COMPARISON OF BEST ESTIMATE PLUS UNCERTAINTIES AND CONSERVATIVE METHODOLOGIES FOR A PWR MSLB ANALYSIS USING COUPLED 3D NEUTRON KINETICS/THERMAL-HYDRAULIC CODE

R. Pericas[1], K. Ivanov[2], F. Reventós[1], L. Batet[1]
[1]Universitat Politècnica de Catalunya , Av Diagonal, 647 08028 Barcelona
SPAIN
Phone +34 93 401 7143
[2]Pennsylvania State University, 206 Reber Building, University Park,
Pennsylvania 16802 USA phone (814)865-0040

raipercas@yahoo.es, francesc.reventos@upc.edu, kni1@ncsu.edu,
lluis.batet@upc.edu

ABSTRACT

This paper provides a comparison between the Best Estimate plus Uncertainty methodology with the Conservative Bounding methodology for Design Basis Accident analysis. Calculations have been performed with TRACE (for thermal-hydraulic system calculations) and PARCS (for neutron-kinetics modeling) under SNAP platform. DAKOTA is used under the SNAP interface for uncertainty and sensitivity analysis. A simplified 3D neutronics model of the ASCO II NPP is used as core kinetic model. The thermal-hydraulic model is a 1D representation of the primary and secondary systems except for the vessel that is represented by a 3D VESSEL component. The design basis transient selected for the comparison is a Main Steam Line Break (MSLB) in a PWR. This transient is characterized by space-time effects and requires coupled 3D kinetics and thermal-hydraulics modeling, especially for the re-criticality scenario. The comparison methodology is as follows. Once the models are created, a Best Estimate Base Case Calculation (BEBCC) is performed. The model is modified by selecting the most important parameters and assigning conservative values to them in order to obtain a Conservative Calculation. Several parameters are modified in this conservative way. These parameters are then perturbed in Best Estimate Plus Uncertainties (BEPU) calculations. At the end, a comparison is made between results obtained in the Conservative Calculation and the BEPU methodology respectively. As a general conclusion the BEPU method has been successfully illustrated in coupled 3D kinetics and thermal-hydraulics system. Also the study it is an effective test for the adequacy of nodalizations for the utilized codes both neutronic and thermal-hydraulic. BEPU methodology gives more margins, which allow for higher operational flexibility of the plant. Results of BEPU methodology help improve plant economics while meeting the safety standards. As a conclusion, the BEPU methodology is been successfully tested in NK-TH calculations. Narrow margins between upper and lower BEPU case are consequence of few perturbed parameters chosen and the transient boundary conditions.

1. INTRODUCTION

Conservative nuclear safety margins limit the present industrial need of increasing Nuclear Power Plant (NPP) electricity output. Best Estimate Plus Uncertainties (BEPU) techniques are superior to the old conservative methodology, where the safety margins are established by experts under operation hypotheses and conservative assumptions. The BEPU methodology is capable of providing a solution in terms of increasing the nuclear power production without compromising the safety margins. This paper exemplifies a comparison between the BEPU methodology and the Conservative Bounding methodology, as applied to a particular accidental scenario.

The transient selected for the comparison is a Main Steam Line Break (MSLB) in a Pressurized Water Reactor (PWR). The failure of a main steam line results in an initial increase in the steam flow, which decreases afterwards driven by the secondary pressure reduction. The break in the secondary causes a reduction in primary system coolant (moderator) temperature and pressure. In the presence of a large negative moderator temperature coefficient, the excess cooling results in a reduction of the core shutdown margin. If the most reactive control rod bundle remains on its fully out position, after the reactor SCRAM, it is possible that the reactor becomes critical and relatively high power is achieved locally in the vicinity of the place where the most reactive control rod should have been inserted. The core critically is finally stopped by the injection of boric acid discharged from the safety injection system.

To perform the comparison between the two methodologies, first a Best Estimate Base Case Calculation (BEBCC) is performed followed by BEPU calculations with a selection of perturbed parameters. Such selection is made by following the recommendations of Priority Identification and Ranking Tables (PIRTs), OECD/NRC PWR MSLB benchmark project report [1] and CRISSUE [2] project guidelines. A Conservative Calculation is done by assigning conservative values to these parameters. At the end, a comparison is made between the predictions of Conservative and BEPU methodologies.

The calculations have been made using the following tools: TRACE [3] for thermal-hydraulic system calculations, PARCS [4] for reactor physics modeling and DAKOTA [5] for uncertainty and sensitivity analysis.

2. MODELS

Nuclear system coupled calculations involve at least two codes: a Neutron Kinetics (NK) code to simulate the core behavior, and a thermal-hydraulic (TH) code to model the coolant system. In this section, the NK and TH models developed with PARCS and TRACE for this study are described.

