SUCCESSFUL DEPLOYMENT OF FRAMATOME ADVANCED PWR CODES AND METHODS WORLDWIDE

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ABSTRACT

Framatome developed a new generation of codes (ARCADIA, GALILEO) that offers 3D coupled neutronics/ thermal-hydraulic/ thermal-mechanical calculations using a modern architecture that supports nodal and fine mesh calculation capability. To justify the coupling approach a set of transient benchmarks were performed with comparisons to analytical and operating plant data. These codes are being integrated into an advanced execution platform to offer an efficient framework for methodology development and innovative engineering services: ARITA (ARTEMIS/S-RELAP5 integrated transient analysis), AREA (ARCADIA Rod Ejection Accident) methodology, CRUD analysis, Spent Fuel pool management and vessel fluence calculation are examples of these new functionalities. The new codes and methods provide margin to key parameters, expanded fuel cycle design space for a cost saving approach to optimize reactor operation. This paper presents an update of the current licensing and implementation status and illustrates their accuracy through benchmark results.

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1.   Introduction

Anticipating customer's needs for the future, Framatome took up the challenge to significantly advance the state-of-the-art in reactor physics. A series of large Research and Development programs was started to prepare the next generation codes and methods capable to handle new challenges in fuel assembly design, core design, safety analysis and core monitoring. These development programs are now reaching maturity and are being deployed to different customers.

2. The main developments axis

The Framatome Codes & Methods development program covers the domains of:

- Advanced thermal-mechanics: Development of GALILEO, Framatome state-of-the-art fuel rod performance code for pressurized water reactor (PWR) applications
- Advanced neutronics: Development of ARCADIA suite, Framatome state-of-the-art neutronics/ thermal-hydraulics (TH)/ thermal-mechanics code system for PWR and BWR application
- Advanced thermal-hydraulics: Development of COBRA-FLX and F-COBRA-TF, Framatome advanced TH sub-channel code respectively for both PWR and BWR
- Advanced numerical simulation: Use state-of-the-art tool (Computational Fluid Dynamics (CFD), deep learning algorithms) to improve physics understanding and solve complex in-core operations.

2.1 Thermal-mechanics

GALILEO [1] is an advanced fuel rod performance code which integrates the best models existing in the legacy Framatome codes COPERNIC2, CARO-E and RODEX4. Special attention is given to support for new products like Cr doped pellets and high burnup fuel. Deterministic as well as statistical methodologies for calculating fuel rod performance were part of the GALILEO development.

One of the pillars of GALILEO development is the extent and the representativeness of its database in terms of designs, power load and burnup conditions. Framatome owns a huge unprecedented database, having benefited from the merging of the databases used for CARO-E, COPERNIC2 and RODEX4 codes. It includes measurements of temperature, fission gas release, dimensional changes…with a maximum rod burnup exceeding 100 GWd/tU.

Once a strong database is available and operational, the modeling stage arises. The physical phenomena involved in a fuel rod are so numerous and so interconnected that a fine compromise has to be reached between a detailed description of the physics, the computational processing time capabilities of current computers, and the needs of the designer. In particular, customers demand for higher discharge burnups implies some special modeling effort concerning high burnup phenomena. The microstructure and the composition of nuclear fuel at high burnup are different from those of the as-fabricated fuel, especially in the pellet rim region where grain polygonization occurs, resulting in a high burnup structure formation. It significantly affects fuel thermal conductivity, porosity and swelling, for which modeling changes are described.

The GALILEO thermal conductivity is composed of a classical phonon term plus an electronic term important at high temperature. The burnup degradation as well as the radiation damage were calibrated by matching the GALILEO centerline temperature predictions with the closed gap centerline temperature measurements from high burnup experiments, up to ~100 GWd/tU. Additives such as Gd, Pu or Cr affect the phonon interactions and are modeled in a consistent manner by adding a term to the inverse of the conductivity phononic part, which is proportional to the additive content. Beside the fuel thermal conductivity, the thermal chain is classically composed of the usual sub-models of gap conductance, gap closure, cladding thermal conductivity, corrosion layer, tabulated radial power profiles and rod-coolant heat transfer. The gap closure is due to cladding creep down, fuel densification, swelling, and a thermal relocation. This latter is modeled as fragment outward shift due to cracking of pellets. The models were globally benchmarked with over 3300 fuel centerline temperature data points. An important correction to conductivity concerns the degradation due to porosity. The local porosity decreases due to densification at beginning of life but increases thereafter due to gaseous swelling at the center of pellet and due to High Burnup Structure at the rim. An original model was introduced into GALILEO to describe this fuel restructuring.

