USAGE OF ARCADIA CODE SYSTEM FOR NEUTRONIC AND THERMAL-HYDRAULIC CORE ANALYSES TO SUPPORT THE CRUD RISK ASSESSMENT OF A THREE-LOOP PLANT

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ABSTRACT

The purpose of this paper is to show the advanced capabilities of Framatome's converged state-of-the-art code system ARCADIA to perform 3D full core analyses in the frame of a Crud Risk Assessment (CRA). The core simulator ARTEMIS with the TH module COBRA-FLX, part of the ARCADIA code system, was used for calculating the necessary core input data at pin-by-pin resolution for the plant chemistry analysis. The usage of state-of-the-art codes and methods together with appropriate tools and interfaces allows for performing a CRA in an efficient and effective manner. Selected results for a 900 MWe, 3-loop plant constitute part of the necessary input data for the final CRA.
Framatome is now able to offer a level III CRA based on the code system ARCADIA to all utilities in the worldwide market.

1 Introduction

Crud is defined as the deposit of corrosion products on fuel cladding in PWRs. Two main issues are associated with the formation of crud: Crud-Induced Localized Corrosion (CILC), which can lead to fuel failure events, and Crud-Induced Power Shift (CIPS), which can lead to erosion of shutdown margin and loss of operational flexibility.

Guidelines of both INPO and EPRI require US utilities to perform a Crud Risk Assessment (CRA) under specific conditions. [1][2] A CRA addresses neither design criteria nor safety analyses. Because of today's electricity market conditions, an economical operation of NPPs requires much higher flexibility than in the past; hence utilities are more and more considering the added value and benefits of a CRA within their cycle design process, also to avoid the additional costs due to fuel leakers.

EPRI defined four CRA levels. [2] Framatome developed a comparable four-level CRA for assisting utilities in the efficient and effective implementation of these requirements for a given cycle. The hierarchy of the levels is going from plant operation review in level I to detailed CFD-based CILC evaluation on identified fuel rods in level IV. The choice of the level depends on the type of changes envisioned by the utility, for example fuel cycle design, water chemistry program and/or steam generator replacement. This enables utilities to benefit from the most suitable and cost effective CRA. Details on Framatome CRA techniques can be found in numerous papers. [3][4][5][6][7][8]

For instance, the Framatome level III CRA process has been successfully applied to units from various NSSS vendors. To date, there have been no plant performance issues that have been outside the performance as predicted by the Framatome risk assessment tools. The scale of the level III assessment is on the order of a subchannel, typically 1.5 cm-by-1.5 cm (or smaller) and 5 to 7 cm axial length. The core analysis portion is based on clean fuel rods, i.e. fuel rods with no crud deposition. This level of detail couples detailed neutronics and TH models together with plant chemistry analysis to evaluate the following:

1. The risk of CILC, including the predicted corrosion resulting from lithium uptake into the zirconium oxide layer of fuel.
2. The estimated magnitude of CIPS resulting from boron accumulation in the fuel deposits.

2 The ARCADIA code system

2.1 Overview

ARCADIA is an advanced 3D coupled code system for steady-state and transient applications. [9][10][11] The U.S. NRC approved the use of ARCADIA code system for PWR core performance analysis in 2013. [12] The code system ARCADIA has been validated for a large number of reactor cycle conditions covering many of the PWRs served by Framatome as well as many transient cases. [13][14][15]

The new software architecture allows for both nodal and pin-by-pin calculations. ARCADIA includes the following main sub-systems:

- The spectral code APOLLO2-A designed for lattice physics calculations and developed in close cooperation with CEA. [16] It is suitable for stand-alone analyses, such as fuel assembly design studies, as well as for generating multi-group neutron libraries required by the core simulators. It uses the latest nuclear data library JEFF3.1.1.
- The 3D core simulator ARTEMIS unifies Framatome most advanced computational methodologies and provides state-of-the-art analysis capabilities. [17] It includes high performance multi-group diffusion and is designed for PWR and BWR applications. The capabilities to perform 3D coupled steady-state and transient calculations are an asset for development of high accuracy safety analysis methodologies. Its transport and pin-by-pin geometry architecture enable it to remain state-of-the-art for the long term. Recent validation experience of ARTEMIS for transient modelling is provided in [14][15].
- The core TH code COBRA-FLX [18] is fully integrated in ARTEMIS as its TH Module (THM), and has the capability of performing 3D steady-state and transient analyses with full-core subchannel-by-subchannel spatial discretization. This way, complex two-phase flow problems can be solved and access to local physical parameters is made possible.
- The Fuel Rod Module (FRM) of ARTEMIS solves the nonlinear heat transfer equation for a cylindrical rod, both in steady-state and transient conditions, to evaluate the fuel temperature for the Doppler feedback in fuel pins. It is fully consistent with the fuel rod thermal properties of the advanced fuel rod performance code GALILEO. [19] The FRM solution also provides heat flux and clad surface temperatures to the TH code COBRA-FLX for both nodal and sub-channel by sub-channel geometries.
- The graphical user environment LADON covers input generation and pre-processing, output evaluation and post-processing. It includes a graphical user interface, job automation and many kinds of post-processing tools needed for fuel assembly and

core design. It is also the frame for setting up and automating methodologies for any kind of applications.