2.1. TRACE model

Ascó NPP is a 3 loops 2900 MW PWR. TRACE model reproduces the whole Nuclear Steam Supply System. TRACE V5 patch2 is the version of the code used in the present study. The model developed has been validated with an actual 50% loss of load transient [6] in this process it has been benchmarked against other TH models [7, 8, 9, 10, 11] widely used in Universitat

Politècnica de Barcelona group. In a coupled NK-TH code calculation, the most relevant part of the thermal-hydraulic model is the reactor vessel. A 3D VESSEL TRACE component represents the Reactor Pressure Vessel (RPV) in the model. The vessel model has 15 axial layers, 6 azimuthal sectors and 5 radial rings. Three axial levels represent the lower plenum; six axial levels represent the active core region, the down comer and the bypass; the top levels model the upper head and the upper plenum. Figure 1 shows an axial layout of the vessel representation. The axial core region (lighter area) is subdivided radially for each layer in 18 TH cells formed by overlapping three rings and six sectors. Outer rings represent the down comer and the bypass. The height of each active core axial node is 0.609 m. The total active core axial height is 3.654 m. The rest of the plant model is a 1D representation of each of the three loops, with the pressurizer `connected to one of the loops. Three main steam lines are modeled in the secondary circuit representation. Main Feed Water and Auxiliary Feed Water systems are also modeled for each loop. In terms of safety injection systems, the three accumulators are modeled. The Low Pressure Injection System (LPIS) and High Pressure Injection System (HPIS) injection nozzles are simulated by means of FILL components. A control block system with more than 700 components is included with the aim to reproduce with accuracy the plant response to different transients.

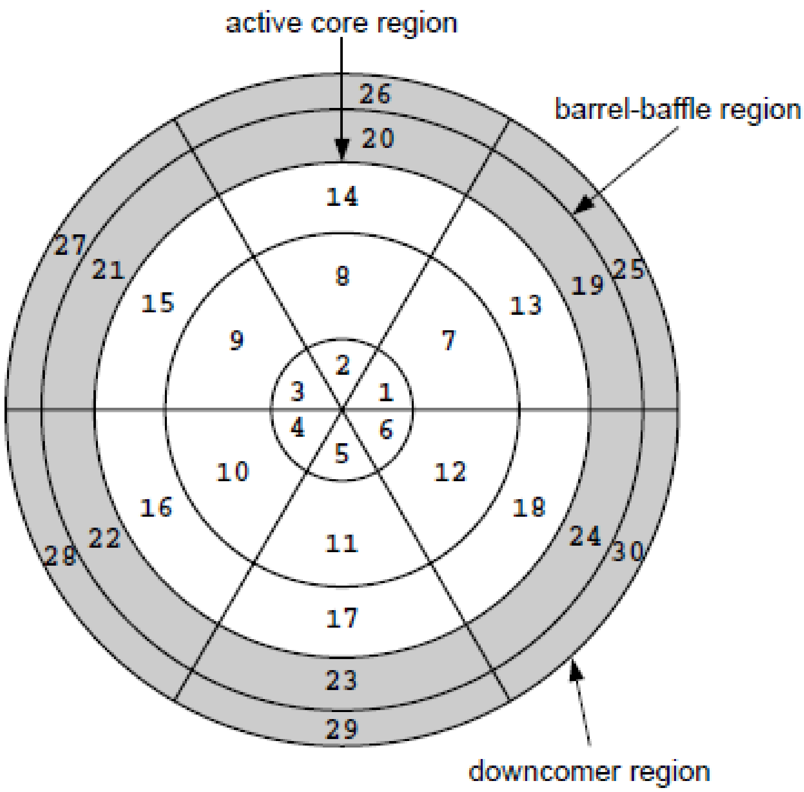

Figure 1. Axial layout of the vessel model

2.2 PARCS model

The core neutronics is modeled with PARCS v3.0. Each of the 157 fuel assemblies (all of them with a 17x17 pin array) is represented by one node in the radial plane. In total there are 221 nodes per axial level because 64 additional nodes are used to describe the reflector. Axially the fuel region is divided in 24 + 2 nodes, 24 for the core active region and 2 for the bottom and top reflectors. The height of the nodes is large in the central region and smaller in the lower and upper regions, in order to reproduce with more accuracy the flux and thus cross-section variation along the core length.

The cross section library, that considers the different types of fuel and burn-ups, and accounts for the 6 control rod banks, contains 648 + 2 different compositions. 648 refers to different kind of fuel while +2 refers to bottom/top and side reflectors. The two-group cross-section library has been generated with HELIOS-1.9 [12] using plant specifications [13].