The migration of the fission gas atoms in the fuel matrix is also affected at high burnup, and the result is that the fission gas release (FGR) is enhanced. The new extended fission gas release model in GALILEO is based on the classical two-step diffusion model (Booth model) following the 3-term formalism of Turnbull for the diffusion coefficients, and which in addition

to the steady-state thermal release, also describes transient effects and a thermal release. The thermal model takes into account grain boundary incubation and saturation and irradiation induced resolution from the grain boundaries, which counteracts the diffusion flux and delays the onset of release. The transient model handles additional burst effects for relevant and rapid power changes. Future higher burnup and duty applications beyond about 65 GWd/tU were the focus of the model extension. Experimental feedback shows potential FGR enhancement with burnup, which can be described by introduction of new mechanisms. The observations led to the introduction of a xenon concentration limit in the fuel matrix which is dependent on local fuel burnup and temperature.

A mechanical model has been implemented in GALILEO consisting of a 1-D classical finite element method to solve the mechanical problem, for which innovative algorithms have been added to combine a pellet cracking model, a plasticity model for cladding and a creep model for pellet and cladding. The rod is considered axis-symmetric during the whole power history, with a monodimensional discretization in the radial direction of cylindrical coordinates. Stresses and strains are obtained as solutions of the equilibrium balance and compatibility relations with the proper boundary conditions. Perfect contact is assumed between pellet and cladding. The introduction of new modeling of complex features (such as visco-plasticity and pellet cracking) is unavoidable in order to assess the future advanced fuel products. From a software point of view, the counterpart of modeling such non-linear phenomena is developing robust algorithms and carefully revises the convergence management in the code. Some numerical solutions have been proposed to meet the needs of a fast calculation, code robustness and accuracy of the predictions.

2.2 Neutronics

ARCADIA [2] is an advanced 3D coupled neutronics/thermal-hydraulics/thermal-mechanics code system for steady-state and transient Fuel Assembly (FA) and core design. ARCADIA encloses the sub-systems (1) spectral code, (2) core simulator and (3) peripheral services (Fig 1.).
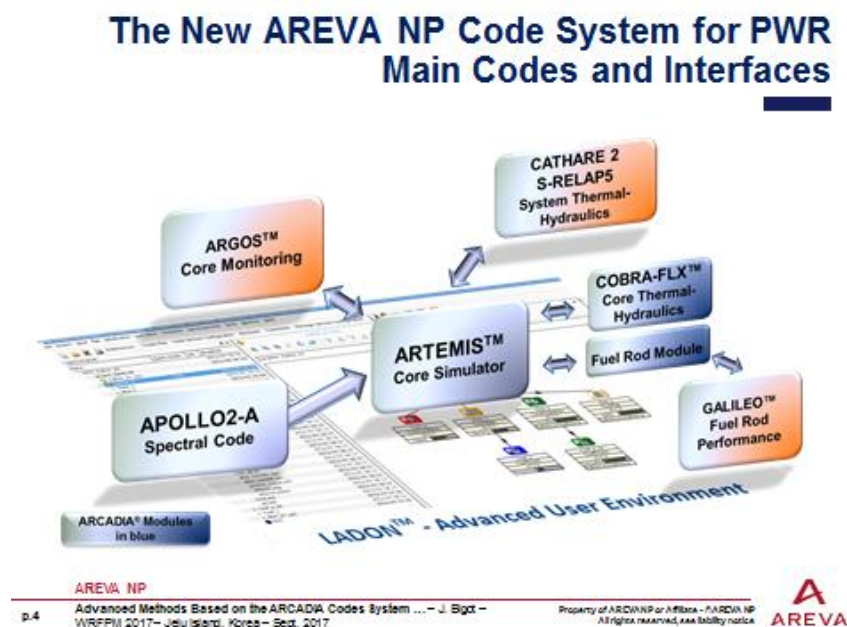

Fig 1. ARCADIA architecture (main module)

The spectral code system (called APOLLO2-A) [3] was developed in close cooperation with the French Nuclear Research Center, Commissariat à l'Energie Atomique (CEA). It is a further development of the CEA code APOLLO2. The newest nuclear data library JEFF3 is used. An extensive set of qualification cases were evaluated covering a large range of LWR applications as part of the ARCADIA development.