2.2 ARTEMIS

2.2.1 Crud boron modelling

The experience feedback gained on the CRA of US plants with a previous Framatome neutronics code [20] resulted in the development of a highly automated “crud model” which has been implemented in ARTEMIS.

An input group is specifically dedicated to the preparation of a neutronic calculation for crudded rods. Required user inputs are:

- axial profiles (1D) of the shape of the boron content;
- 2D radial map of the boron load per fuel assembly;
- a self-shielding factor for B10 in the crud.

The combination of the axial profiles and the radial map allows a 3D representation of the boron distribution due to the crud to estimate the AO deviation.

The crud boron is modelled in ARTEMIS as a node dependent addition to the soluble boron in the water. For this reason, a B10 self-shielding factor is used to represent the effect of a thin boron deposition on the surface of the fuel rod cladding; that is, the ratio of soluble boron increase required to produce the same reactivity worth as the coating, to the soluble boron increase that would result if all the boron atoms in the coating were to be dissolved in the coolant volume. This ratio is estimated using detailed 2D spectral calculations with the heterogeneous Fuel Assembly (FA) geometry performed with APOLLO-2A.

2.2.2 Checking of the ARTEMIS crud model

The ARTEMIS crud model has been checked with the previous Framatome neutronics code NEMO [20] that has been widely used for CRA in the US market, using existing neutronic analysis to support a recent CRA of a US 17x17 plant.

Table 1 shows the absolute values of the AO deviations calculated by ARTEMIS and by the previous code for hypothetical amounts of crud boron deposited on individual batches of a core.

The good agreement between the AO deviations predicted by the two codes, especially the larger batches N and N-1 which are the most important for the total core AO deviation, checks the correct implementation of the ARTEMIS crud model with respect to the previous Framatome neutronics code. ARTEMIS results are within ±3% of those of the previous code.

	Batch	ARTEMIS	previous code
BOC	N-2	0.1	0.1
	N-1	7.6	7.4
	N	9.0	8.7
EOC	N-2	0.2	0.2
	N-1	9.3	9.6
	N	12.2	12.4

Table 1: AO deviation, ARTEMIS/NEMO comparison

2.2.3 Thermal-Hydraulic and Fuel Rod modelling at pin-by-pin resolution

The COBRA-FLX code is based on COBRA IIIC/MIT-2. [21] Framatome’s own improvements include versatile computational capabilities to cover the full spectrum of TH analyses for both safety and non-safety applications.

COBRA-FLX contains a specialized solution algorithm that allows the subchannel/fuel rod calculations in full core geometry to be performed fast enough to evaluate local conditions for a number of Effective Full Power Days (EFPD) for different cycles without compromising accuracy or computational time. In this way, it is not necessary to restrict the analysis to a 1/8 core avoiding questions on the actual most limiting locations. This is also thanks to the extensive usage in COBRA-FLX source code of OpenMP directives for parallel execution. [7]

For calculating the steaming rate needed for CRA, the total heat flux is modelled in COBRA-FLX as composed of a single-phase forced convection and a nucleate boiling heat flux. [22] The first is obtained with a Dittus-Boelter like single-phase heat transfer coefficient while the latter is based on the Thom correlation. [23] The steaming rate flux (SRF) is finally calculated using the standard formulation of dividing the nucleate boiling heat flux by latent heat of vaporization (h_{fg}), which is pressure dependent.

$$SRF = \frac{\left(\frac{e^{p/1260}}{0.072}\right)^2 (T_{wall} - T_{sat})^2}{h_{fg}}$$

Temperatures are expressed in degrees F, pressure in psia and the latent heat of vaporization in BTU/lb.

When going to full core pin-by-pin analysis, steaming rates are calculated for each individual fuel pin using an azimuthally averaged wall temperature. The fuel pin is the minimal scale used for a level III CRA. It is only when going to a level IV CRA that the wall temperature is allowed to vary azimuthally in the CFD calculations.

To improve the prediction accuracy of COBRA-FLX, the Holloway, Beasley, and Connor (HBC) correlation [24] has recently been included in the available models. The HBC correlation measures the single-phase convective heat transfer coefficients for turbulent flow through the rod bundles downstream of the spacer grids. This correlation is dependent only on the spacer grid pressure loss coefficient (PLC). The implementation of HBC correlation in COBRA-FLX to include the distance from upstream spacer grid model showed better crud deposition prediction downstream of the spacer grids, as discussed in detail in [6].

Within ARTEMIS, having the THM and FRM together in the same code drastically speeds up the change of meshing from nodal (used for depletion calculations) to pin-by-pin (used for local evaluations, as those required for a CRA).

The pin-by-pin input file for FRM describes each fuel rod in the core with its own geometrical and material properties, allowing to accurately model the difference between UOX and gadolinium rods as well as the presence of axial blankets.

The axial meshes used for THM and FRM shall be identical. In the present study an equidistant fine mesh is used.

2.3 AUTOCRUD

The post-processing of the TH analysis uses the automation code AUTOCRUD already described in [6]. The purpose of this dedicated post-processing is to generate a multi-cycle, clean rod, lifetime history for the pair (fuel rod segment, subchannel) in the core at the axial location of interest. The limiting locations, defined on the basis of lifetime integration and using the selection parameters set by the plant chemistry department, are used as an input to the plant chemistry module called the Fuel Deposit Interactive Chemistry (FDIC) for the final CIPS and CILC assessment. Key TH variables for the assessment includes, among others, clad surface temperatures (CST) and SRF, for the fuel loaded in the Cycle N. Core distributions of these variables will be presented in this paper to illustrate the complexity of the analysis.

