, the maximum enthalpy rises occur in fuel pins with burn-up values up to 30 MWd/kg. For higher burn-up states the enthalpy rises are decreasing with increasing exposure because of the loss of reactivity in these fuel segments. For the highest local burn-up values up to nearly 80 MWd/kg, which belong to the axial lower part of the core, the enthalpy increase is nearly vanishing due to the fact that the power excursion is highly localized in the very top of the active core because the neutron flux distribution is strongly top peaked in un-voided core states. The other fact that can be seen in this figure is the strong influence of the reactivity worth of the dropping rod on the enthalpy rise of the fuel. In typical core loadings rod worths which cause prompt reactivity insertions of up to 0.5 % are reached. The limiting value of the rod worth for cycle specific loading pattern optimization depends on the licensed fuel failure limit line, which today can be different from plant to plant.

As the reactivity insertion of the dropping rod is higher than the fraction of delayed neutrons, the reactor becomes prompt critical during the transient and so the power increases rapidly. For a typical rod worth with a prompt reactivity insertion of about 0.5 % the power peak value is about 450 % of rated

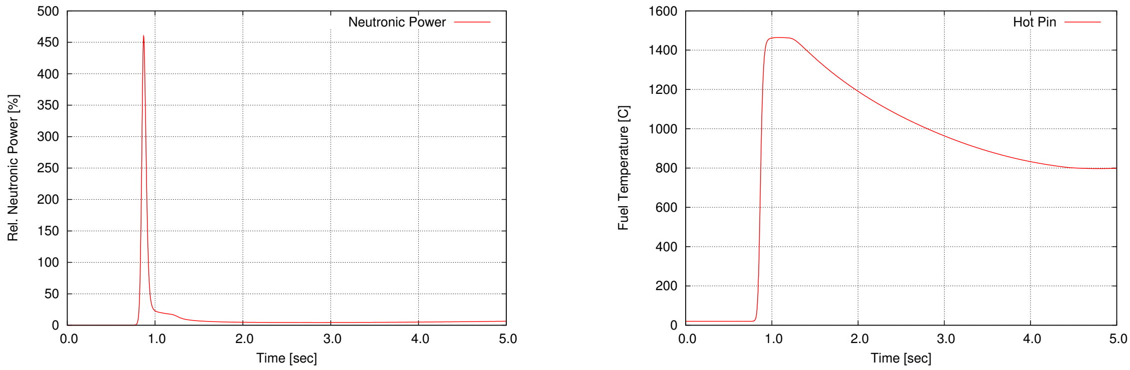

Figure 8: Transient values of neutronic power (left) and Fuel Temperature (right) during a typical RIA.

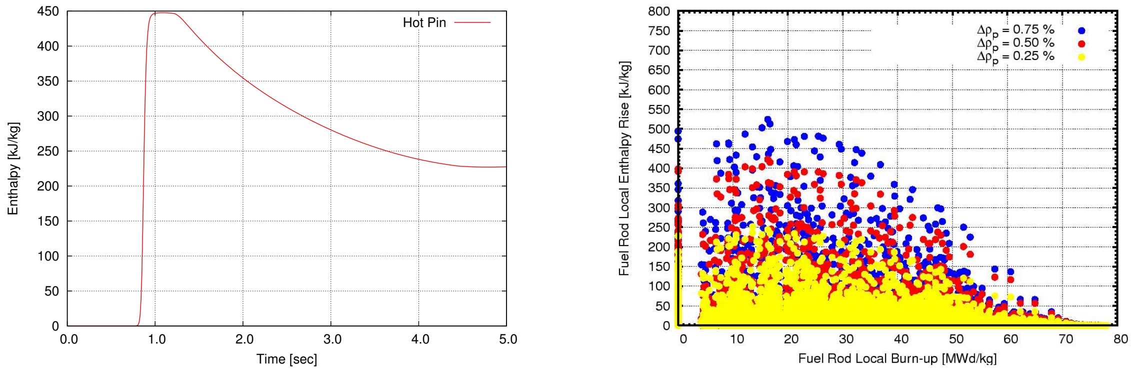

Figure 9: Left: Transient values of enthalpy during a typical RIA. Right: Fuel rod local enthalpy increase vs. fuel rod local burn-up for different promt reactivity insertions.

power (see fig. 8 left). This power is concentrated in the region adjacent to the dropping control rod and causes a large increase in the fuel temperature, the maximum radial averaged fuel pin temperature in such a typical case is about 1450°C (fig. 8 right). Due to the Doppler feedback the power is very fast reduced again, the power pulse has a duration of about 50 ms full width at half maximum. Corresponding to the temperature the radial averaged fuel pin enthalpy also increases very fast and reaches in this example a maximum value of about 450 kJ/kg (fig. 9 left).