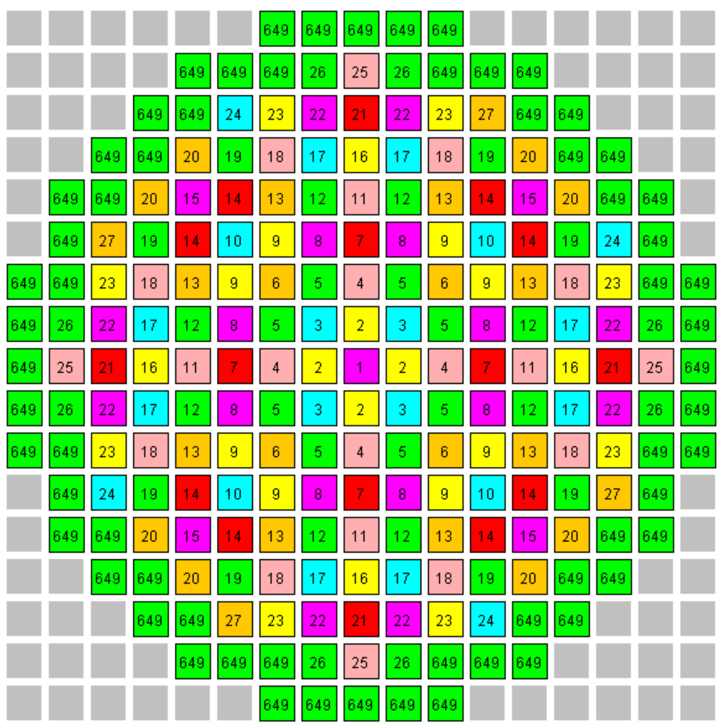

Figure 2. Radial core assembly layout

Figure 2 shows the fuel assemblies (FA) radial distribution in the core. There are 27 different types of FA with different enrichments (ranging from 2.1% to 4.55%). The newest FA contain burnable absorbers with varying Gd_2O_3 concentration from 2.0% to 8.0% depending on the FA. Cross-section library has been generated as a function of moderator temperature, fuel temperature, moderator density, boron concentration and control rods insertion. Extended ranges of change of the thermal-hydraulic feedback parameters have been selected in order to cover both initial steady-state and expected transient conditions.

2.3 Coupled model

The thermal-hydraulic model has been set up so as to meet the requirements of the neutronics model. There are 157 Heat Structures (HS) in the core region, each one representing one FA.

There are 18 radial thermal-hydraulic cells in each axial thermal-hydraulic layer in the core region. Figure 3 shows the assignment (mapping) in a radial plane of the active core and reflector TH cells to neutronics nodes. Every different color area represents a thermal-hydraulic cell. Regarding axial nodalization, the 24 non-equidistant axial nodes of the neutronics model are equivalent to those of the HS; however, only 6 nodes exist in the hydraulic model. Consequently the axial mapping between the HS and neutronics nodes is one to one, whereas several neutronic and HS nodes are linked to one single thermal-hydraulic node. Such thermal-hydraulic nodalization was chosen in order to be consistent with the previous works made with the non-coupled systems and also in order to easy compare with previous cases. Faster computing times were also among the reasons of such nodalization choice.

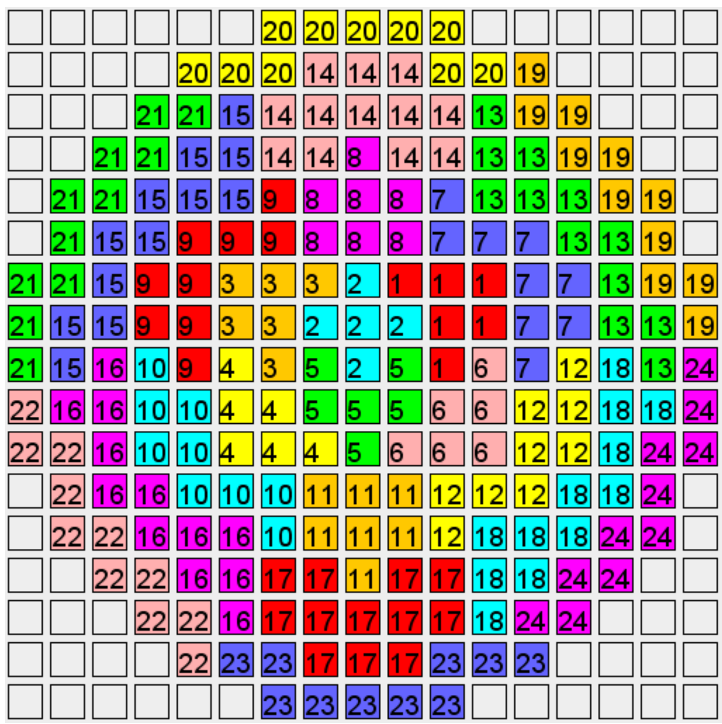

Figure 3. Core neutronics and HS nodes mapping to TH cells.

3. EVENT DESCRIPTION

A hypothetical MSLB is the transient chosen for the comparison of methodologies. The initiating event is a double ended break in the main steam line in loop 2. Right after the break, the high differential pressure between the steam lines causes the activation of the high pressure injection systems. At the same time the turbine and the reactor are shut down. Power during the transient gets reduced due SCRAM and safety injection systems. For this analysis it is postulated that a control rod remains stuck out during the scram. The asymmetric heat transfer to the secondary caused by the broken loop translates into temperature and coolant density asymmetries in the primary system. These density asymmetries are propagated into the core. Although some mixing occurs in the lower plenum, colder water circulates through the core region where the control rod remains on its fully withdrawn position, while SCRAM signal is on. Even when there is an increase of the total reactivity, mainly due to the density changes of the coolant (moderator) and to the fact that one of the control rods has not been inserted, the power stays low and decreases quickly, as it can be seen in Figure 4. Total reactivity contributing components are shown in Figure 5.

