ARTEMIS / RELAP5 INTEGRATED TRANSIENT ANALYSIS APPLICATION TO NON-LOCA TRANSIENTS

WILLIAM WALTERS, RUXANDRA BOBOLEA, KEITH MAUPIN, RICHARD DEVENEY, JOSHUA PARKER, KEVIN SEGARD, and ROBERT BARNER

US Fuels, Framatome Inc.

3315 Old Forest Rd, Lynchburg, VA 24501, USA

ABSTRACT

Framatome is implementing a realistic non-LOCA safety analysis methodology for application to PWRs. Realistic assessment of the behaviour of the reactor system under accident conditions requires the core power, core thermal-hydraulics, fuel thermal-mechanical, and plant thermal-hydraulics calculations to be executed in a coupled manner. Framatome's ARITA methods package brings this capability to the nuclear industry by coupling the advanced 3D transient core simulator ARTEMIS to the S-RELAP5 plant thermal-hydraulics simulator. Within the ARITA methods package, proximity to fuel safety limits is determined by COBRA-FLX for DNB and GALILEO for fuel centreline melt and transient clad strain. To complement the elimination of unrealistic conservatism and resolve the event outcome space at the desired level of coverage and confidence ARITA also incorporates a rigorous uncertainty evaluation method based on Monte Carlo simulation, uncertainty propagation, and non-parametric statistics. This paper summarizes results from the application of ARITA to select non-LOCA events.

1.0 Introduction

As a supplier of fuel to the nuclear power generation industry Framatome is continuously investing in new technologies designed to support its customers operation initiatives, to proactively respond to and provide flexibility for changes in regulatory environments, and to further its commitment to safe operation of nuclear power. Framatome has developed and is now implementing the new codes ARCADIA [1] and GALILEO [2] thus adding a state of the art neutronic/thermal-hydraulic/thermal-mechanical PWR core simulator and a state of the art fuel rod performance code to its product package. Framatome's ARTEMIS/RELAP5 Integrated Transient Analysis (ARITA) methods package is the calculational framework (evaluation model) that has been developed to apply these new codes in the evaluation of the reactor system behavior during a postulated transient or design basis accident.

Comprised principally of the core simulator within the ARCADIA system named ARTEMIS [3] and the plant thermal-hydraulic simulator S-RELAP5, the ARITA model couples these tools to obtain core power, core thermal-hydraulic, and core thermal-mechanical responses (ARTEMIS) along with the plant thermal-hydraulic response (S-RELAP5) during transient simulation of a wide variety of non-LOCA transients. Data generated in the transient event simulation are then further processed and passed along for Departure from Nucleate Boiling (DNB) evaluation using COBRA-FLX [4] and Fuel Centreline Melt (FCM) and Transient Clad Strain (TCS) in GALILEO.

As the calculational framework for the use of Framatome's advanced computer codes ARITA covers a wide range of topics that must be addressed to support its use in reactor safety licensing calculations including:

- Methodology description including specification of calculation procedure
- Accident scenario and important parameter identification process
- Code applicability assessment
- Code uncertainty analysis
- Plant / event specific model inputs determination process

The items listed above define a scope of work that provides all of the information needed to show that the method:

- Has chosen computer codes that are suitable for application to the class of transients covered by the methodology and that are shown to predict important physical phenomena reasonably well through extensive benchmarking
- Uses all computer codes and other models within the associated ranges of applicability
- Includes rigorous processes to:
  - Identify important physical phenomena and reactor components that have an impact on the determined margin to specified figures of merit
  - Assess code uncertainty / adequacy with sound bases for application of necessary biases and uncertainties.
  - Assess the plant operating space and operating space uncertainties

This work lays the foundation upon which the methodology for the analysis of non-LOCA transients for reactor safety calculations is constructed. Final specification of inputs requires that all approximations, simplifications, uncertainties, and probability distribution treatments have been addressed in a reasonable fashion. Once established for a given plant, the ARITA model serves as a calculational tool that is used to perform a Monte Carlo simulation and uncertainty propagation of the outcome space for a given event. Each transient is executed a pre-defined number of times in order to ensure that the acceptance criteria are met with at least a 95% probability and a 95% confidence using the well-known Wilks' formula [5] for non-parametric order statistics.

2.0 Application Examples

Several examples follow to highlight the benefits of the ARITA method. In each of these examples, the transient is evaluated 59 times, which is the minimum number of cases required for the 95/95 order statistic based on the Wilk's formula.