3 Framatome crud risk assessment (level III) evaluation method based on ARCADIA core modelling

Framatome level III CRA analyses and evaluates the specific physical and chemical nature of the deposits as they develop at the limiting locations over time. This allows a more thorough understanding of the risk associated with the changes being considered. In the Framatome evaluation method, the interaction between the reactor coolant chemistry (e.g. metals concentration, reactor coolant lithium and boron concentrations, $pH_{300°C}$) and localized TH fuel rod conditions -- including SRF, CST and heat flux -- creates deposits which are different in physical characteristics and in chemical composition on a daily level of resolution; thus, the deposits are transformed in time, and especially whenever any of the above conditions change.

The primary advantage with a Framatome level III CRA is that CILC risk is assessed at a localized position in the core rather than as an average core-wide risk; also, the modelling is performed on a daily basis which captures the interaction between coolant chemistry conditions and TH conditions as they change over the anticipated life of the fuel. Defining such a level of resolution is critical in assessing the risk for CILC.

A Framatome level III CRA also estimates the risk of CIPS by considering the boron incorporation in crud deposits at a local core location, especially in an axial direction. The boron accumulation phenomenon is caused by the presence of crud deposits and is enhanced with increased steaming rate fluxes and crud thickness.

Framatome level III risk assessment process, employing the ARCADIA system to model the core both in Neutronics and TH, is outlined in Figure 1.

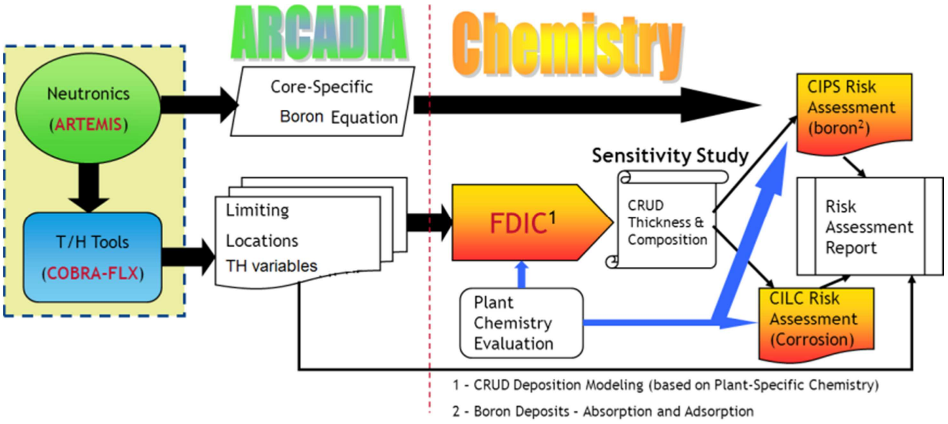

Fig. 1: Overview of the Framatome level III risk assessment process

4 Selected results

4.1 Modelled cycles

Three cycles have been modelled: N, N-1 and N-2 using actual (when available) or estimated core loading patterns. This requires that FAs not older than those of cycle N-2 are loaded in the core for cycle N. The core loading patterns are designed to provide a low leakage core.

Cycles are modelled using core follow data, when available. This includes the modelling of the control rods movement during the cycle. The depletion analysis is not done in all control rods out condition. This provides a best-estimate analysis of the AO change during the cycle.

Several changes in the cycle design between the cycle of interest for the CRA (cycle N) and the previous ones are assumed to happen; those impacting the results presented in this section are listed below:

- Core inlet temperature (which is higher by almost 10 degrees C in cycle N)
- Increased fresh fuel batch size
- Increased cycle length (including stretch-out operation).

4.2 Determination of axial offset deviation due to crud boron

The AO deviation is estimated with ARTEMIS crud model in a parametric study using a 3D crud boron distribution. This input is provided by the Framatome plant chemistry department and the values are proprietary. They include the total mass of boron per FA and the axial distribution of the deposit, which is top peaked since boron deposits in the crud where the steaming rate is higher, that is in the uppermost spans. Crud boron self-shielding factors are predicted for the specific FA design loaded in the core with APOLLO-2A calculations. This parametric study addresses the three loaded batches one at a time, evaluating the impact on the AO of the selected crud boron distribution at different burnup points. The change of the AO deviation is fitted as function of the clean core total soluble boron concentration and constitutes the boron equation which is delivered to the plant chemistry department for the CRA. The boron equation is calculated for the three loaded batches in cycle N, the cycle of interest for the CRA.

Selected results include absolute values of the AO deviation predicted by applying the fitted boron equation for fresh and twice burned fuel batches, shown in Figure 2. The points in Figure 2 can be correlated to EFPD via the clean core soluble boron concentration. The calculated data points in Figure 2 are fitted with quadratic fits as shown. The absolute value of the AO deviation should not be misinterpreted as it does not represent an actual power shift but only the change in AO which would happen in case the selected batch, and only this, has the amount of boron used in the calculations. The actual CIPS or AO Anomaly (AOA) estimation is the outcome of the successive plant chemistry analysis.