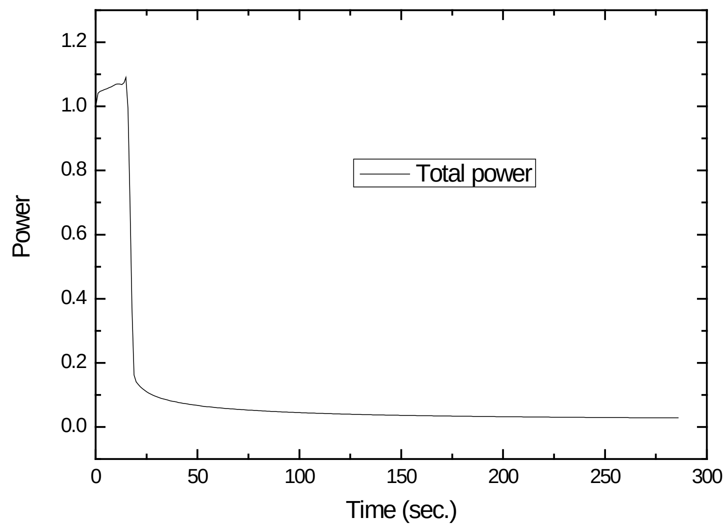

Figure 4. Best Estimate Base Case calculation (BEBCC) simulated by TRACE-PARCS.

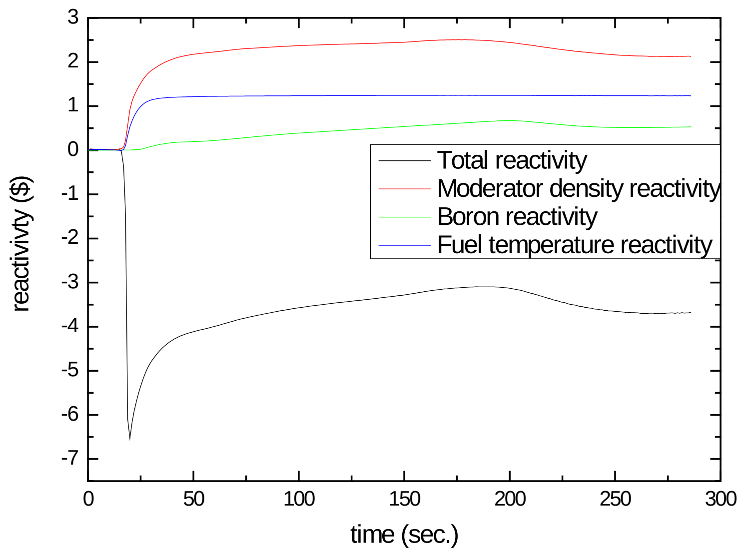

Figure 5. Total reactivity and its components in BEBCC with TRACE-PARCS.

4. CONSERVATIVE CASE CALCULATION

In order to check the adequacy of the Best Estimate Plus Uncertainty (BEPU) methodology, a Best Estimate Base Case Calculation (BEBCC) by using some conservative assumptions for important input parameters in order to ensure conservative calculation has been performed to be used as a reference for comparison purposes. Essentially, the conservative calculation differs from the (BEBCC) because some conservative assumptions are made in order to ensure the safety margins [14]. These conservative assumptions are related to boundary conditions Table 1. Figures 6 and 7 illustrate the total power and the total reactivity evolution for the conservative calculation. Conservative results, as expected, provide a less pronounced fall of power and reactivity than the BEBCC case.

Table 1. List of the conservative assumptions and modifications made to the Best Estimate base case model

Thermal hydraulic input TRACE	**Physical impact of the selected parameter modification to make it conservative**
Small delay on the pumps trip.	Increase of initial positive reactivity
Slightly increase (+5.0%) of pressure and temperature of the boundary conditions (BREAK components) which receive the fluid from the MSLB.	Increase of initial positive reactivity
Slightly increase (+5.0%) of the OPENING/CLOSE time from the valves system which are composing the MSLB nodalization.	Slow break flow
Slightly decrease (-2.0%) of the temperature from the ECCS system.	Delay moderator reactivity feedback
Slightly decrease (-2.0%)of the temperature from the FW system.	Delay moderator reactivity feedback
Neutronic input PARCS	
Initial status of the control bank D, six steps withdrawn with respect to the BE case	More conservative, introduce more reactivity
Bottom core length without Control rod (+5.0%) when the control rod is fully inserted.	More portion of the core with higher flux
(+2.64%) increase in the control rod step size.	Less accuracy on the rodded areas
Increase on the delay of the SCRAM signal by (+0.05) seconds.	Delay control rod reactivity feedback
Increase on the delay of the rod insertion time by (+0.5) seconds.	Delay control rod reactivity feedback
Increase power trip card, which defines the power level where SCRAM occurs.	Increase of initial positive reactivity

Note: All the above (%) values are computed over the nominal values.