4. Operational Transients

For the investigation of operational transients a good representation of the entire plant including the protection system is necessary. Therefore the coupled code system S-RELAP5/RAMONA has been developed. In this coupled system RAMONA only represents the active core whereas all other system components like the recirculation system, the steam and feedwater system and the reactor control devices are modeled in S-RELAP5. This code system has successfully been applied to the OECD/NEA BWR turbine trip benchmark as presented in [12].

The Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) has proposed the Peach Bottom Turbine Trip 2 experiment as a coupled system codes benchmark. In this pressurization event the coupling between core space-dependent neutronics phenomena and system dynamics plays an important role. The available measured plant data makes this benchmark problem very valuable for validating best-estimate 3D core thermal-hydraulic system codes. For this purpose, AREVA NP set up an S-RELAP5/RAMONA three dimensional plant model using the data provided in the benchmark specifications. The best-estimate results are given together with comparisons to measured data in figures 10 and 11. AREVA NP's code system S-RELAP5/RAMONA shows excellent results for the parameters of importance during the transient, namely the pressure wave propagation and the resulting power peak.

In the thermal-hydraulic test facility KATHY also transient tests are performed to investigate the fuel design behavior under non-steady-state conditions, such as those arising in a BWR as a result of a pump trip or turbine trip [13]. Major system parameters such as power, pressure and mass flow are varied according to real reactor transient behavior. These results give additional information for code validation and are the basis for the investigation of the transient applicability of the dry-out correlation.

The operational limit in boiling water reactors regarding critical heat flux includes a safety margin which serves for operational transients, for example the loss of the main heat sink. Up to now this contribution to the operational limit results from transient plant analysis with a simplified one dimensional representation of the reactor core. With a 3D representation of the core a more realistic description of the transient behavior of the entire plant and the fuel assemblies can be achieved [14].

As it can be seen in figure 12 the transient power peak in a calculation with a 3D representation of the core is significantly lower than in comparable cases with a simplified 1D representation. In addition there are some redistribution effects of the radial power shape in the 3D calculations as plotted in figure 13, so the power increase in the hot channels is damped. These effects lead to a smaller transient reduction of the minimal critical power ratio (MCPR) in the 3D case of about 0.07 (see fig. 14) which provides additional margins for normal operation and transients. This advanced 3D transient methodology for operational transients has already been approved by German licensing authority.

5. Conclusion

The design of modern BWR fuel assemblies and reload cores is governed by ongoing basic requirements: safe and reliable performance, optimal fuel utilization, and a high degree of flexibility in core operation. AREVA NP's comprehensive steady state and transient BWR methodology COMPASS allows the designer to respond quickly and effectively to customer needs.

The transient code system S-RELAP5/RAMONA is the appropriate model for the representation of the entire plant. The 3D neutron kinetics and thermal-hydraulics code has been developed for the prediction of system, fuel and core behavior and provides additional margins for normal operation and transients.

At AREVA NP S-RELAP5/RAMONA has been extensively and successfully applied for operational transients and reactivity-initiated accidents (RIAs). The high predictive quality of RAMONA regarding global and regional oscillations even sometimes allows replacement of stability measurements.

The introduction of 3D transient methodology for operational transients has been approved by licensing authority and shows significant margin regarding the operational limit of critical power ratio.

To maintain the high level of availability for BWRs the predictive capability of the code system is of major importance. Based on plant measurements, benchmarks and on transient and stability tests performed for the various fuel designs in AREVA NP's multi-function thermal-hydraulic test loop KATHY the validation of the code system S-RELAP5/RAMONA is secured.