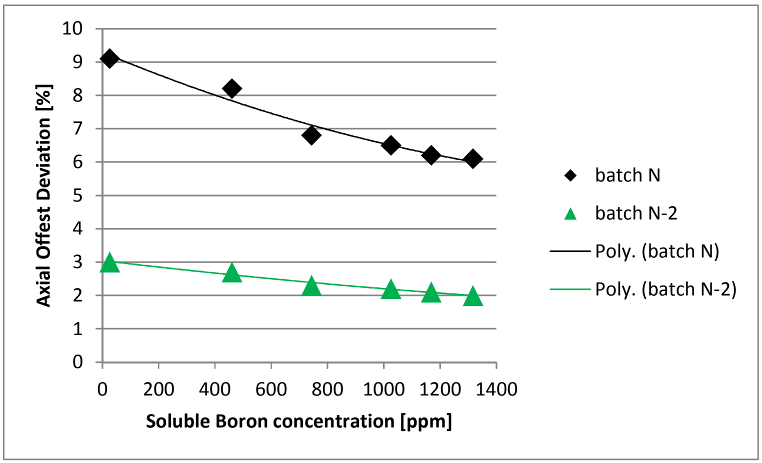

Fig. 2: Absolute value of Axial Offset deviation between clean rods and with boron in crud

4.3 Pin-by-pin TH core analyses

For each cycle (i.e. N, N-1 and N-2) several burn-up points are analysed using a fine mesh pin-power distribution generated by the neutronic solver ARTEMIS. All TH results are based on clean rod conditions. Core follow data are used as the boundary conditions for the core TH analyses, instead of bounding conditions as those used for the safety analysis. This is necessary to provide BE, local TH conditions to the plant chemistry department.

First a core-wide perspective is considered plotting the maximum SRF as a function of the burn-up in Figure 3. Cycles N-1 and N-2 are very similar while cycle N exhibits a SRF that, during the whole cycle, is almost double the SRF of the previous cycles. The three changes between cycle N and the previous ones listed in the previous section explain the higher SRF. The SRF of cycle N reaches its maximum value around 380 EFPD as the AO becomes positive in the last part of the cycle. This is due to the combined effect of depletion and to the accurate modelling of control rods movement, which is made possible with ARTEMIS.

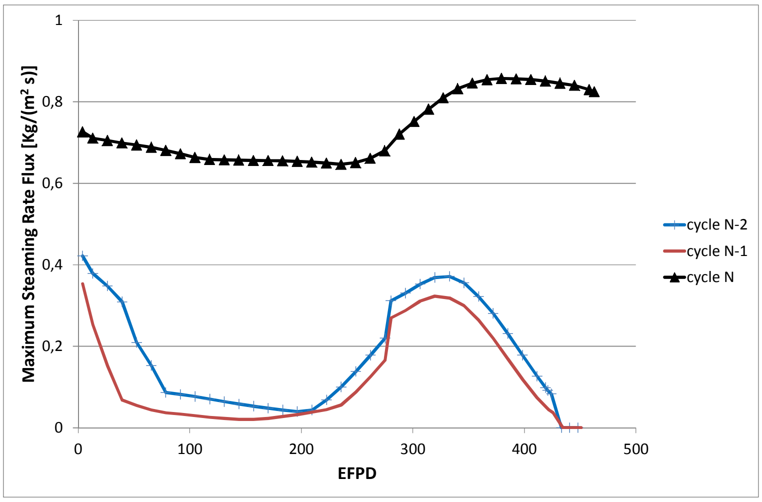

Fig. 3: Core wide maximum steaming rate flux (kg/s/m^2)

More insight into the complexity of the 3D core analysis is provided by looking at sub-channel specific data: at each selected EFPD the sub-channel having the core maximum SRF is considered for plotting both SRF and CST in Figures 4 to 7. Certain curves show dips in the SRF; these are a consequence of the axial flow redistribution following a spacer. SRF is non zero only in the upper spans, as the sub-cooled nucleate boiling regime has to be established. Comparing the cycles, the same considerations already drawn for the total core steaming rate, which is higher for cycle N than N-2, are also valid when considering the sub-channel specific values. CST is, on the other hand, much less sensitive to the change of the water temperature as the axial dependent values for cycle N and N-2 are very close. The observed dip in the clad surface temperature corresponds to the increase of the heat transfer coefficient downstream of the mixing grid, as predicted by COBRA-FLX using the mentioned HBC model.