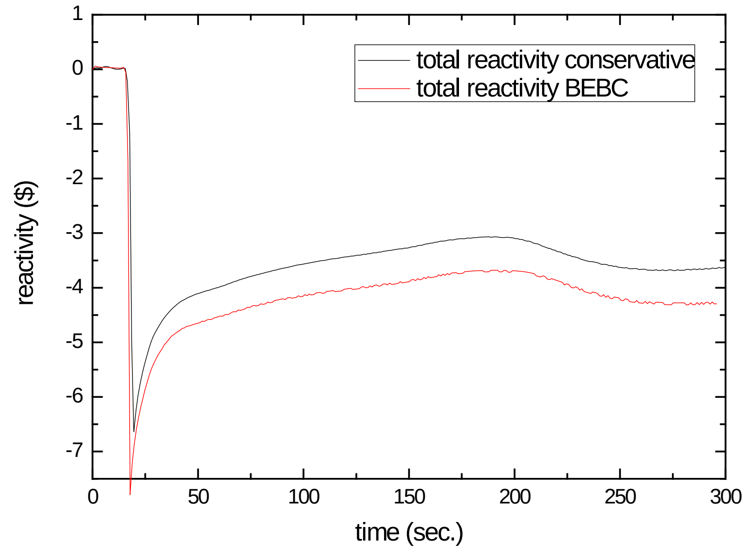

Figure 6. Conservative Case total reactivity time evolution

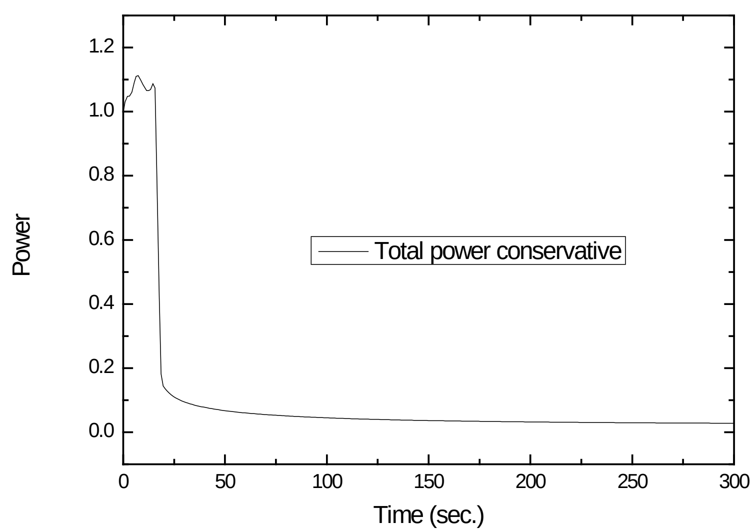

Figure 7. Conservative Case total power time evolution

5. BEST ESTIMATE PLUS UNCERTAINTY CALCULATIONS

The first of the BEPU calculations is a BE base case calculation, which consists on assuming that all the parameters have their nominal values. This calculation is done by first letting the coupled model achieve steady state and then initiating the transient event. Table 2 summarizes the most relevant events as simulated in the BE base case. Figures 4 and 5 show the evolution of power and reactivity for the BEBCC and in comparison with the conservative calculation.

Table 2. Sequence of events in MSLB transient

Time (s)	Event
15.00	Double-ended loop 2 Main Steam Line Break opens
15.05	High differential pressure between steam lines signal. Safety Injection Signal. SCRAM signal
15.50	Steam isolation
151.00	Steam generator 2 empties
195.00	Manual AFW turbo-pump trip
285.00	Manual regulation of AFW valves (15%)
300.00	End of simulation

For the uncertainty methodology, DAKOTA is applied after the BE transient computations. The general sequence of steps for performing an Uncertainty Analysis in a best-estimate model is outlined below:

1. Specify Uncertainty Analysis input such as sampling method, number of samples, etc.
2. Select the set of input parameters to be modified.
3. Assign probability distributions and range of variation to each input parameter.
4. Generate the sets of random variables.
5. Generate an input file for each set of random variables
6. Execute each case.
7. Extract response data from each case run.
8. Calculate uncertainty and sensitivity results.
9. Compile a report summarizing the Uncertainty Analysis

Some of the steps (like the selection of input parameters, definition and assignment of probability distributions, and input requirements given to the DAKOTA software such as the sampling method, number of samples, and the random seed) are user defined quantities. Other steps from the above list are the result of the calculations with the DAKOTA Uncertainty package.

The authors of the paper have compiled a list of one hundred thermal-hydraulic parameters plus forty neutronic parameters, which are relevant to PWR MSLB analysis, to be used as modified parameters in a BEPU calculation. Such list has been reduced based on Phenomena Identification and Ranking Tables (PIRTs), CRISSUE [2] reports and experts conclusions to a list of twenty two relevant parameters. Twelve thermal-hydraulic parameters and ten neutronic parameters are representative of the most relevant parameters to the MSLB transient in a PWR. The Tables 3 and 4 show the list of the twelve thermal-hydraulic parameters and ten neutronic parameters. Each table contains mean values for each parameter, Probability Density Functions (PDFs), standard deviations, Maximum and Minimum values in case there are any. In addition the reference [15] has been used to determine the parameters and their associated probability density functions.

Previous to the DAKOTA analysis and under the SNAP interface, an Extract Data step is required in order to retrieve data from coupled calculations and prepare them to be read and treated by DAKOTA. Extract Data step, where the data from the BEPU calculations is taken from the calculation data base to the DAKOTA procedure step, bridges the gap between analysis code outputs and the DAKOTA uncertainty input.

