Acta Polytechnica CTU Proceedings **44**:13–18, 2023
https://doi.org/10.14311/APP.2023.44.0013

EVALUATION OF ATLHAMC12 SUBCHANNEL CODE FOR TOTAL LOSS OF FLOW SCENARIO

Tomáš Křeček[a], Štěpán Foral[b, a, *]

[a] *Brno University of Technology, Faculty of Electrical Engineering and Communications, Department of Electrical Power Engineering, Technická 12, 616 00 Brno, Czech Republic*

[b] *TES s.r.o., Pražská 597, 674 01 Třebíč, Czech Republic*

[*] corresponding author: foral@tes.eu

Abstract. The ALTHAMC12 subchannel code is a new subchannel code developed by the ALVEL company. The code is intended for DNBR safety analyses of the Czech nuclear power plants. In order to validate the code, a code to code comparison with THALES and VIPRE-01 is provided in this work. The reactor core model was developed and set of initial and boundary conditions has been adopted from a reference study. The comparison is done for steady state nominal parameters and Total Loss of Flow (TLOF) type of accident. The results show that ALTHAMC12 provides a good agreement with the reference codes in the terms of MDNBR value and its positions in the reactor core.

Keywords: ALTHAMC12, subchannel analyses, APR1400, PLUS7, CE-1.

1. Introduction

The core of a nuclear reactor is designed and operated in a way that ensures sufficient heat dissipation resulting from fission and complies with all prescribed temperature limits for fuel cladding and fuel pellets at all locations within the core. The temperature limit for the reactor core structure is often determined based on the limit of fuel cladding temperature, which is defined by the maximum permissible heat flux at the coolant-fuel cladding interface.

Subchannel analysis is performed to determine the coolant parameters within the reactor core and it plays an important role in determining the safety limits of nuclear reactor operation, for example, the determination of Minimum Departure from Nucleate Boiling Ratio (MDNBR) which is one of the essential factors for assessing the thermal safety margin of a pressurized water reactors.

The principle of subchannel analysis is that the core is divided into several so-called subchannels in the radial direction. The subchannels are then divided in the axial direction and so a matrix of nodes is created. During the calculation, the equations of conservation of mass, momentun and energy are solved when the initial and boundary conditions are imposed in each node.

The ALTHAMC12 subchannel code is developed by the Czech company ALVEL, it is primarily designed for Czech nuclear power plants (VVER type). The ALTHAMC12 code includes several Critical Heat Flux (CHF) correlations, such as PG-S, PI3, PG-I, OKBM-Bezrukov and EPRI correlations. In the case of the first four correlations, these are the correlations primarily used for VVER reactors. For the PLUS7 fuel design used in the APR1400 nuclear reactors, KNF has developed the KCE-1 correlation on the basis of the CE-1 correlation, but KCE-1 details are not publicly available [1].

The primary objective of this paper is to utilize the ALTHAMC12 subchannel code and documentation from USNRC [2] to construct a subchannel model of the APR1400 reactor core, specifically with PLUS7 fuel. Then perform a comparative analysis to compare the ALTHAMC12 results obtained with the results published in the reference study. [3]

2. APR1400 reactor core model

In the subchannel code ALTHAMC12, a simplified model of the APR1400 reactor core was created which is in compliance with the model developed in VIPRE-01 in reference study [3]. Figure 1 shows the radial nodalization and subchannel numbering of the model. The model is simplified in the axial direction to include only the fuel portion. So, the modelincludes all 241 fuel assemblies, limited to active fuel length. The internal structures of fuel assemblies are not a part of the model except for grids. The basic parameters of the model are given in Table 1. All the geometry data of fuel assembly and core layout were adopted from [2].