2.1 Locked Rotor

The ARITA methodology is applied to the Reactor Coolant Pump (RCP) Rotor Seizure event for a Westinghouse designed 3-loop plant. This postulated event is caused by the instantaneous seizure of a reactor coolant pump rotor while operating at Hot Full Power (HFP) conditions. The flow through the faulted Reactor Coolant System (RCS) loop decreases rapidly. The Reactor Protection System (RPS) senses a Low RCS Loop Flow signal and actuates a reactor trip causing the insertion of the control rods and leading to a turbine trip. A Loss of Offsite Power (LOOP) is also assumed, which causes the remaining RCPs to trip and coast down, further reducing the core flow. The relatively high fuel rod heat flux combined with the decreasing core flow results in core coolant temperature escalation which challenges the DNB safety limit. The reactor coolant expansion causes a pressurizer in-surge and increases the RCS pressure. This may actuate the automatic pressurizer spray system and may even open the pressurizer Power Operated Relief Valves (PORVs) in an effort to control RCS pressure within normal operating bounds. The reduction in pressure by this non-safety grade system can cause a closer approach to the DNB Safety Limit.

A number of figures are presented to illustrate the advantages conferred by the application of ARITA methods to RCP Rotor Seizure event. Figure 2-1 visually depicts the change in axial power distribution (relative axial power times fraction of power) during the event as calculated by ARITA. The coupling of neutronics and thermal-hydraulics provides ARITA with the ability to capture the effect of continuously changing axial power distribution during the event, allowing for a more realistic simulation of the event progression. The effect of negative reactivity insertion from the loss of flow is immediately seen in the first 1 to 2 seconds. Thereafter, the negative reactivity from control rod insertion reduces the power starting from the top of the core and moving to the bottom of the core from 1.4 to ~5 seconds. This initial power reduction in the top of the core translates into positive Departure from Nucleate Boiling Ratio (DNBR) margin and earlier DNBR recovery for an event traditionally resulting in DNB fuel rod failure. In traditional methods, conservative reactor trip characteristics are needed because a point kinetics calculation by definition has no axial power distribution to change during the event.

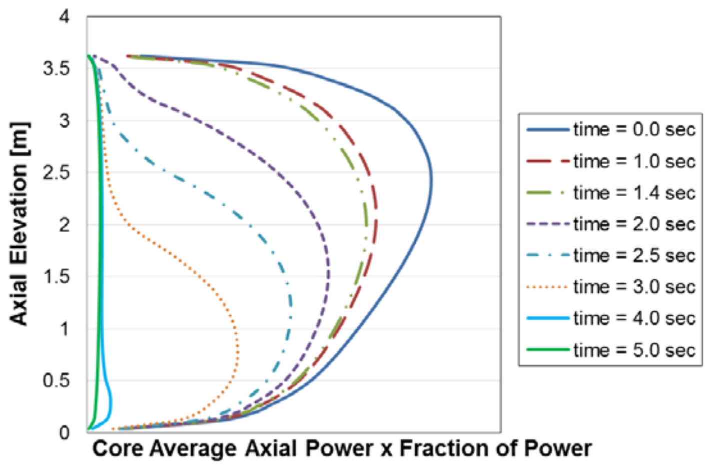

Figure 2-1: Representation of Axial Power Distribution during RCP Rotor Seizure Event

Figure 2-2 presents a comparison of power to flow ratio for ARITA method versus the traditional approach. The application of non-parametric statistics as well as the advantages introduced by code coupling are translated into less severe conditions and therefore improved DNB margin outcome for this event. This is also illustrated in Figure 2-3, which gives an example of ARITA minimum DNBR margin for 59 cases along with the minimum DNBR margin from traditional methods, which predict DNB fuel failure.