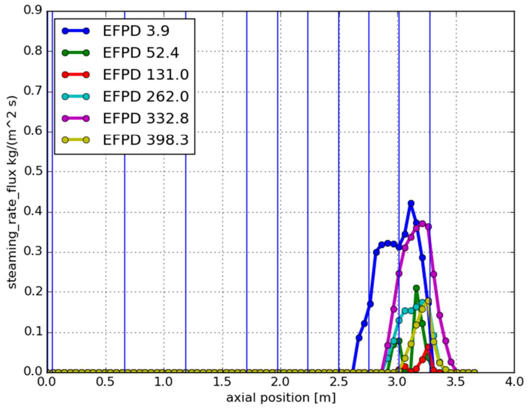

Fig. 4: Cycle N-2: axial variation of SRF (kg/s/m^2) with burn-up

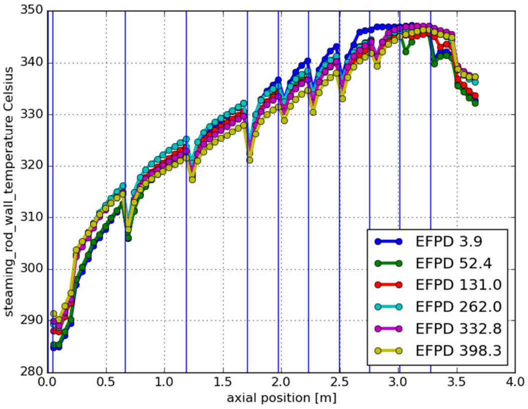

Fig. 5: Cycle N-2: axial variation of CST (deg C) with burn-up

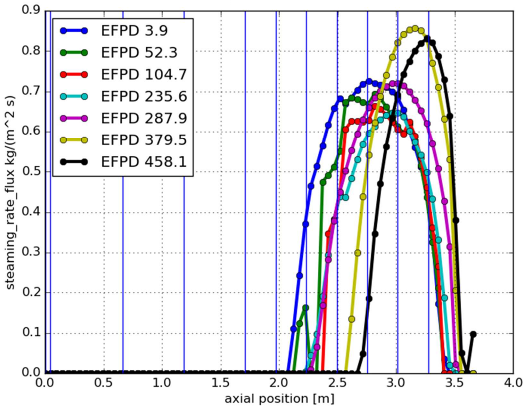

Fig. 6: Cycle N: axial variation of SRF (kg/s/m^2) with burn-up

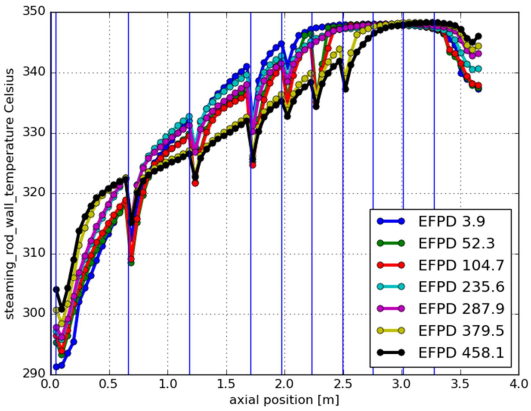

Fig. 7: Cycle N: axial variation of CST (deg C) with burn-up

All the cross-sectional TH plots shown below display only the lower right corner of the core to improve readability. SRF and CST are plotted in Figures 8 to 11 and 12 to 15 respectively. The cross-sectional views are done for cycles N and N-2 at both BOC and EOC at the axial location where the maximum SRF value is observed during cycle N, which is in the 2nd uppermost span. SRF is a sub-channel specific property, while CST is a rod specific property, and this is well represented by the Framatome visualization technique. To improve readability of CST plots, only hot rods are depicted.

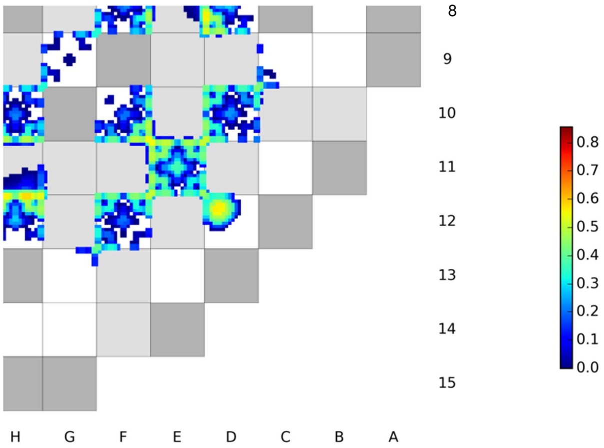

Fig. 8: Cycle N BOC: SRF (kg/s/m^2)

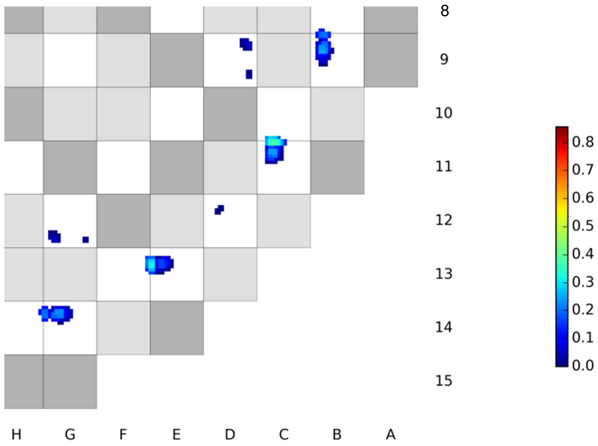

Fig. 9: Cycle N-2 BOC: SRF (kg/s/m^2)