Due to the lack of some technical data in [3], simplifications were made in the case of modeling of the fuel assemblies. The simplifications are basically identical to those used in the VIPRE-01 core model. There are 236 fuel rods in the fuel assembly. These rods have been replaced by one fictitious rod that is surrounded by coolant. The gap between fuel assemblies is divided among adjacent subchannels. Thus, each fuel assembly is represented by a single so-called “rod-centered” subchannel type. Next, the gap between the outermost fuel assembly and the core shroud is larger than the gaps between the central fuel assemblies, so each

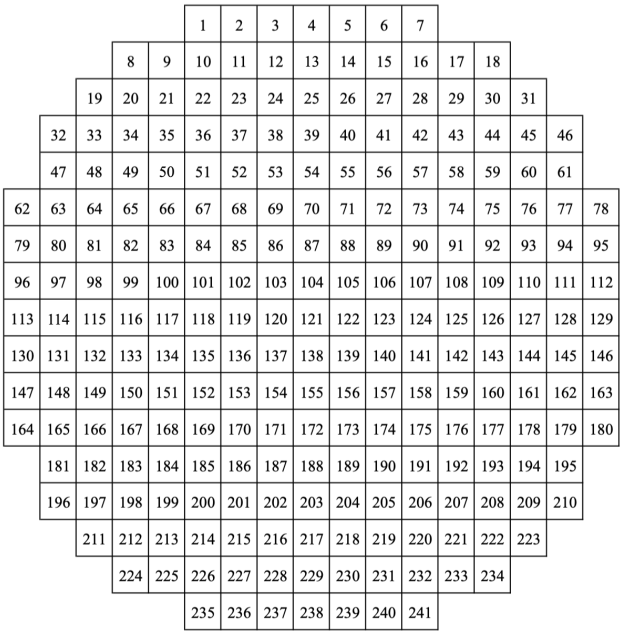

Figure 1. Radial nodalization of APR1400 reactor core model of ALTHAMC12 and VIPRE-01 codes.

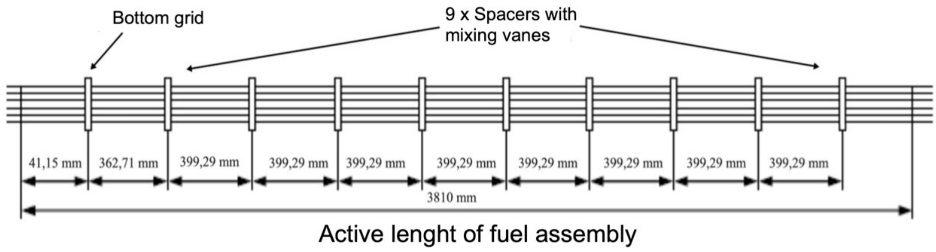

Figure 2. Position of spacers in the PLUS7 fuel assembly.

Parameter	**Value**
Number of subchannels	241
Number of axial nodes	50
Axial node length (equal division) [mm]	76.2
Number of junctions	448
Energy generated in coolant [%]	0.0
Surface roughness [µm]	6.0
Turbulent mixing coefficient [-]	0.038
Grid pressure loss coefficient (SMV/BG)	1.0/0.7

Table 1. Main parameters of the APR1400 reactor core model.

outer subchannel should have a slightly larger area. In fact, the larger gap is ignored. Keeping the same area for the outermost fuel assemblies is expected to have a negligible impact on the MDNBR results because MDNBR is expected to occur in the central most loaded fuel assemblies.

Based on data in [1], the axial position of spacers and one bottom grid was determined. The Figure 2 shows the geometry of the fuel assembly with location of the bottom grid and spacers with mixing vanes. The pressure loss coefficients of grids are not available for public so the approach from [4] is adopted. For Spacers with Mixing Vanes (SMV), the grid pressure drop coefficient is 1.0 and for bottom grid (BG) is 0.7, as presented in Table 1.