The core simulator ARTEMIS [4] [5] and [6] represents an advanced 3D coupled neutronics/

thermal-hydraulics/thermal-mechanical code system for steady-state and transient core design with new software architecture allowing for nodal and pin-by-pin calculation capability. ARTEMIS includes also among other high performance multi group diffusion and transport flux solver and has been approved by the US NRC for PWR application. Development is ongoing to extend it to BWR applications.

Peripheral services system covers input generation and pre-processing as well as output evaluation and post-processing. It includes graphical user interfaces, job automation and post-processing tools needed to facilitate fuel assembly and core design, e.g. statistical evaluation, graphics and automated report generation tools.

ARGOS [7] is Framatome's new state-of-the art core monitoring system which offers highly accurate power distribution monitoring and Technical Specifications surveillance for all types of commercial light water reactors. As a completely modular universal platform implementing consistently modern open software standards, ARGOS can utilize different neutronic simulators and a variety of proven and innovative power reconstruction and adaptation methodologies. Besides supporting general operator assistance and a wide set of specific functions, ARGOS features a highly flexible and powerful prediction module assisting in precise planning of projected power maneuvers. After having been tested successfully in parallel runs, ARGOS now enters the final phase of commercial market implementation.

Besides state-of-the-art physical modeling, numerical performance and industrial functionality, the ARCADIA system features state-of-the-art software engineering. A decisive achievement in this regard is the implementation Framatome's new ARTEMIS/S-RELAP5 integrated transient analysis (ARITA) methods package (see chapter 4 and [8] for more details).

The use of the ARCADIA-1 code system for PWR core performance analysis has been approved by US NRC in 2013. In 2015 a supplement to the approved Licensing Topical Report (LTR) was submitted to the U.S. NRC for review. The aim was to (1) cover code modifications (e.g. related to physical changes in cross section representation in the spectral code APOLLO2-A), (2) introduce new functionalities to meet requirements of new methodologies, (3) specify additional criteria for code validation and (4) reduce uncertainty on total control rod worth and moderator temperature coefficient. The LTR Supplement is close to get the U.S. NRC approval.

2.3 Thermal-hydraulics

Thermal-hydraulic subchannel analysis represents a major part of the FA design and development. The R&D strategy in this respect is two-fold [9]:

- Standard TH analysis: For standard applications requesting high computational speed the subchannel code COBRA-FLX has been integrated into the ARCADIA environment for PWR calculations. The code was developed based on the physical models of COBRA-3CP and integrating best functionalities of FLICA III, XCOBRA-T, LYNXT. It is a completely new code, with high computational performance allowing for 3D full core transient and steady-state calculations.
- Advanced TH analysis: In the past few decades the need for improved nuclear safety analysis led to a rapid development of advance methods for multidimensional TH analysis. The advanced TH subchannel code COBRA-TF is used worldwide for best-estimate evaluations of the nuclear reactor safety margins. In the frame of a joint research project between Pennsylvania State University (PSU) and Framatome, the theoretical models and numerics of COBRA-TF have been improved. Under the name of F-COBRA-TF, the code has been subject to an extensive verification and validation program. F-COBRA-TF is now an advanced and state-of-the-art tool used in Framatome mainly for BWR industrial applications.

2.4 Advanced numerical simulation

After a decade of investment into state-of-the art tool providing more accurate assessment of margin to safety limits and enhanced operational and design margin, the next R&D effort will be done toward advanced numerical simulations. Several areas of improvement are being under investigation and include:

- Digital twin: Develop CFD tools [10] based on FA model and full core model for normal and accident conditions
- Use of emerging technology domain: Develop method based on advanced analytical technics (e.g. meta-modelling, artificial intelligence, big data and neural networks).