Table 3. Thermal-hydraulic parameters list

	TH parameter	Unit	Mean	PDFs	Standard deviation	Max	Min
BREAKS							
1	Initial mixture temperature	K	554.245	Normal	0	-	-
2	Initial pressure	Pa	1.00E+05	Normal	1.00E+04	-	-
FILLS							
3	Initial mixture temperature	K	333.15	Normal	30	-	-
4	Initial solute ratio (Boron solved into the moderator)	ppm	2200	Normal	500	-	-
PIPES							
5	Wall roughness	m	4.00E-05	Normal	4.00E-06	-	-
JUNCTIONS							
6	Wall roughness	m	4.00E-05	Normal	4.00E-06	-	-
TEE's							
7	Wall roughness	m	4.00E-05	Normal	4.00E-06	-	-
VALVES							
8	Wall roughness	m	0	Normal	0	-	-
9	Maximum valve rate (speed opening/close valve)	1/s	10	Normal	1	-	-
10	Min position	-	0	Normal	0.05	-	0
11	Max position	-	1	Normal	0.05	1	-
HEAT STRUCTURES							
12	Gas GAP (heat transfer coefficient)HTC	$W/m^2/K$	1.14E+04	Normal	500	-	-

Table 4. Neutronic parameters list

	Neutronic parameters	Unit	Mean	PDFs	Standard deviation	Max	Min
1	Rod insertion	cm	110.37	Normal	5.52	-	-
2	Rod step	cm	1.579	Normal	0.079	-	-
3	Control rod banks positions	steps	196	Normal	5	-	-
4	Core power to initiate trip	%	100	Normal	10	-	100
5	SCRAM signal delay time	s	0.2	Normal	1	-	0
6	Rod insertion time	s	2.2	Normal	0.5	-	-

7	Time step size	s	0.07	Normal	0.05	0.5	1.00E-03
8	XS change criterion for requiring at least one nodal update	-	0.01	Normal	5.00E-04	-	-
	Transient convergence parameters						
9	Local fission source convergence criterion	-	1.00E-04	Normal	5.00E-06	-	-
10	Fuel temperature convergence criterion	-	1.00E-06	Normal	5.00E-08	1.00E-05	1.00E-07

Every parameter from the tables above is treated by DAKOTA, under the specifications given by the user. After author's choices, DAKOTA gives to every parameter a different value for each case calculation. The new values depend on the PDFs and on the standard deviation of each initial parameter value. Figure 9 illustrates the wide range of selected values for the rod insertion time in this case. Notice that all the values oscillate around one mean value of 2.2 seconds in this case.

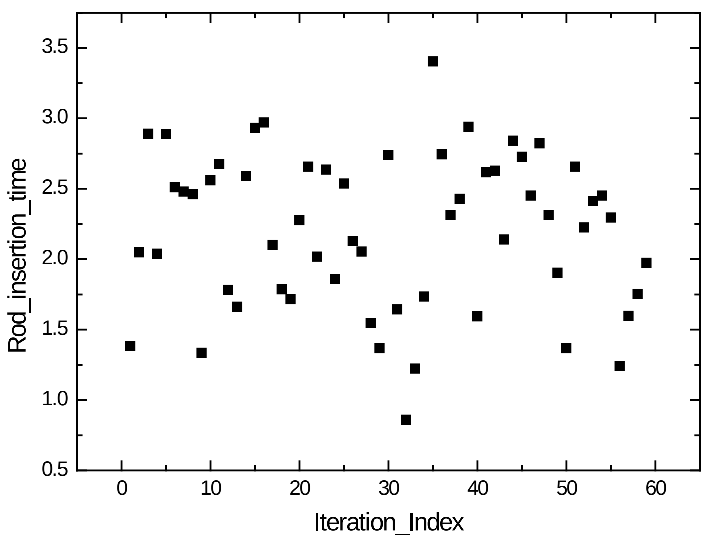

Figure 8. Illustration of the randomness of the uncertainty simulation.

The Uncertainty Analysis methodology uses Wilks [16, 17] method for computing sample sizes. The method is used to determine a number of random samplings that must be made to assure a certain degree of confidence that a given probable range of inputs have been covered. In order to illustrate the methodology, for the present calculations, 95% of probability and 95% of confidence have been considered. Figure 9 and Figure 10 show the total reactivity distribution and total core power distribution against time, for the 59 cases used in the BEPU methodology.