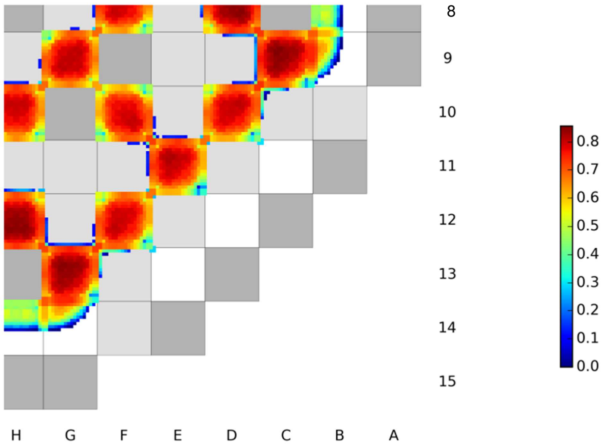

Fig. 10: Cycle N EOC: SRF (kg/s/m^2)

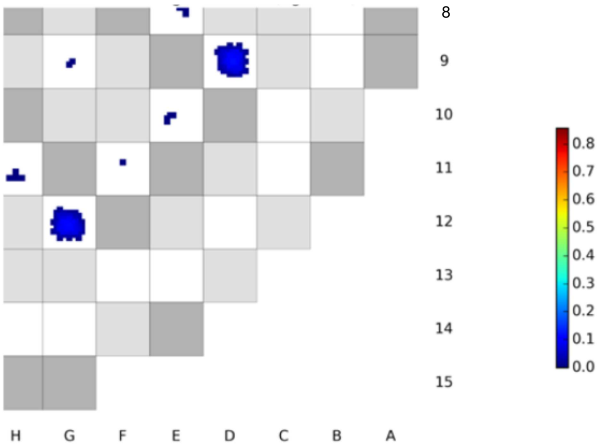

Fig. 11: Cycle N-2 EOC: SRF (kg/s/m^2)

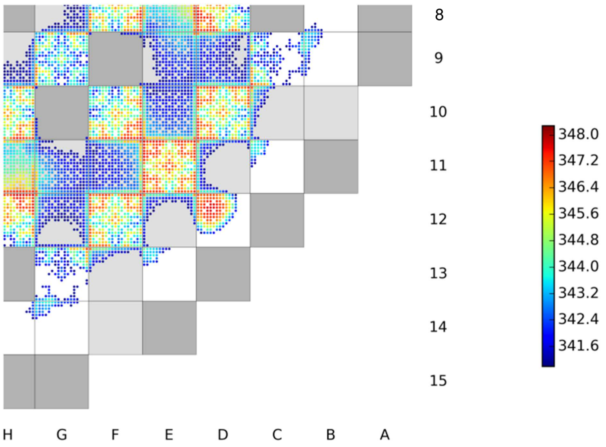

Fig. 12: Cycle N BOC: CST (deg C)

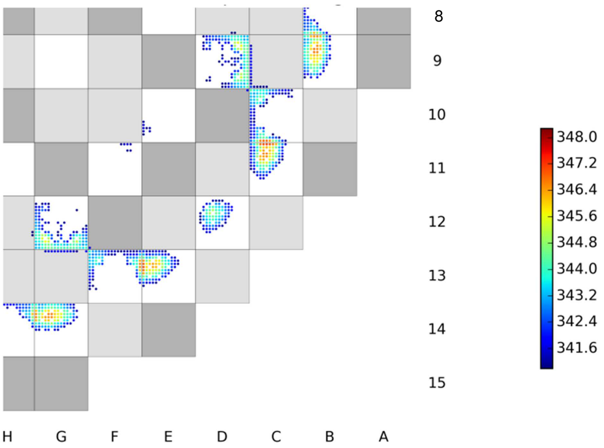

Fig. 13: Cycle N-2 BOC: CST (deg C)

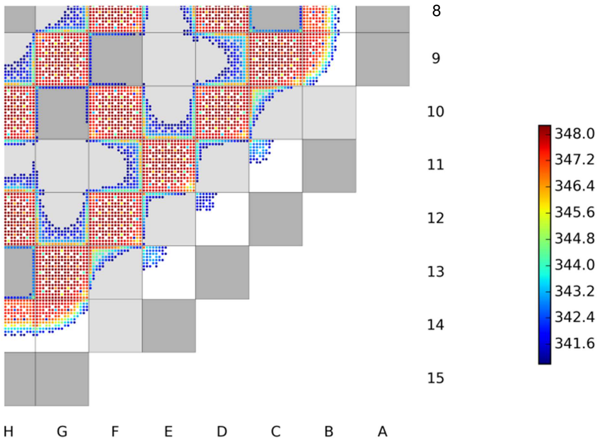

Fig. 14: Cycle N EOC: CST (deg C)

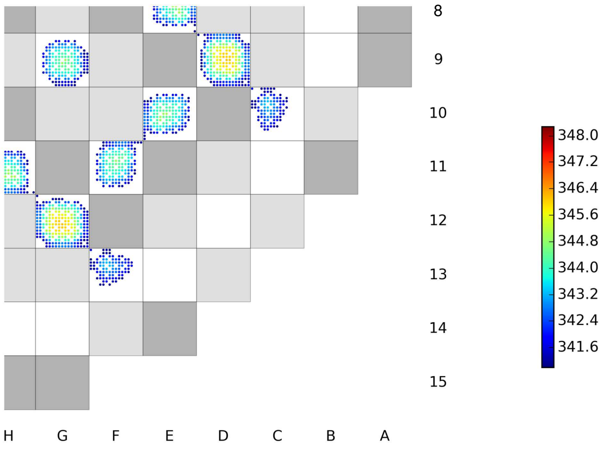

Fig. 15: Cycle N-2 EOC: CST (deg C)

The change of these local TH variables, which are key parameters in the CRA, underlines the importance of having pin-by-pin core modelling, of using realistic boundary conditions for the analysis and of considering various burn-up steps for each cycle as the individual pin data varies considerably among the core positions and with the burn-up.