The objective is to provide on-line support to plant operation via numerical simulation with predictive capability and Optimization of plant operational strategy.

3. Benchmark results

Numerous benchmarks have been performed to demonstrate the accuracy of the new code platform: international code and analytical benchmarks include (TWIGL, NEACRP Rod Ejection, NEA/NSC Bank withdrawal…), analysis of measurements include (SPERT rod ejection, customer transient, etc.) and code-to-code comparisons. Two benchmarks are highlighted in this article:
- A loss of load
- A pump shaft break event

The first benchmark is a planned loss of external load to station service performed at end of cycle in stretch-out conditions at a German 1300 MWe plant. A loss of load results in a nearly instantaneous decrease of the generator power to the level required for station service (~3 % of nominal power). The Instrumentation and Control (I&C) system detects the loss of load due to a significant mismatch between the reactor and the generator power and automatically reacts by reducing the core power through a series of sequential rod drops to reduce core power to approximately 30 % of the rated thermal power. Following the control rod drops power level is stabilized to match the station load demand using normal steady state controls. For validation purposes only the initial portion of the transient up to 120s are of interest.

As this event was a planned test, a large set of measured data is available. The data includes the core power and all in-core self-powered neutron detectors and ex-core detector signals as well as primary and secondary system parameters like coolant and steam pressures, valve positions and steam generator levels. However, some of the measured signals are affected by a 'dead-band' effect, limiting the time resolution. In order to benefit from this comprehensive data set, the recalculation was performed using the 'system-coupled' application mode. Thus, a nodal calculation with the ARCADIA core simulator ARTEMIS coupled to a plant thermal-hydraulics and system calculation using S-RELAP5. S RELAP5 uses the code NLOOP to simulate the plant-specific I&C functions.

The initial core state was calculated with ARCADIA. The initial state in S-RELAP5 was adapted to the stretch-out conditions of the measurement. The S-RELAP5 model assumes homogeneous mixing in the lower plenum. As the turbine is not simulated in the model, the initiating event of the transient in the calculation is a forced reduction of the generator power to the level required for station service, at t=20 s. The effect of the turbine over-speed on the coolant pumps is simulated as a forcing function in S-RELAP5. Otherwise plant and core responses are completely simulated by the coupled ARTEMIS/S-RELAP5 system. The measured and the calculated short term corrected reactor power are compared in Figure 2. The short term corrected reactor power is a signal derived from the ex-core detector signals (calculated by ARTEMIS) and the coolant temperatures (calculated by S-RELAP5). It shows the power reductions due to the control rod pair drops initiated by the I&C system (simulated by NLOOP). Thus, it is a well suited indicator for the quality of the recalculation, as it is derived from values determined by all component codes of the coupled system. In the figure the sequential control rod pair drops beginning directly after the initiating event are clearly visible. After the drop of the last rod pair the power increases slightly due to the moderator and fuel temperature feedbacks. The calculated and measured 'short term corrected reactor power' agree very well.
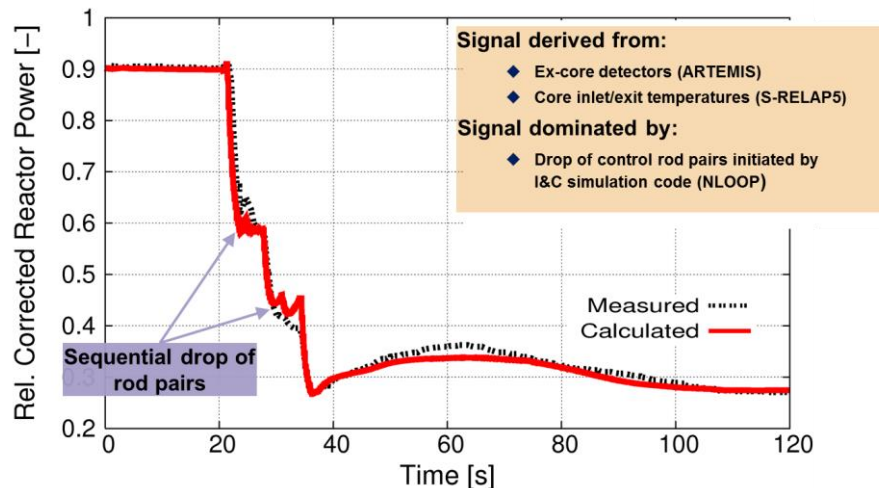
Fig 2. ARCADIA/S-RELAP5 Transient Validation: Load Rejection