Due to the type of radial nodalization, the developed APR1400 reactor core model cannot be used for detail MDNBR analyses. The subchannels determine the coolant properties which are averaged for the whole FA. Contrary to that, the CE-1 correlation was developed on the basis of local coolant data measured between the fuel rods (the description of the CE-1 correlation is provided in [4]) and so it can be used only with detail model of FA. In other words, the combination of the CE-1 correlation and ALTHAMC12 core model would yield inaccurate predictions of MDNBR. On the other hand, the core model developed in ALTHCAM12 can be used for code-to-code comparison as intented in this study. Moreover, it can be used for determination of cross-flows between the fuel assemblies. This can be used then as boundary condition for MDNBR calculation in combination with detail subchannel model of FA.

Parameter b_i	Value
b_1	$2.8922 \cdot 10^{-3}$
b_2	-0.50749
b_3	405.32
b_4	$-9.9290 \cdot 10^{-2}$
b_5	-0.67757
b_6	$6.8235 \cdot 10^{-4}$
b_7	$3.1240 \cdot 10^{-5}$
b_8	$-8.3245 \cdot 10^{-2}$

TABLE 2. Parameters of the CE-1 critical heat flux correlation.

3. CE-1 critical heat flux correlation

The CE-1 correlation developed by Westinghouse is used for evaluation of MDBNR. This correlation has the same mathematical formula as the KCE-1 correlation which was derived directly for PLUS7 fuel design [1] but is available for public. The CE-1 correlation is defined by the following equation [5]:

$$q''_{CHF} = \frac{b_1\left(\frac{d}{d_m}\right)^{b_2}\left((b_3 + b_4 p)\, G^{(b_1 p + b_8 G)} - G H_{fg} X\right)}{G^{(b_7 p + b_8 G)}}, \tag{1}$$

where q''_{CHF} is critical heat flux, in BTU/ft^{2}hr, p is pressure in psia, d is heated equivalent diameter of the subchannel in inches, d_m is heated equivalent diameter of a matrix subchannel with the same rod diameter and pitch in inches, G is local mass velocity at CHF location in lb/hr-ft, x is local coolant quality at CHF location in decimal fraction, h_{fg} is latent heat of vaporization in BTU/lb. Parameters b_1 to b_8 are given in Table 2.

To consider the non-uniform distribution of the axial heat flux, the correction factor FS is used, which is defined as [5]:

$$FS = \frac{q''_{CHF,\text{Equiv Uniform}}}{q''_{CHF,\text{Non-uniform}}}, \tag{2}$$

$$FS_{(J)} = \frac{C_{(J)}}{q''_{CHF,\text{Non-uniform}} \, (1 - e^{-C_{(J)} \cdot x_{(J)}})} \cdot \int_0^{X_{(J)}} q''(x) e^{-C_{(J)} \cdot (x_{(J)} - x_{(J-1)})} dx, \tag{3}$$

where for CE-1 CHF correlation:

$$C_{(J)} = 1.8 \cdot \frac{(1 - x_{CHF})^{4.31}}{G^{0.478}}. \tag{4}$$

The index (J) represents the addressed node. DNBR in node (J) is then calculated as:

$$DNBR_{(J)} = \frac{1}{FS_{(J)}} \cdot \frac{q''_{CHF,\ Equiv\ Uniform}}{q''_{(J)}}. \tag{5}$$

Since the ALTHAMC12 code does not include the CE-1 correlation, an EXCEL file was created to post-process the output data when considering the CE-1 correlation in MDNBR analyses.

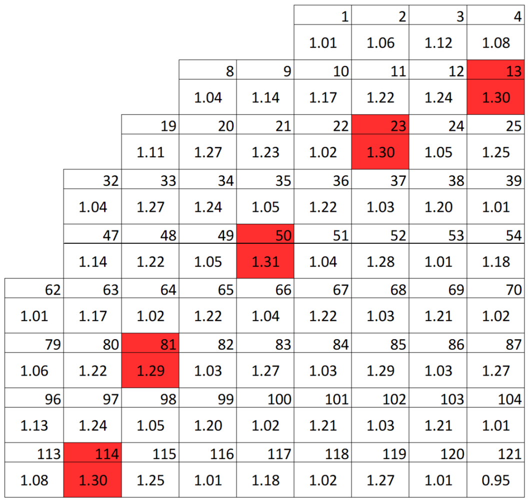

FIGURE 3. Core radial power distribution for MDNBR analyses [3].