The amount of TH data generated with COBRA-FLX is impressive, but only a fraction of these data is actually required to perform the final CRA. The selection criterion is based on the so-called lifetime integration.

4.4 Lifetime integration

The selection of the locations to be analysed to address the type of deposit is done using the lifetime integration. This operation shall be done for all fuel rods loaded into cycle N. Hence, multi-cycle analysis is required together with the mapping of each individual fuel rod in the loading plan to provide the history of, for example, CST and SRF.

In Figures 16 to 19, two among the top ranking fuel rod segments are shown for illustration purposes. When looking at CST, shown in Figure 16, it is possible to clearly distinguish the three cycles of operation of the two fuel rod segments (cycle N-2 and N-1 do have a stretch out operation at reduced power and hence reduced fuel temperature). Some trice burned FAs experienced steaming only during part of their first insertion cycle and some others during the third irradiation cycle, as shown in Figure 17. This is strongly dependent on the neighbouring FA which may drive a high power in the peripheral rods of trice burned FAs.

Batch N top ranking locations clearly exhibit less scattered data, Figures 18 and 19.The SRF varies considerably during the cycle, as shown previously when looking at the cross sectional plots.

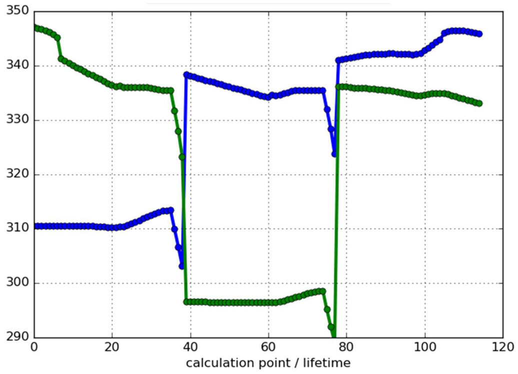

Fig. 16: Batch N-2 CST [deg C] for top ranking pin segments

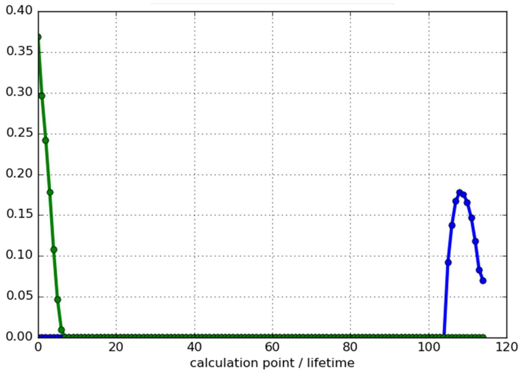

Fig. 17: Batch N-2 SRF [kg/s/m^2] for top ranking pin segments

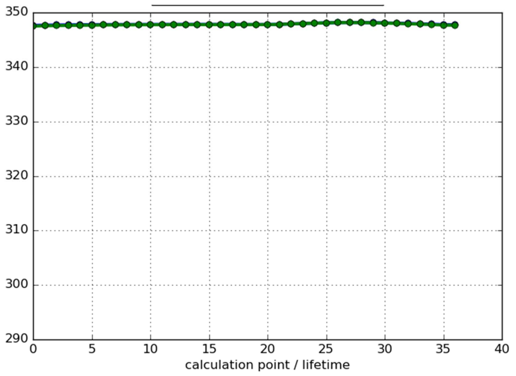

Fig. 18: Batch N CST [deg C] for top ranking pin segments (the two curves overlap in the plot)

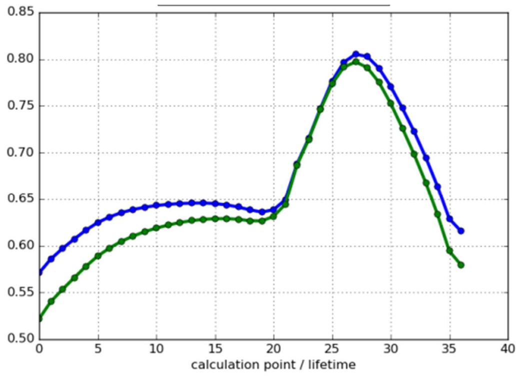

Fig. 19: Batch N SRF [kg/s/m^2] for top ranking pin segments

The cross-sectional results of SRF and CST on a lifetime averaged basis at the axial location where the maximum SRF is located are shown below for illustration purposes for cycle N core loading. These parameters, particularly the SRF criterion, provide a good representation of the core locations that will experience the most deposition and have the greatest CILC risk. The core-wide extent of crud deposition will govern the CIPS risk.