The second benchmark is a pump shaft break event: a pump shaft of one of the three main coolant pumps broke, separating the pump rotor from the electric drive that continued to run. As a consequence the pump failure was not detected via the drive speed criterion of the I&C system of the plant and no reactor trip was triggered. It took approximately two minutes after the beginning of the transient to trip the pump based on a high bearing temperature indication. This pump trip triggered the drop of selected control rods to reduce the reactor power to approximately 30 %. The pump shaft break led to a quick reduction of the coolant mass flux to a level of about two thirds of the nominal value. Consequently the coolant temperature increased resulting in a power reduction to about 70 % of nominal power through the moderator temperature feedback. The power decrease leads to a decrease in fuel temperature, so that after a few seconds the fuel temperature feedback resulted in a moderate increase in power until it stabilized at about 75 to 78%. Due to the relatively late response of the I&C system, the first few seconds of the transient are exclusively governed by the physics of the reactor, making this transient a valuable validation case. As the event was not planned in advance, the available measured data is limited. For transient validation the first ten seconds of the event are evaluated using an ARTEMIS stand-alone nodal calculation. The initial burnup state in cycle 6 is calculated with ARTEMIS. The following thermal-hydraulic boundary conditions are used for the transient:

- The initial coolant temperature at the core inlet (294 °C) is available from the measurement. During the recalculated initial ten seconds of the transient it is assumed to remain constant.
- The transient evolution of the primary system pressure is available from the measurement and is used as a forcing function in the calculation.
- The initial coolant mass flow is taken from the cycle design calculations.
- The transient evolution of the mass flow is not available in the measurement. It was derived from a simulation of the pump behavior under the assumption of a shaft break. The resulting data is used in ARTEMIS as a forcing function.

The core power and the simulated in-core SPND signals are compared to the measurement data. The transient thermal core power can be predicted well with ARTEMIS. The two measurement curves were derived from independent signals and therefore give an idea of the measurement uncertainty. The fact that the ARTEMIS results are located between these two curves indicates good agreement with the measurement. Calculation and measurement for a representative example of the in-core SPND signals are compared in Figure 3.
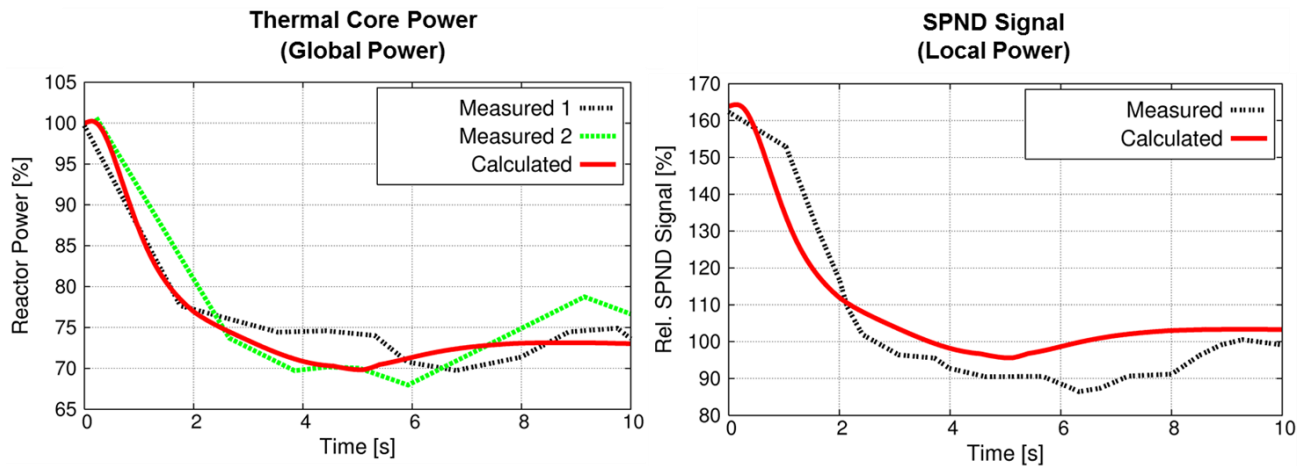
Fig 3. ARCADIA Transient Validation: Pump Shaft Break