4. Validation of core model

As recommended in SRS-23 [6], the model should be validated against steady state and transient data. The created model is validated using reference data from reference study [3] by code-to-code comparison. The reference study is focused on comparison between two subchannel codes, THALES and VIPRE-01. In both codes, the CE-1 critical heat flux correlation is also used to calculate the critical heat flux. The mathematical and physical models used in the codes, including ALTHAMC12 are compared in Table 3. In turbulent mixing models, a and b are constants for turbulent mixing correlation, Re is Reynold's number, $\overline{G}$ is averaged flow rate, $\overline{D_h}$ is average hydraulic diameter, s is the flow gap between subchannels, n is the total number of subchannels along one side a fuel assembly, and w is the gap width for one side of fuel assembly. Further, it can be seen in Table 3 that ALTHAMC12 shares many parameters with the VIPRE-01 code.

Core power distribution is based on Shinkori 3&4 design from reference study [3]. Radial power factors are presented in Figure 3. Fuel assemblies with maximum values of radial power factors are shown in red color as these are expected to experience MDNBR. Only quarter of the reactor core is shown in Figure 3 as the radial power map is symmetrical. There are five core axial power distributions presented in Figure 4 which are used in the MDNBR analysis, and which reflect the impact of fuel burnup. At the same time, Figure 3 shows the nodalization of the VIPRE-01 model meanwhile the nodalization of the THALES model is identical to the ALTHAMC12 model.

5. Code to code comparison at steady-state conditions

Data for steady-state were adopted from reference study [3] and they are shown in Table 4. Following the methodology presented in reference [3], the ASI-1

Model / Correlation	ALTHAMC12	VIPRE-01	THALES
Node division modelling	Channel only	Channel only	Subchannel and Channel
Turbulent mixing	$aRe^b\overline{G}\,\overline{D_h}$	$aRe^b\overline{G}\,\overline{D_h}$	$aRe^b\overline{G}\,\overline{D_h}s(12n/w)$
Two phase friction multiplier	Armand	Armand	Sher-Green and Martinelli-Nelson
Void model	Madsen	Armand	Armand
Flowing quality	Equilibrium	Equilibrium	Equilibrium
Subcooled nucleate boiling	Jens-Lottes	Jens-Lottes	Jens-Lottes
Thermal diffusion coefficient	Design Value	-	Design Value
CHF correlation	CE-1	CE-1	CE-1
Core inlet flow	Distributed	Distributed	Distributed
Core outlet pressure	Uniform	Uniform	Distributed

Table 3. Comparison of models and correlations between codes [3, 7].

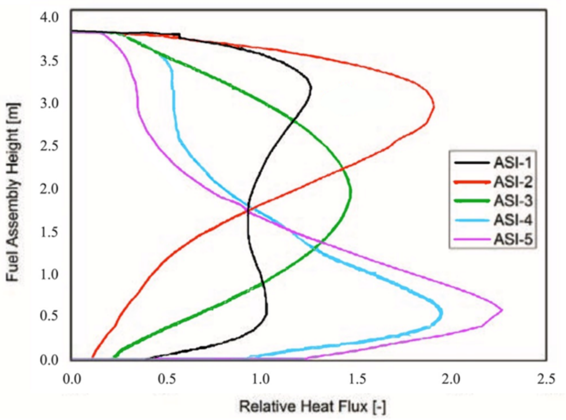

Figure 4. Core axial power distribution for MDNBR analysis [3].

Parameter	Value
Thermal power [MW_t]	3983
Pressure [MPa]	15.51
Core inlet temperature [°C]	291
Mass flow [kg/m^2s]	3496

Table 4. Initial conditions for validation at steady-state [3].