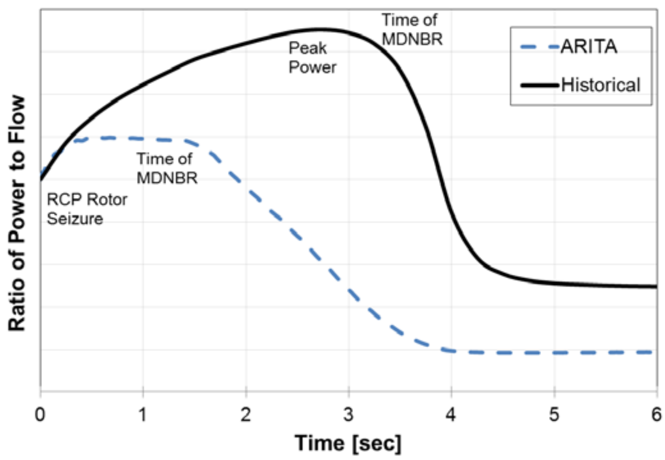

Figure 2-2: Illustration of Power to Flow Ratio During RCP Rotor Seizure event – ARITA versus Historical Approach

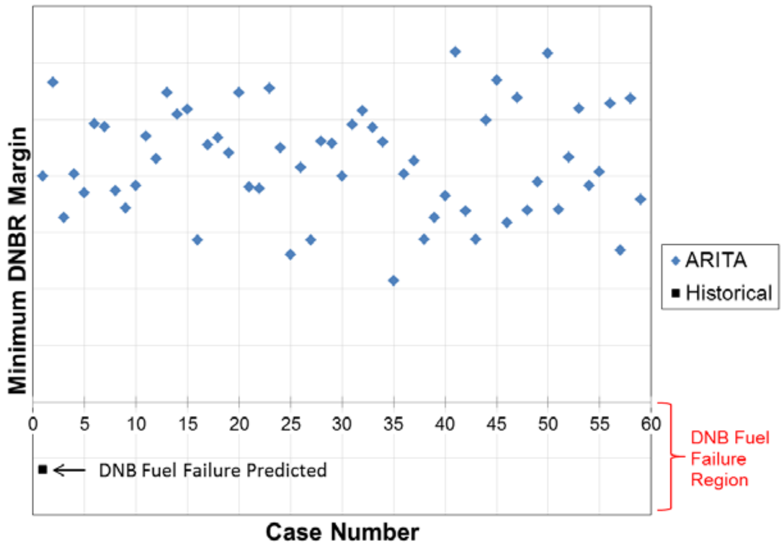

Figure 2-3: Illustration of RCP Rotor Seizure Minimum DNBR margin – ARITA versus Historical Approach

Figure 2-4 illustrates coolant mass flow distribution for the ARITA RCP Rotor Seizure, at time of minimum DNBR and different axial elevations. As can be seen, because of cross flow in the core the effect of non-uniform flow distribution due to the RCP Rotor Seizure dissipates as flow moves through the core, and therefore the flow maldistribution impacts are mostly limited to the bottom half of the core.

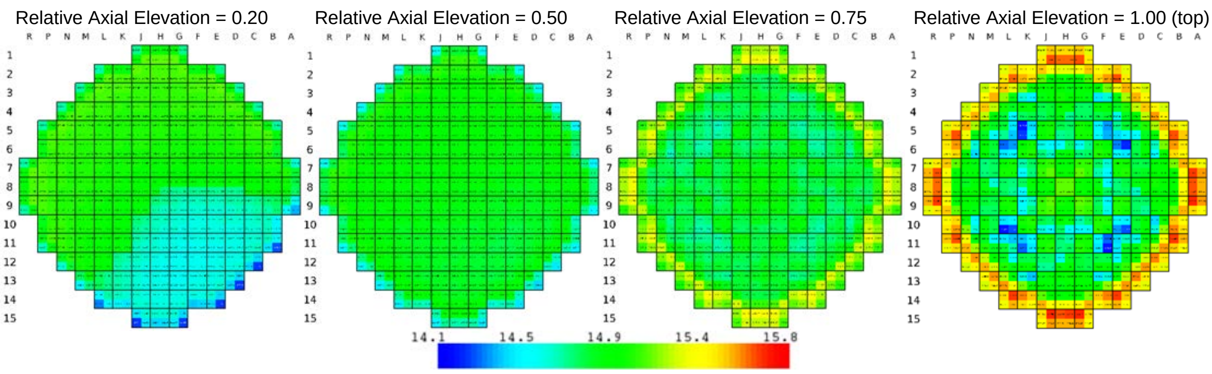

Figure 2-4: RCP Rotor Seizure Coolant Mass Flow Distribution at Different Axial Elevations and Time of MDNBR