4. Implementation benefits

Implementation benefits of Framatome's advanced codes and methods were addressed in detail in [11] and [12]. Recent results [8] are showing increased margin for a challenging subset of non-LOCA transient events. Application of Framatome's advanced codes and methods to accident analysis demonstrates the capability to extend the fuel cycle design space available for a given plant. This enables in principle improved fuel cycle economics [11].

The collection of design limitations that typically reduce flexibility in the cycle design process are imposed by a multitude of stakeholders including the utility, the regulator, and the fuel vendor. Together, these limitations define a design space; a given cycle design must "fit" within this space to get accepted. Assessment of a given plant for a typical cycle design would provide the needed information to determine the plant's current position within the acceptable design space. This concept is illustrated in Figure 4 where the normalized acceptable design space is the area shaded in scarlet, the current position of a virtual plant design within this space is shaded in grey, and the distance from the borders of the grey region to the borders of the scarlet region represents available margin or flexibility relative to each design limitation. The benefits of Framatome new codes and methods can be used to increase the design and operating space.
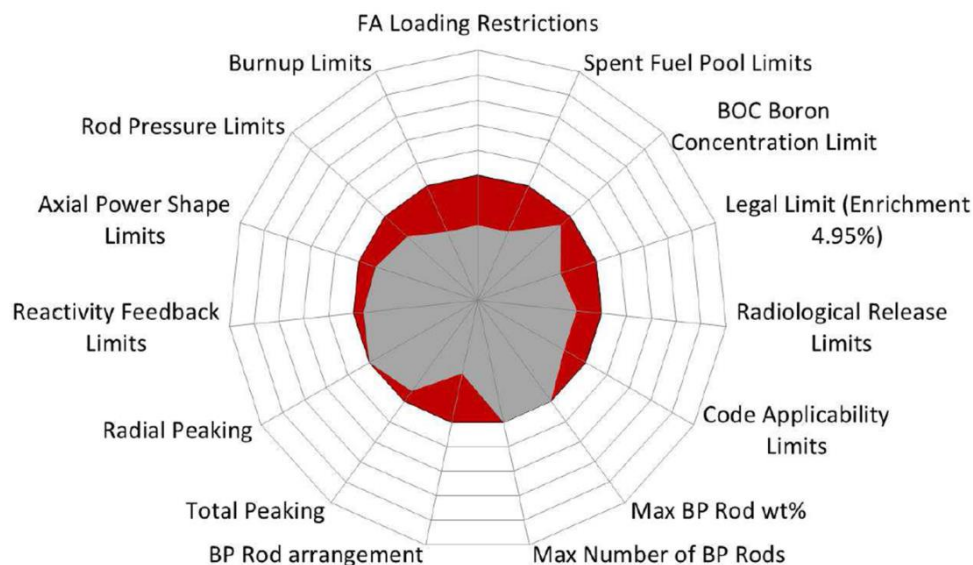

Fig 4. Normalized Design Space (scarlet) and a representative plant position within Normalized Design Space (grey)

[BOC= Beginning Of Cycle, BP=Burnable Poison, FA= Fuel Assembly]

Application of 3D coupled transient methods provides the ability to meet emerging U.S. NRC requirements for RIA or for non-LOCA transients and can remove "obstacle" transients that may restrict core designs and/or impede power upgrades. Several examples have highlighted the benefits of the new AREA methodology [13] for Rod Ejection Accident or the ARITA method for non LOCA transient [8].