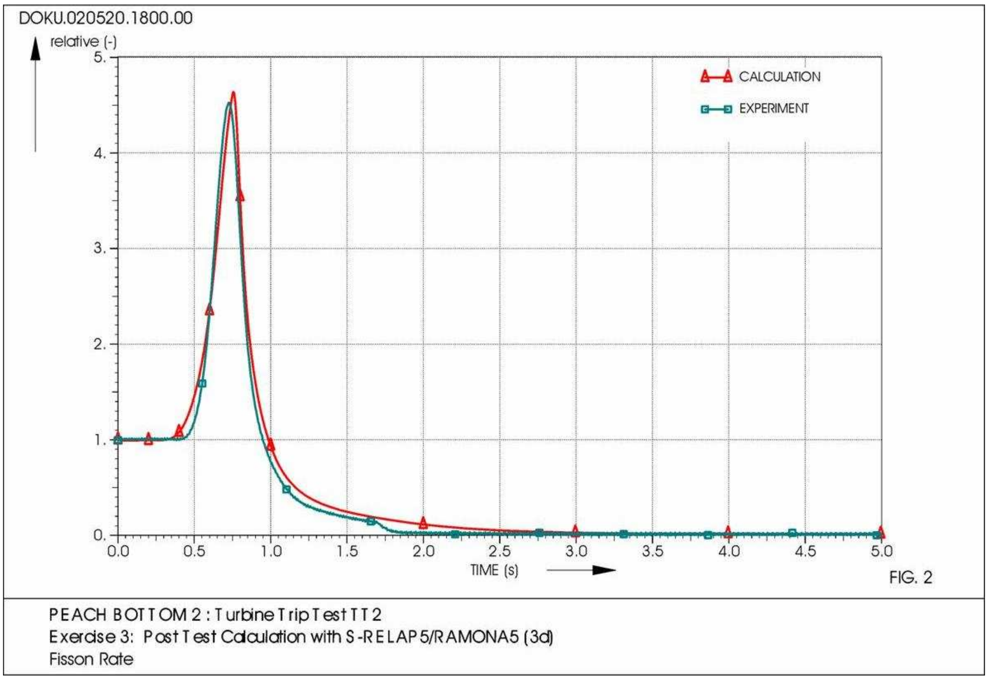

Figure 10: Measured and calculated reactor power in the Peach Bottom turbine trip benchmark

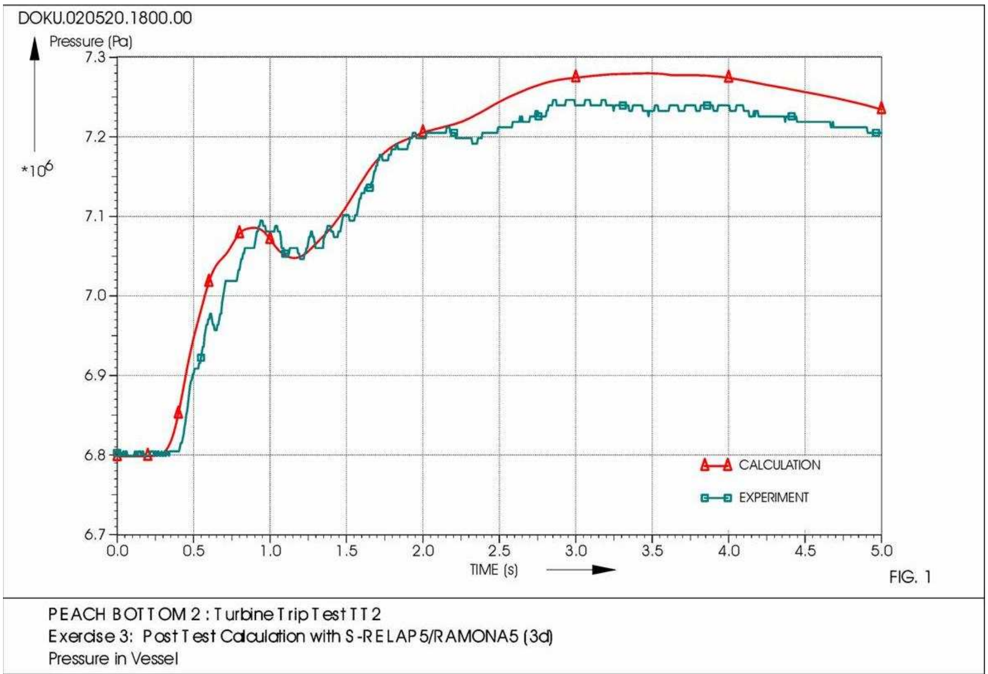

Figure 11: Measured and calculated reactor pressure in the Peach Bottom turbine trip benchmark