2.2 Uncontrolled Control Rod Bank Withdrawal at Power

The ARITA methods package was applied to the Uncontrolled Control rod Bank Withdrawal (UCBW) at power event for a Westinghouse designed 4-loop plant. The uncontrolled withdrawal of a bank of control rod assemblies while operating at power leads to an increase in the core power and the core heat flux. The steam generator heat removal initially lags behind the core power generation, so the increase in core power leads to increases in the core moderator temperature, the pressurizer pressure, and the pressurizer level. Control systems may take action in response to the event, but the power and temperature continue to increase, and have the potential to result in DNB and/or FCM. In order to protect the core, the Reactor Protection System (RPS) includes several trip functions that may be reached during an UCBW, including the high flux trip, the high positive flux rate trip, the Overtemperature ΔT (OTΔT) trip, the Overpower ΔT (OPΔT) trip, the high pressurizer pressure trip, and the high pressurizer level trip. Safety analyses demonstrate that if an UCBW were to occur while operating within the allowed operating space plus uncertainty, then the fuel cycle design and the RPS trip setpoints are appropriate to preclude DNB and FCM with at least a 95% probability and a 95% confidence.

The space of possible outcomes for the UCBW transient is determined by multiple key input parameters that may change during the event. The power increase during an UCBW is based on the reactivity insertion rate, which in turn is determined by the initial position of the control rods, the rate at which the control rods are removed from the core, the evolution of the axial power distribution during the event, and the moderator and Doppler reactivity feedback. Each of these parameters may also affect other aspects of the UCBW transient. The initial control rod position and the rate of rod withdrawal determine the duration of the reactivity insertion. The control rod position and the axial power distribution, both of which change throughout the event, influence the power peaking in the core. Similarly, the axial power distribution affects the rate of positive reactivity insertion as the control rods are withdrawn, the core power peaking, the rate of negative reactivity insertion following a reactor trip, the DNB calculation, and the evaluation of the OTΔT and OPΔT reactor trip setpoints. This is not an exhaustive review, but is sufficient for demonstrating the complex interaction of multiple parameters that define the total outcome space for an UCBW event.

In traditional methods, which typically include independent models for the system thermal hydraulics, the core thermal hydraulics, the core neutronics, and the fuel rod behavior, the entire event outcome space is not evaluated. Instead, the outcome space is bounded by considering limiting values for many of the key parameters and by performing sensitivity studies. A conservative bias can be applied to parameters where the limiting condition is well understood, such as using a minimum value for the reactor coolant system flow to produce a more conservative evaluation of DNB. In some cases, the same parameter affects multiple models and the conservative direction is not the same for each model. This is demonstrated by examining the treatment of the axial power distribution, which continuously changes during an UCBW as the control rods are withdrawn. Traditional methods calculate the power progression in the plant thermal-hydraulics model using point kinetics, which does not model the axial power distribution during the transient. For conservatism, the reactivity insertion rate and the reactivity feedback parameters in the point kinetics model are independently varied through sensitivity studies to find the limiting core power and thermal conditions without considering if the combinations evaluated can realistically occur. The one parameter in the point kinetics model that is tied to a particular axial shape is the negative reactivity insertion following reactor trip, which is typically based on a bottom-peaked axial power distribution in a static model to delay the benefits of control rod insertion. Meanwhile, in the core thermal-hydraulics model, multiple axial shapes are evaluated independently to demonstrate that the limiting conditions from the plant thermal-hydraulics model do not result in DNB. Consequently, it is often the case that the limiting axial power shape for calculating DNB in the thermal-hydraulics model is a top-peaked axial shape even though the point kinetics model uses a reactivity model based on a bottom-peaked axial shape for the reactor trip characteristics.

Through coupling, ARITA removes modeling simplifications and conservatisms by maintaining a consistent value for the key parameters in all aspects of the transient simulation. Returning to the example of the axial power distribution, ARITA calculates the 3D power distribution as it evolves over time, as demonstrated in Figures 2-5 and Figure 2-6. Figure 2-5 presents the change in the axial power offset during one of the UCBW cases evaluated using ARITA. Figure 2-6 presents the 3D power distribution in one quadrant of the core for the same case calculated using ARITA at a) the initial conditions for the event and at b) the time when minimum DNB is calculated. In this case, the power is initially concentrated in the bottom of the core, away from the inserted control rod assemblies. As the control rods are removed from the core, the axial offset approaches zero and then shifts toward the top of the core. At the minimum DNB condition, the control rods are fully withdrawn and the power is well concentrated in the top half of the core. By modeling the change in the axial power distribution over the event, ARITA automatically includes the effects of the changing power distribution on the reactivity insertion rate, the reactivity feedback parameters, the core power peaking, the DNB calculation, and the evaluation of the OTΔT and OPΔT trip setpoints. The ARITA method then applies Monte Carlo sampling of the event outcome space and non-parametric statistics to ensure that the acceptance criteria are met with at least a 95% probability and a 95% confidence. Figure 2-7 demonstrates the improvements in the calculated DNB margin for an UCBW event by comparing the results for the cases calculated by ARITA to the margins calculated when using traditional methods.