For rod ejection (Fig 5.) it has been shown that the maximum enthalpy rise at any burnup is considerably reduced and the enthalpy rise of the different cladded fuels is known at its respective burnups. In addition, significant margin exists to the limits to allow for more demanding core designs for the ejected rod accident and/or extra biasing to minimize cycle specific analysis costs.

For non-LOCA event (Fig 6.), when ARITA is used for plant safety analysis, the resulting margins for the transient events shown are significantly increased compared to the one obtained from traditional deterministic methods where the core response is based on a point kinetics model with conservative core models. This additional margin can be used by customer to get more operational flexibility.
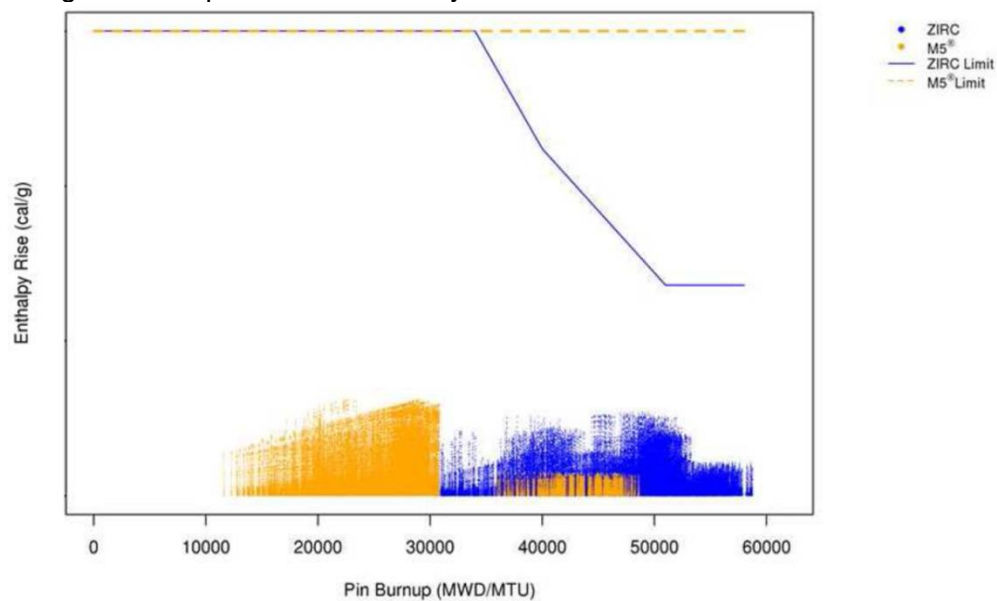

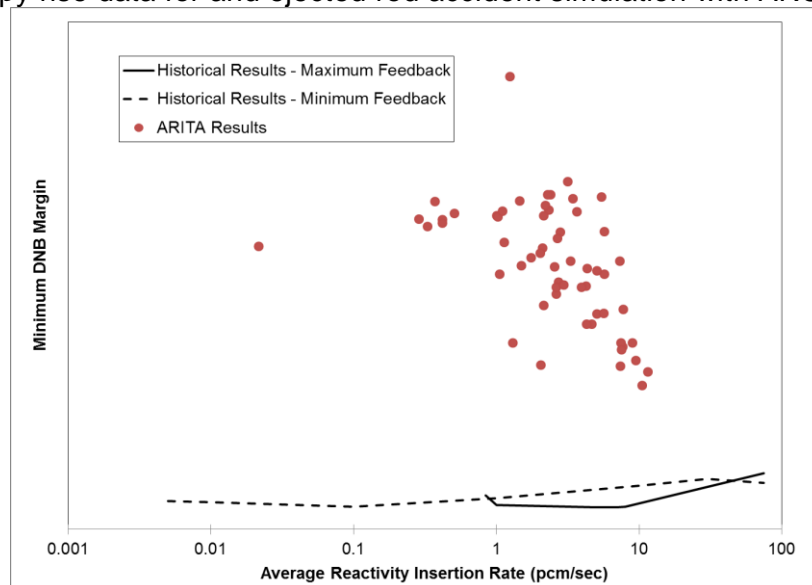Fig 5. Enthalpy rise data for and ejected rod accident simulation with ARCADIA from [1]

Fig 6. Comparison of ARITA Minimum DNB Margin to Traditional Methods for the Uncontrolled Control Rod Bank Withdrawal Event [8]

5.   Conclusion

After a phase of intensive R&D investment in reactor physics, thermal-hydraulics and thermal-mechanics, Framatome has accomplished the development of a state-of-the-art codes in all of these domains and is now implementing a new generation of advanced methodologies for fuel assembly design, core design, safety analysis and core monitoring. Advanced features such as 3D coupled transient core power distribution characterization allow more realistic modeling of plant behavior and removal of overall uncertainty by high resolution and access to local information.