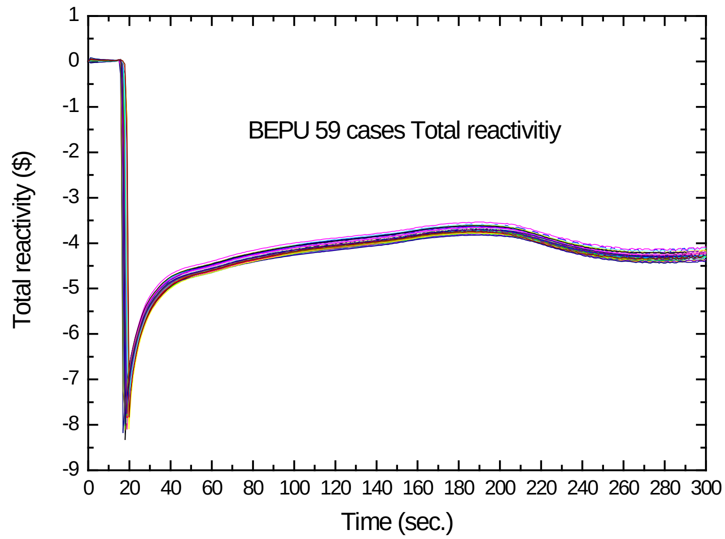

Figure 9. BEPU calculation total reactivity results

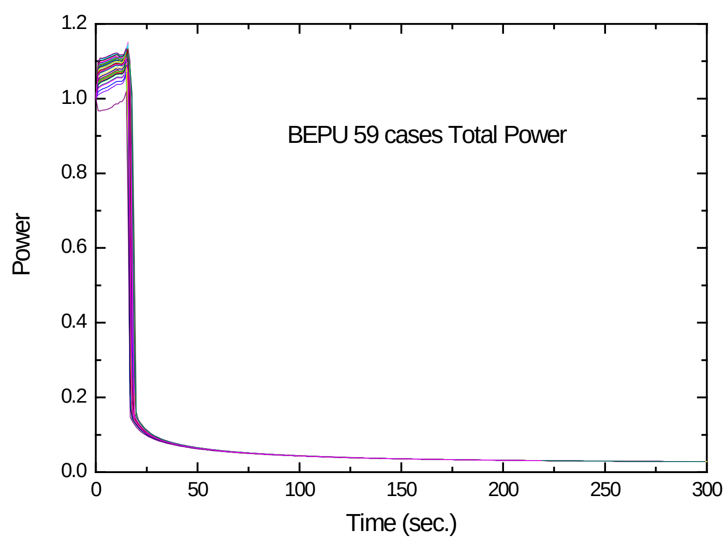

Figure 10. BEPU calculation total power results

6. COMPARARING BEPU with CONSERVATIVE CALCULATION

Finally, BEPU calculations are compared to the Conservative Case calculation in order to provide an insight into the advantages of the former methodology. BEPU calculations provide a range of values covering the 5 to the 95 percentile with 95% confidence (in this case) for each time step of the calculation, such range has been compared with conservative calculation in Figure 11. BEBCC is plotted in the same figure, such calculation falls within the BEPU margins.

The power time evolution figure has been omitted due to the narrowest BEPU margins in it. Same comparison can be done in terms of the local temperature. This comparison will be more interesting if it is done for the region where the control rod remains stuck out. The peak temperatures in this core region will rise higher and the comparison between BEPU, BE and conservative methodologies will be more significant there.

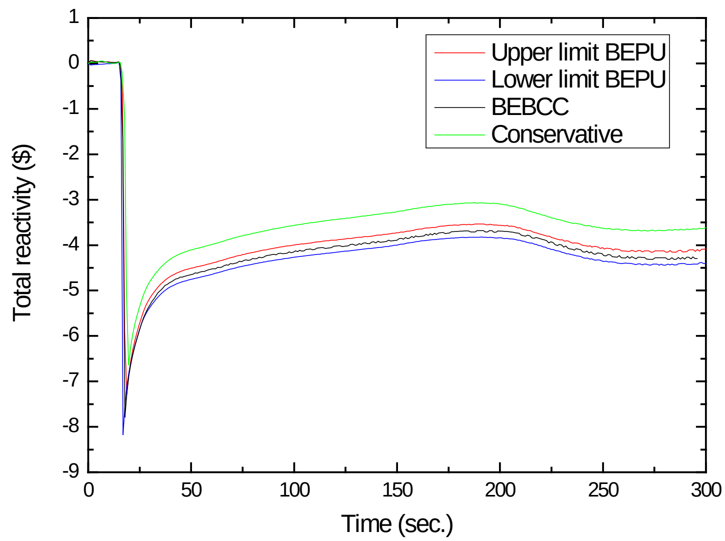

Figure 11. Comparison between Conservative and BEPU calculations

8. CONCLUSIONS

The general conclusion of this study is that the introduced BEPU method has been successfully illustrated. The study shows the interactions between thermal-hydraulic and neutron-kinetic codes in the context of BEPU analysis. The study is an effective test for the adequacy of nodalizations for the utilized codes both neutronic and thermal-hydraulic. While multi-physics approaches are still pending of specific developments, this work helps checking the consolidation of crucial aspects like the interface between codes. The development shows a maturity in combining core with system behavior. From now on, further development can focus on pending aspects. A second general conclusion can be found in the consistency between the results of BEPU calculations compared with conservative ones. Several additional conclusions are obtained from the comparison of the two methodologies used in the paper: Conservative and BEPU. The first conclusion is concerning the width of the BEPU range: such range of values seems to be very narrow. Increasing the number of modified parameters in the BEPU methodology and redefining some of the PDFs will lead to a wide range of values into the BEPU range; this applies as well to neutronic parameter, not considered in the present uncertainty calculations. More global parameters than total power and total reactivity could be compared but since this paper aims to illustrate the methodology, the authors agreed to limit the number of uncertain parameters. BEBCC falls into the middle of the BEPU calculations as it was expected. In terms of the total reactivity, the comparison between Conservative Case and Best Estimate Plus Uncertainties calculation concludes that the BEPU methodology gives more margin, and will probably allow for higher operational flexibility of the plant, which can be helpful to improve plant economics while meeting the safety standards. Further works might show higher safety margins in terms of wider BEPU bands. Different transient analysis such anticipated transients without SCRAM or any transient where asymmetries (in terms of fluxes) in the core are relevant will reinforce the use of BEPU methodology with thermal-hydraulics and neutron kinetics coupled calculations.