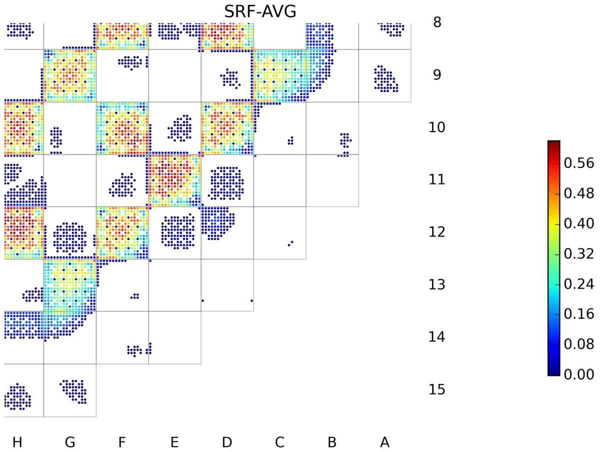

Fig. 20: cross sectional view of life-time averaged SRF [kg/s/m^2] in the core of cycle N

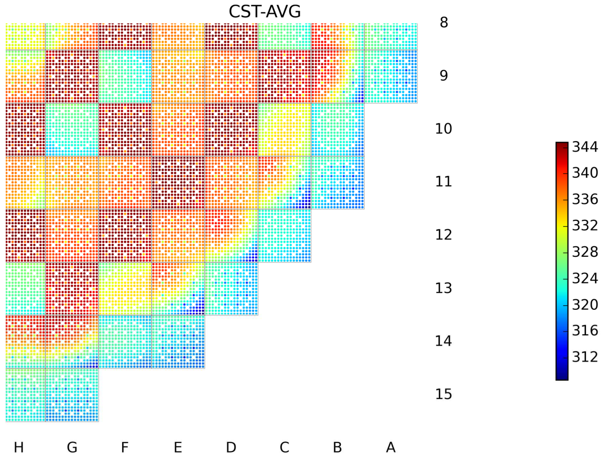

Fig. 21: cross sectional view of life-time averaged CST [deg C] in the core of cycle N

4.5 Brief extract of the final crud risk assessment

The three analysed cycles show undeniable differences in the values of steaming rate fluxes and clad surface temperature, which are explained by the changes in the cycle operating parameters. In these cases, a CRA must inevitably go through the whole assessment, including the plant chemistry analysis as the simple cycle comparison would not be sufficient to address if cycle N has a risk of crud knowing that cycle N-2 had no crud related failures or power shift.

The plant chemistry assessment requires raw/daily reactor water chemistry data which includes changes of the lithium/pH program as well as zinc injection. End of cycle manoeuvres are important as well as significant hardware changes, such as, steam generator replacement.

The final level III CRA done by the plant chemistry department for the considered cycle N concluded that:

- The CIPS evaluation indicates that a small amount of CIPS is likely to occur, but it is not expected to affect the axial power shape significantly. Depending on the accuracy of the core axial offset prediction, CIPS would be hardly measureable during plant operation or even not detected. The expected CIPS does not pose a challenge to plant operability or fuel integrity.
- The overall risk of CILC with the present core design is low and comparable with Framatome experience at other plants that have successfully operated without CILC damage. This covers:
    - Deposit thickness and under-deposit temperature
    - Deposit species
    - Chemical attack, lithium-induced damage

Considering the comparison among the TH data for the cycles, it is possible to state that the duty of the plant can be increased while maintaining an acceptably low risk of CIPS and CILC.

5 Conclusions

The selected results show the importance of the coupled multi-physics analysis, as those offered by ARCADIA, in providing best-estimate core analysis. ARTEMIS is capable of covering the complete Fuel design department scope required to support a level III Crud Risk Assessment. The importance of fine and accurate cycle specific core modelling is highlighted in this paper showing the impact that some operating parameters have on steaming rate fluxes and clad surface temperature.

ARTEMIS has a dedicated model to estimate the impact of crud boron on the axial offset for arbitrary 3D crud boron distributions. In addition, having neutronics, TH and fuel rod models available in the same code allows rapid refinement of the spatial resolution from nodal analysis, required for depletion, to pin-by-pin modelling used for evaluating local steaming rate fluxes and clad surface temperatures. Lifetime integration is the key for determining the limiting locations for a CRA and the existing AUTOCRUD tool, widely used for COBRA-FLX stand-alone analysis, has been successfully ported to the ARCADIA environment handling the ARTEMIS output.

The usage of ARCADIA does not change the way Framatome plant chemistry department performs the final level III CRA. Framatome is hence now able to offer a level III CRA based on the code system ARCADIA to all utilities in the worldwide market while profiting from the valuable experience feedback gained in US for different NSSS vendors. Thanks to these assessment capabilities, Framatome can provide support and guidelines on crud risk assessment and crud mitigations strategies to the customers needing or willing to change their plant operation strategy and to replace ageing components of the primary circuit, with a clear economic benefit for them.

6 Acronyms

AO	Axial Offset
BE	Best Estimate
BOC	Begin Of Cycle
CILC	Crud-Induced Localized Corrosion
CIPS	Crud-Induced Power Shift
CFD	Computational Fluid Dynamics
CRUD	Corrosion Related Unidentified Deposit
CST	Clad Surface Temperature
EFPD	Effective Full Power Days
EOC	End Of Cycle
FA	Fuel Assembly
FDIC	Fuel Deposit Interactive Chemistry (code)
FRM	Fuel Rod Module
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
SRF	Steaming Rate Flux
TH	Thermal-Hydraulics
THM	TH Module

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