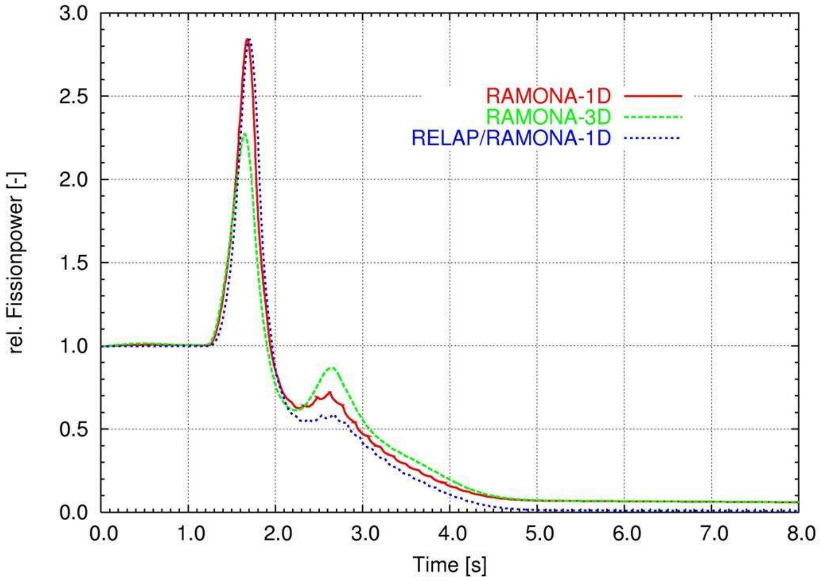

Figure 12: Comparison of the transient reactor power during the loss of the main heat sink with 1D and 3D representation of the reactor core.

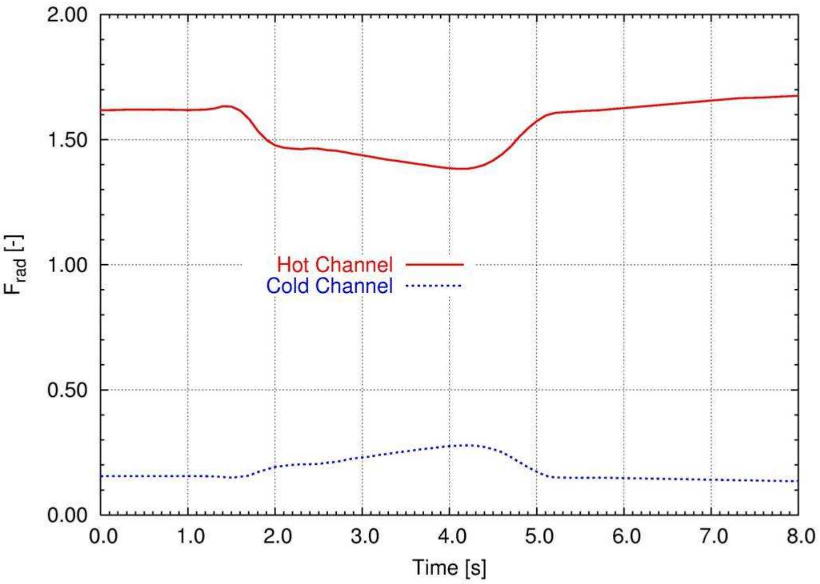

Figure 13: Transient values of the radial power factor during the loss of the main heat sink with a 3D representation of the reactor core.