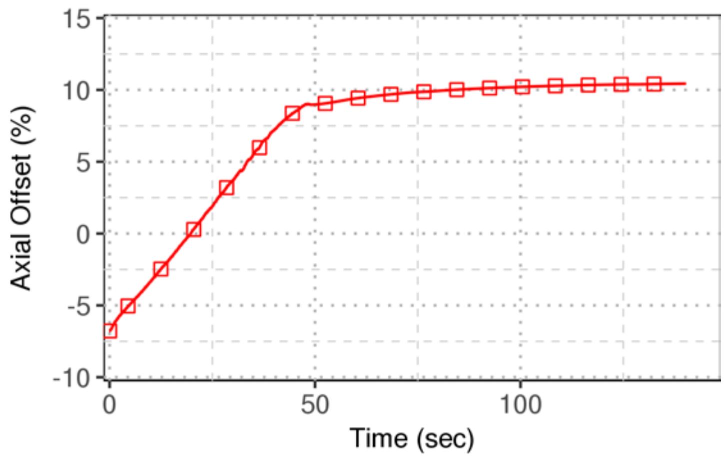

Figure 2-5: Axial Power Offset [(Power in Top Core Half – Power in Bottom Core Half) / Total Power] during an Uncontrolled Control Rod Bank Withdrawal Event

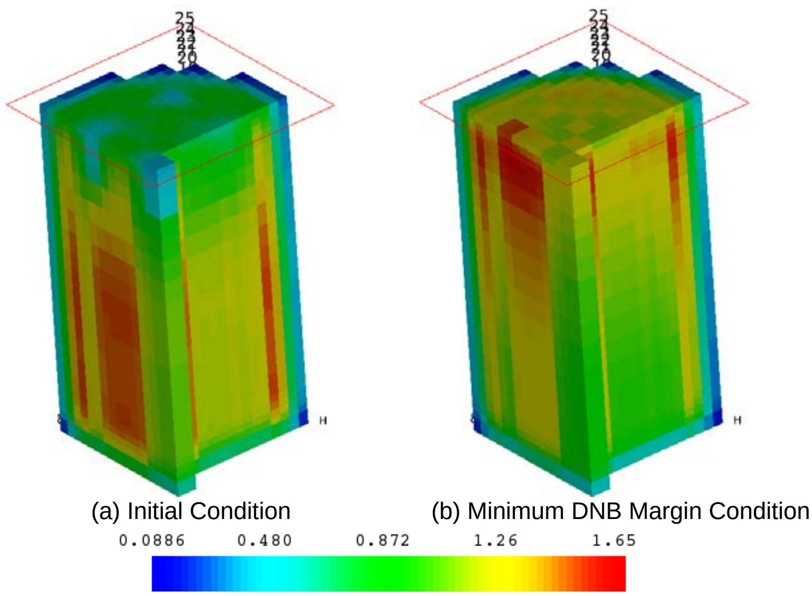

Figure 2-6: Relative Nodal Power Density during an Uncontrolled Control Rod Bank Withdrawal Event