9. ACKNOWLEDGEMENTS

The authors of this paper express their appreciation for the support offered by ANAV *Asociación Nuclear Ascó Vandellòs* in providing the plant data from Ascó NPP.

10. REFERENCES

[1] US Nuclear Regulatory Commission OECD Nuclear Energy Agency. "*PRESSURISED WATER REACTOR MAIN STEAM LINE BREAK (MSLB) BENCHMARK, Volume I, Volume II and Volume III.*

[2] Neutronics/Thermal-hydraulics Coupling in LWR Technology, Vol.1, Vol.2 and Vol.3, CRISSUE-S - WP1, WP2 and WP3. 5th Euratom Framework Programme OECD/NEA 2004.

[3] US NRC. *TRACE V5.0 THEORY MANUAL. Volume 1: Field Equations, Solution Methods, and modelling Techniques.*

[4] T. Downar, Y. Xu, V. Seker, PARCS v3.0, U.S. NRC Core Neutronics Simulator "*USER MANUAL*" Department of Nuclear Engineering and Radiological Sciences University of Michigan Ann Arbor. December, 2009

[5] Brian M. Adams, Keith R. Dalbey, Michael S. Eldred, Laura P. Swiler Optimization and Uncertainty Quantification Department William J. Bohnhoff Radiation Transport Department John P. Eddy System Readiness and Sustainment Technologies Department Dena M. Vigil Multiphysics Simulation Technologies Department. DAKOTA, "*A Multilevel Parallel Object-Oriented Framework for Design Optimization, Parameter Estimation, Uncertainty Quantification, and Sensitivity Analysis Version 5.2 Manual*" Sandia National Laboratories

[6] R. Pericas "*Contribution to the validation of best estimate plus uncertainties coupled codes for the analysis of NK-TH nuclear transients*" Ph.D Thesis. Universitat Politècnica de Catalunya. Barcelona, May 2015.

[7] F. Reventós, C. Llopis, L. Batet, C. Pretel, I. Sol, "*Analysis of an actual reactor trip operating event due to a high variation of neutron flux occurring in the Vandellòs-II nuclear power plant*" Nuclear Engineering and Design, Volume 240, Issue 10, October 2010, Pages 2999-3008, ISSN 0029-5493

[8] F. Reventós, L. Batet, C. Llopis, C. Pretel, and I. Sol, "*Thermal-Hydraulic Analysis Tasks for ANAV NPPs in Support of Plant Operation and Control,*" Science and Technology of Nuclear Installations, vol. 2008, Article ID 153858, 13 pages, 2008. doi:10.1155/2008/153858

[9] F. Reventós, L. Batet, C. Llopis, C. Pretel, M. Salvat, I. Sol, "*Advanced qualification process of ANAV NPP integral dynamic models for supporting plant operation and control" Nuclear* Engineering and Design, Volume 237, Issue 1, January 2007, Pages 54-63, ISSN 0029-5493

[10] F. Reventós, L. Batet, C. Pretel, M. Ríos, I. Sol, "*Analysis of the Feed and Bleed procedure for the Ascó NPP: First approach study for operation suport*" Nuclear Engineering and Design, Volume 237, Issue 18, October 2007, Pages 2006-2013, ISSN 0029-5493

[11] R. Pericas ; Reventós F.; Batet L.; "*Sensitivity Analyses of a hypothetical 6 inch break, LOCA in Ascó NPP using RELAP/MOD3.2*." NUREG/IA-243. 2007

[12] HELIOS 1.9 manuals and documentation, Studsvik Scandpower, November 2005, Norway.

[13] ENUSA. ITEC-779 Rev-2 "*Informe de diseño nuclear del ciclo 13 de la central nuclear Ascó II*". December 1999

[14] Ivan Gajev. "*Sensitivity and Uncertainty analysis of Boiling water reactor stability simulations*". Ph.D thesis Stockholm Sweden 2012.

[15] Code of Federal Regulations. "*Acceptance criteria for emergency core cooling systems for light water nuclear power reactors*". Appendix K; 10 CFR 50.46, March 1996.

[16] S. S. Wilks "*Determination of Sample Sizes for Setting Tolerance*" Limits Princeton University Annals of Mathematical Statistics, vol. 12; no. 1; pp. 91-96, March 1941.

[17] S. S. Wilks "*Statistical prediction with special reference to the problem of tolerance limits*". Annals of Mathematical Statistics, vol. 13; no. 4; pp. 400-409, March 1942.