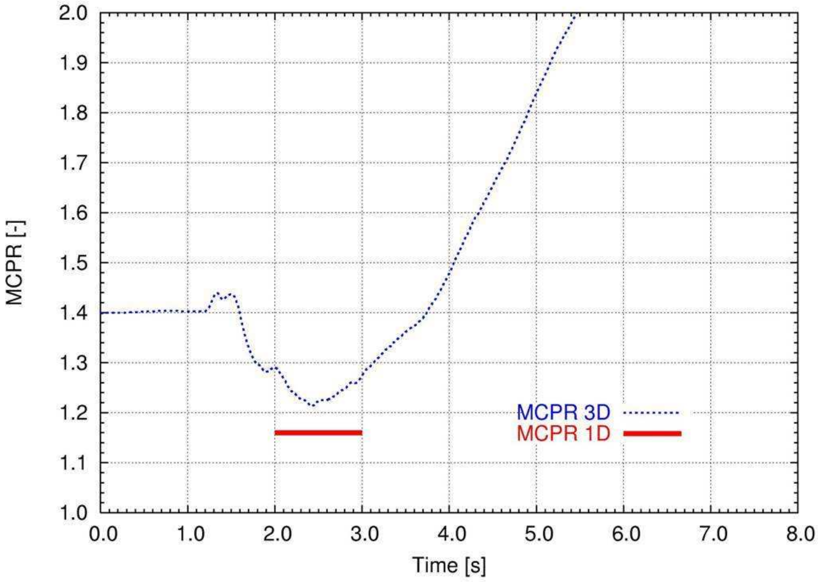

Figure 14: Transient minimal critical power ratio (MCPR) during the loss of the main heat sink with a 3D representation of the reactor core in comparison with the value out of 1D calculations.

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References

[1] St. Misu, H.D. Kiehlmann, H. Spierling, F. Wehle, H. Moon and R.G. Grummer, "The Comprehensive Methodology for Challenging BWR Fuel Assembly and Core Design Used at Framatome ANP", PHYSOR 2002, Seoul, Korea, October 7-10.

[2] L. Moberg and J.P. Rea, "BWR Space Time Kinetics Applications with RAMONA", Proceedings of ANS 1990 Annual Meeting, Nashville, Tennessee, June 10-14.

[3] H. Chow, T.H. Chen and D.C. Kolesar, "Assessment of S-RELAP5 for Realistic Analysis of PWR Large Break LOCA", 4th International Topical Meeting on Nuclear Thermal Hydraulics Operation and Safety, Taipei, Taiwan, April 6-8.

[4] F. Wehle and W. Kraemer, "Siemens-KWU Experience on BWR Stability Analysis", European Nuclear Conference (ENC'90), Lyon, France, September 23-28, 1990.

[5] D. Kreuter and F. Wehle, "Siemens-KWU Experience on Linear and Nonlinear Analyses of Out-of-Phase BWR Instabilities", TOPFUEL 1995, Würzburg, Germany, March 12-15, 1995.

[6] S. Mojumder, D. Kreuter, S. Opel, R. Velten, F. Wehle, M. Wickert and R. Zimmermann, "Siemens Stability Methodology", Fachtagung der KTG-Fachgruppen 'Thermo- und Fluiddynamik' und 'Reaktorphysik und Berechnungsmethoden', Rossendorf, Germany, January 31 / February 1, 2000.

[7] F. Wehle et. al., "BWR Stability Measurements and Methodology", TOPFUEL 2001, Stockholm, Sweden, May 27-30, 2001.

[8] F. Wehle, A. Schmidt, S. Opel and R. Velten, "Non-linear BWR Stability Analysis", ICONE 2008 May 11-15, 2008, Orlando, Florida, USA accepted

[9] G.M. Grandi and L. Moberg, "Application of the Three Dimensional BWR Simulator RAMONA-3 to Reactivity Initiated Events", Topical Meeting on Advances in Reactor Physics, Knoxville, Tennessee, 1994.

[10] D. Bender, H. Bauer,F. Wehle, D. Kreuter, H. Finnemann, J.L. Maryott and R.A. Copeland, "Methodology and Results of RIA Studies at Siemens", OECD Specialist Meeting on Transient Behaviour of High Burnup Fuel, Cadarache, France, September 12-14 1995.

[11] L. Heins, "Representative Core Calculations of RIA for BWR", Proceedings of the Topical Meeting on RIA, Aix-en-Provence, France, May 13-15 2002.

[12] R. Velten, B. Karrasch, F. Wehle and S. Franz, "Peach Bottom 2 Turbine Trip Benchmark: Results Obtained with Framatome ANP's Code System S-RELAP5/RAMONA5", Annual Meeting on Nuclear Technologie, Berlin, Germany, May 20-22 2003.

[13] M. Wich et. al., "Critical Heat Flux Tests in the KATHY Test Loop", TOPFUEL 2003, Würzburg, Germany, March 16-19, 2003.

[14] F. Wehle, S. Opel and R. Velten, "Advanced Methods for BWR Transient and Stability Analysis", Fachtagung der KTG-Fachgruppen 'Betrieb' und 'Reaktorphysik und Berechnungsmethoden', Rossendorf, Germany, February 13-14.