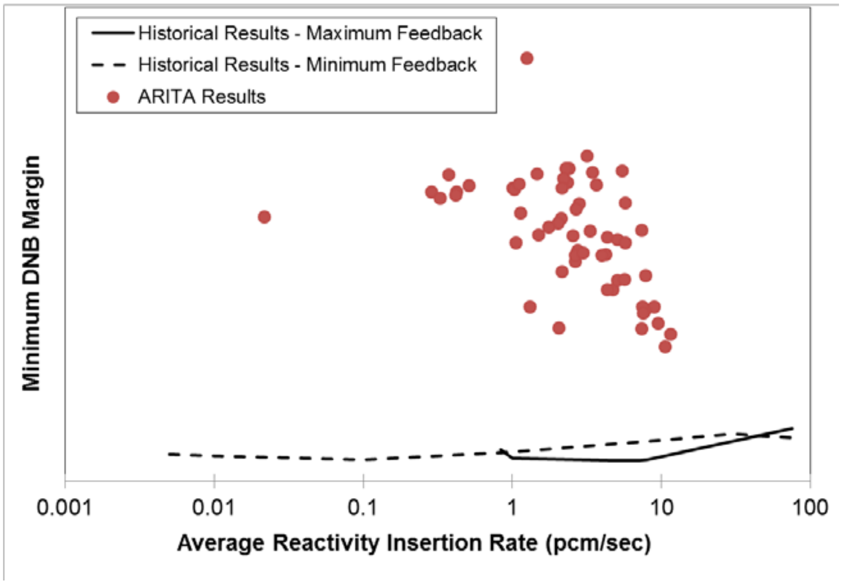

Figure 2-7: Comparison of ARITA Minimum DNB Margin to Traditional Methods for the Uncontrolled Control Rod Bank Withdrawal Event

2.4 Post Scram Main Steam Line Break

The ARITA methods package was also applied to the post scram Main Steam Line Break (MSLB) event for a Westinghouse designed 4-loop plant. The initial conditions include Hot Zero Power (HZP) temperatures with the RCPs running. Rupture of a steam line will cause the affected steam generator (SG) pressure and temperature to rapidly decrease. Protective systems will isolate the main steam lines by closing the Main Steam Isolation Valves (MSIVs), but the affected loop will continue to depressurize if the break is upstream of the valve. The SG depressurization causes a rapid cool down in the Reactor Coolant System (RCS) loop associated with the affected SG and in the sector of the core cooled primarily by water from the cold leg of the affected loop. The RCS cooldown inserts positive reactivity from both fuel temperature and moderator temperature feedback. As the RCS cooldown continues the integrated positive reactivity insertion erodes the negative reactivity margin that is required during a normal shutdown. The analysis includes consideration for a stuck control rod in the fully withdrawn position and leads to a condition where the neutron flux will be highly peaked in the region of the stuck rod. If the reactivity insertion exceeds the minimum shutdown margin, criticality may be re-established and power escalation may follow. When the core sector with the stuck control rod is also the sector that is cooled primarily by the affected loop cold leg, the magnitude of the positive reactivity inserted is increased and power production in the core and particularly in the location of the stuck control rod must be evaluated. The power excursion is eventually terminated by either a loss of inventory in the affected SG or by the addition of boron from safety injection. Safety analyses demonstrate that if the MSLB event were to occur with suitably conservative consideration for variation of conditions that may be present due to the allowances for operation in the plant technical specifications and uncertainties associated with important parameters, that the radiological release limits established for the plant are not exceeded due to fuel rod failure from DNB or FCM.

ARITA is well suited for application to the complex post scram MSLB event because the core and system response are resolved simultaneously allowing for accurate consideration of the reactivity insertion of the cold water relative to the available shutdown margin and the complex feedback effects in the stuck rod area as power is produced.

The initial shutdown margin and its subsequent erosion are both functions of a sizeable list of parameters including the reactivity worth of the control rods minus the stuck rod, initial boron concentration and its impact on moderator feedback, the break size, and many others. The

ARITA method addresses each of these parameters from the code uncertainty / adequacy perspective as well as the allowed operating space perspective. Careful consideration is applied to the assessment of important parameters and an informed combination of conservative input biases and random input sampling of certain parameters is specified. The result, with respect to initial shutdown margin and time of re-criticality, is a spectrum of responses (Figure 2-8) each of which represents a potential path for the evolution of the event in the total event outcome space at the plant.