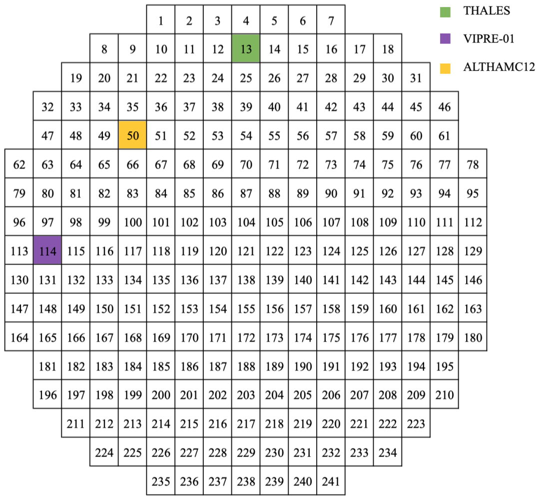

Figure 5. Position of subchannels with MDNBR for steady-state conditions.

Parameter	THALES	VIPRE-01	ALTHAMC12
MNDBR [-]	2.957	2.954	3.003
Subchannel	13	114	50
Elevation [mm]	3357	3276	3276

Table 5. Comparison of MDNBR parameters for steady-state conditions.

axial power profile shown in Figure 4 was selected for calculations. The reference radial power distribution is taken from Figure 3.

The results of calculation MDNBR between the THALES and VIPRE-01 computational codes are published in reference study [3] as well. The comparison with the ALTHAMC12 code is presented in Table 5 and in Figure 5. In steady-state conditions, the MDNBR value calculated by the THALES code was 2.957 in subchannel number 13, while the VIPRE-01 code yielded a calculated value of 2.954 in subchannel number 114, which is symmetric to subchannel number 13 as shown in Figure 3. The results show that the MDNBR results obtained with the the ALTHAMC12 code align well with the values in the reference study. In the axial position, MDNBR matches with the VIPRE-01 code (axial node no. 41), but the axial position of MDNBR in the THALES code is very similar (axial node no. 42). Since the power of the fuel assemblies no. 13, 50, and 114 is almost the same, the difference in the predicted MDNBR radial position in the core may be attributed to a different coolant overflow betwen FAs. The most probable source of the difference are the values of the grid loss coefficients which had to be estimated in the ALTHAMC12 core model.

6. Code to code comparison at transient conditions

As the transient scenario for validation, the Total Loss of Flow Accident (TLOFA) was selected. The TLOFA in the reactor is a result of a simultaneous loss of electrical power to all main coolant pumps

Parameter	Value
Heat Power [MW_t]	4062.7
Pressure [MPa]	16.03
Core inlet temperature [°C]	287.8
Mass flow [kg/s]	23619

Table 6. Initial conditions for validation at TLOFA [3].