The new Framatome codes & methods are largely based on first principles modeling with an extremely broad international verification and validation data base. This enables Framatome and its customers to access more predictable licensing processes in a moving regulatory environment (new safety criteria, requests for enlarged qualification databases, statistical applications, uncertainty propagation…). In this context, the advanced codes & methods and the associated verification and validation represent the key to avoiding penalties on products, on operational limits, or on methodologies themselves. Advanced and integrated methods (coupling of the relevant physical domain) are keys to maintaining overall safety margin and improving operating margin.

5.   References

[1] Strumpell, J.H. et al., "Application of GALILEO: AREVA NP's Advanced Fuel Performance Code and Methodology", Proceedings of 2013 LWR Fuel Performance Meeting / Top Fuel, Charlotte, North Carolina, U.S.A.,September 15-19, (2013).

[2] Curca-Tivig, F. et al., "ARCADIA- A New Generation of Coupled Neutronics/ Core Thermal- Hydraulics Code System at AREVA NP", Proc. of the 2007 Int. LWR Fuel Performance Meeting, San Francisco, California, Sept. 30 – Oct. 3, (2007).

[3] Martinolli, E. et al., "APOLLO2-A – AREVA NP's New Generation Lattice Physics Code: Methodology and Validation", Proc. of. PHYSOR 2010, Pittsburgh, Pennsylvania, U.S.A., May 9-14, (2010).

[4] Pautz, A. et al., "The ARTEMIS Core Simulator: A Central Component in AREVA NP's Code Convergence Project", Joint Int. Topical Meeting on Mathematics & Computation and Supercomputing in Nuclear Application, (M&C + SNA2007), Monterey, California, April 15-19, (2007).

[5]. Hobson, G. et al., "ARTEMIS: The Core Simulator of AREVA NP's next Generation Coupled Neutronics/ Thermal-Hydraulics Code System ARCADIA□", PHYSOR 2008, Interlaken, Switzerland, September 14-19, (2008).

[6] Hobson, G. et al., "ARTEMIS Core Simulator: Latest Developments", Joint International Conference on Supercomputing in Nuclear Applications and Monte Carlo 2013 (SNA + MC 2013), Paris, France, October 27-31,(2013).

[7] Paul T., ARGOS – Implementation of Framatome's core monitoring system on the European market, Proc. TopFuel LWR Reactor Fuel performance, Prague, September 30-October 4, 2018

[8] W. Walter et al, ARTEMIS / RELAP5 Integrated Transient Analysis application to non-LOCA transients,., Proc. TopFuel LWR Reactor Fuel performance, Prague, September 30- October 4, 2018

[9] Leberig, M. et al., "AREVA NP's Advanced Thermal Hydraulic Methods for Reactor Core and Fuel Assembly Design", Top Fuel 2009, Paris, France, September 6-10, (2009).

[10] J. Dumond et al, Industrial applications of Framatome's state-of-the-art CFD methods to nuclear reactor analysis,., Proc. TopFuel LWR Reactor Fuel performance, Prague, September 30- October 4, 2018

[11] K. Maupin et al., AREVA's ARCADIA Code System – Implementation Benefits, Top Fuel 2015, Zurich, Switzerland, September 13-17 , 2015.

[12] F. Curca-Tivig, ARCADIA and Advanced Methods Licensing and Implementation Update, Top Fuel 2016, Boise, Idaho, U.S.A., September 11-15, 2016.

[13]. AREVA, "AREA - ARCADIA Rod Ejection Accident Topical Report," ANP-10338NP, Oct 2015.