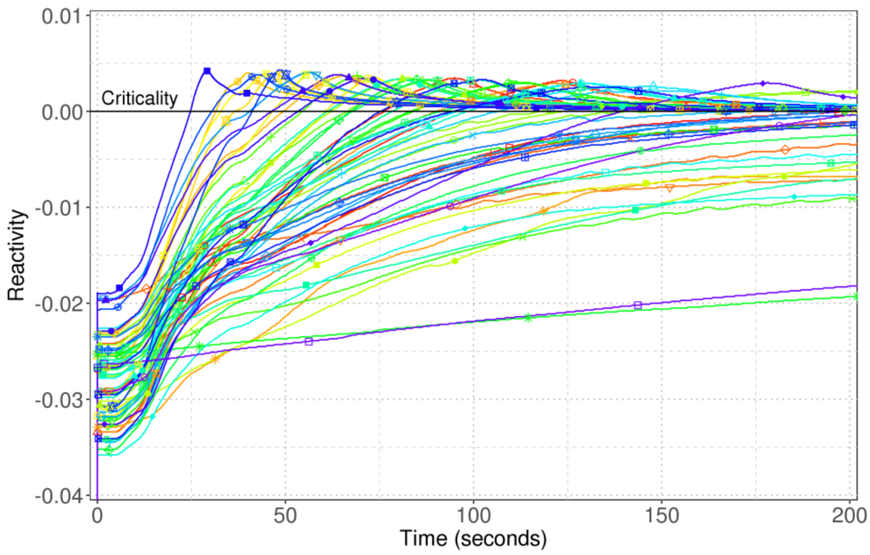

Figure 2-8: Core Reactivity Estimate as a Function of Time for a Set of 59 HZP MSLB Simulations

Because the MSLB event is strongly asymmetric with respect to failed steam generator resulting in cold inlet temperatures to the core, the S-RELAP5 input models all of the RCS loops explicitly. In this manner the asymmetric core inlet temperature distribution is directly coupled from the system simulation to the core simulation. As with erosion of shutdown margin discussed above, the RCS loop temperature response is also a function of multiple input parameters and here again the various combinations of sampled conditions over the number of cases analyzed lead to a spectrum of RCS loop temperature transients. A mixing method is applied in the simulation that facilitates the consideration of the exchange of liquid at dissimilar temperatures between loops when flow from these loops meet in the reactor vessel. Together these features in the ARITA method deliver the asymmetric core inlet temperature distribution to the core simulator where the core power and 3D power distribution responds to this abnormal event.

With the core inlet conditions delivered from S-RELAP5 to the core simulator (ARTEMIS) the core power and 3D power distribution are passed to COBRA-FLX and GALILEO to evaluate the DNB and fuel centerline temperatures, respectively. The method of modeling the core in the ARITA methodology ensures that the entire core and the very large number of combinations of local conditions (local coolant temperature, local power peaking, etc.) are evaluated for each of the cases within a given set of N cases. Figure 2-9 provides an illustration of the spatial resolution of power distribution in the core by showing relative pin power density at several times during a given MSLB transient simulation.

The MSLB event in the ARITA methodology includes the execution of a case set (e.g. min 59 cases) where uncertainty treatments and initial conditions are sampled as prescribed by the rigorous validation of important physical phenomena and reactor components. The set of results is assessed as a collection of samples from the event outcome space at the plant. Figure 2-10 shows a collection of the more limiting MDNBR results from a 59 case evaluation. Application of ARITA to the post scram MSLB event demonstrates significant margin to DNB failure. Using order statistics the case representing the desired tolerance limit is identified and if fuel rod failure is predicted then the number of failed fuel rods is determined and the subsequent radiological release calculation is performed.

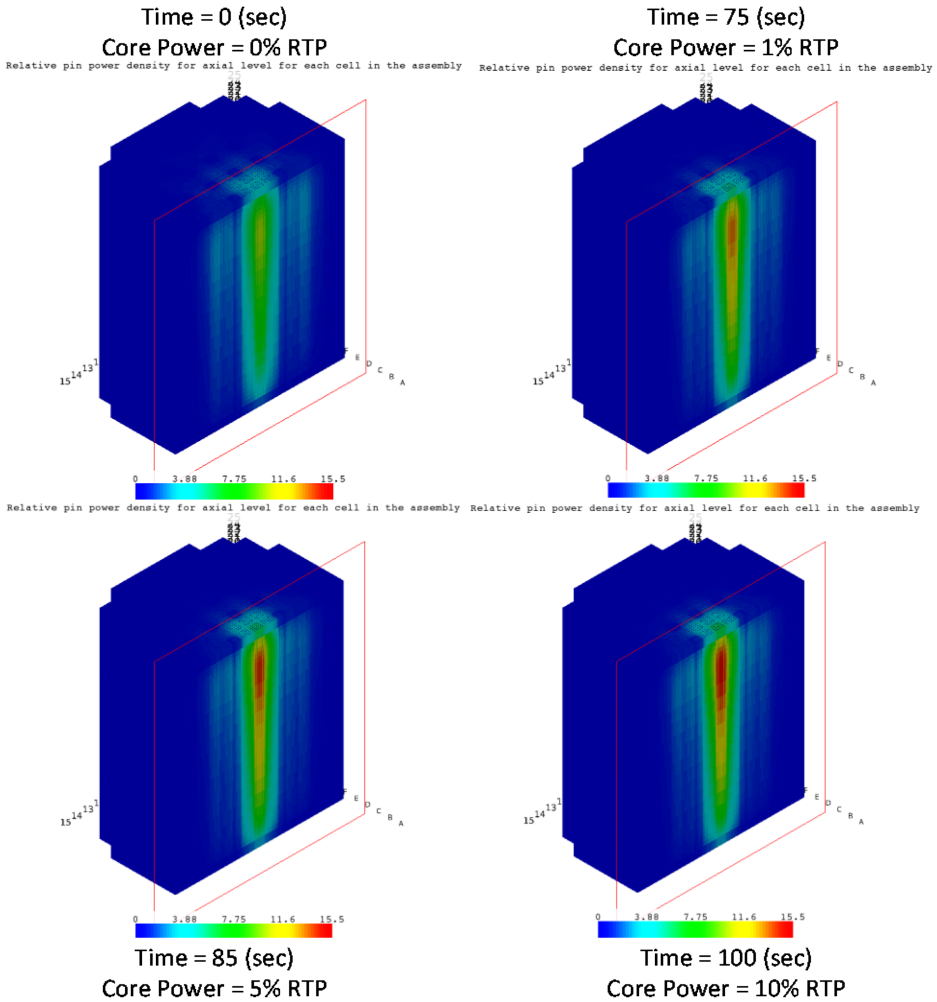

Figure 2-9: Relative Pin Power Density at Four Times in the HZP MSLB Transient

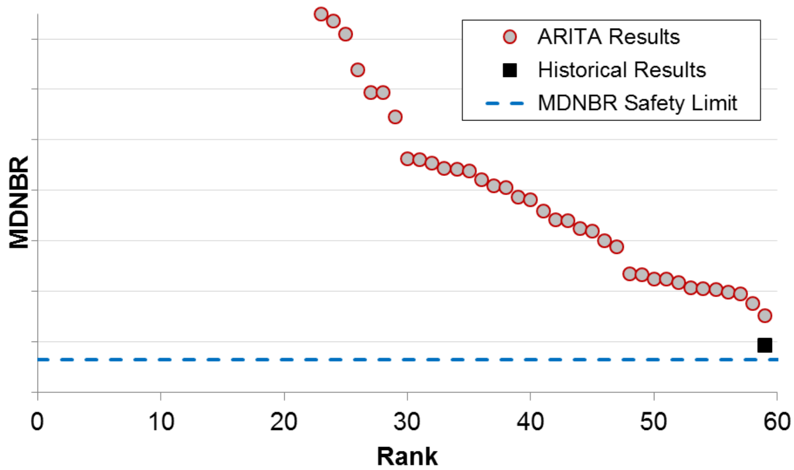

Figure 2-10: MDNBR by Rank for the More Limiting Results from a 59 Case Post Scram MSLB Evaluation

3.0 Conclusion

The ARITA methods package is the culmination of over a decade of new computer code and methods investment and development and brings a realistic coupled multi-physics simulation capability with a backbone of state of the art tools to the demonstration of nuclear fuel performance and safety in reactor licensing basis calculations. When ARITA is used for plant safety analysis, the resulting margins for the transient events shown are significantly different than those obtained from traditional deterministic methods where the core response is based on a simple point kinetics model with conservative core models. Implementation benefits of Framatome's advanced codes and methods were addressed in detail in [6], and [7]. This paper adds to the advanced code and method value demonstration portfolio by showing increased margin for this challenging subset of non-LOCA transient events. Application of ARITA in a plant's licensing basis will deliver more accurate assessment of margin to safety limits and enhanced operational and design margin available for customers to use in whatever manner is valuable to them.

4.0 References

[1] F. CURCA-TIVIG 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, USA, September 30- October 3, 2007.

[2] 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).

[3] 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).

[4] M. LEBERIG 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.

[5] Wilks, S. S., "Determination of Sample Sizes for Setting Tolerance Limits", The Annals of Mathematical Statistics, 12(1), 91-96, 1941

[6] K. Maupin, D. Porsch, S. Kuch, S. Opel, F. Le Bars, G. Simonini, R. Brock, D. Deveney, AREVA's ARCADIA Code System – Implementation Benefits, Top Fuel 2015, Zurich, Switzerland, September 13-17 , 2015.

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