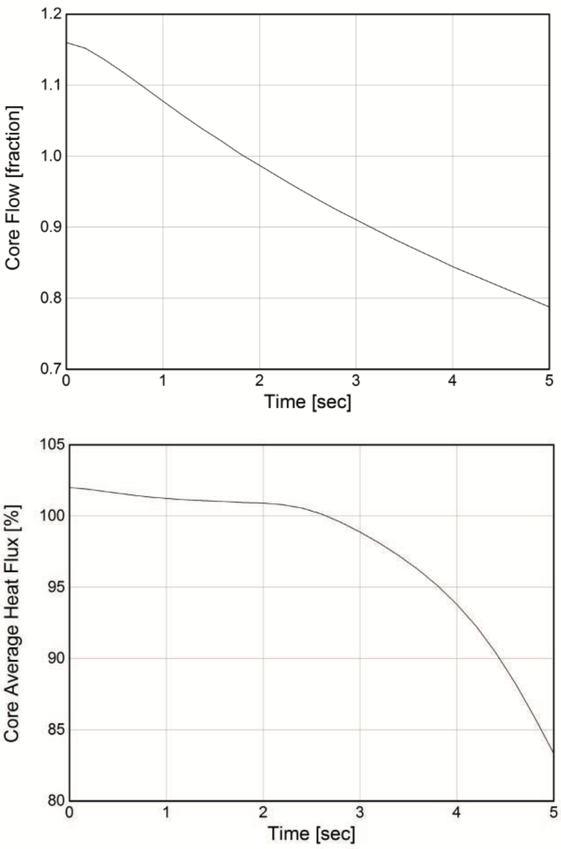

Figure 6. Core coolant flow rate (top) and average heat flux (bottom) during TLOFA [3].

(MCPs). The only fault that can lead to this situation is the total loss of power to the unit. During TLOFA, DNBR will manifest itself during the first few seconds of the transient phase as coolant flow decreases but the reactor has not yet been completely shut down. Reactor protection system generates a signal to shut down the reactor if the speed of any of the four MCPs drops below 95 % of their nominal speed. This rapid reactor shutdown provides reasonable assurance that MDNBR caused by the event will remain above the specified acceptable fuel design limit for DNBR. [8]

To perform the MDNBR analysis of the TLOFA, it is necessary to determine the initial conditions. For a conservative approach, initial conditions are chosen to simulate the worst-case initial scenario for a given accident or event, deviating from the nominal reactor operating parameters (higher power, higher/lower flow, etc.). These initial and boundary conditions were selected to be the same as in reference [3]. The initial conditions are presented in Table 6.

Parameter	THALES	VIPRE-01	ALTHAMC12
MDNBR [-]	2.953	2.876	2.949
Subchannel	13	50	50
Elevation [mm]	3357	3276	3276

Table 7. Comparison of MDNBR parameters for TLOFA.

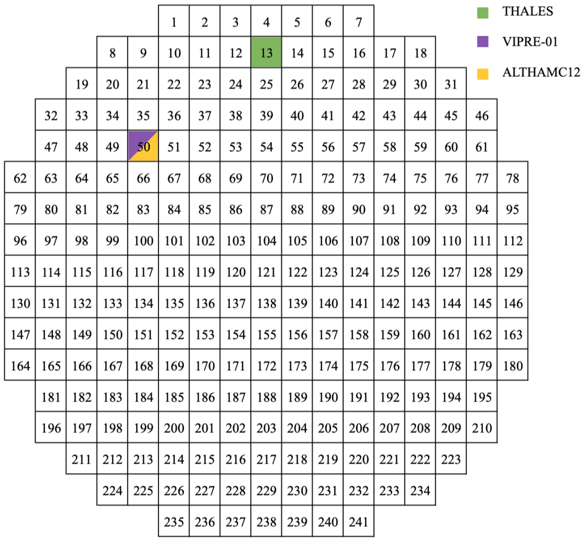

Figure 7. Position of subchannels with MDNBR for steady-state conditions.

The coolant inlet pressure and temperature are kept constant for five seconds to give conservatism. The core average heat flux and mass flow rate change during the transient. Their evolution are shown in Figure 6 for the first five seconds of the transient.

In the calculation, the axial power profile ASI-1 from Figure 3 and the maximum radial power distribution from Figure 4 were used again.

Comparison of calculated MDNBR is presented in Table 7. Figure 7 then shows the position of subchannels with MDNBR and Figure 8 shows the axial profiles of MDNBR. As it can be seen in Figure 8, the MDNBR profile calculated by the ATLHAMC12 code closely matches the MDNBR profile calculated in the reference study using the THALES code. Only slight deviations in the curve are observed at the beginning and end of the transient, but the difference between the curves does not exceed 2.5 %. The axial positions of MDNBR are very similar for all three codes. VIPRE-01 aligns with ALTHAMC12 both in terms of axial position and subchannel location, specifically subchannel number 50 and axial position of 3 276 mm corresponding to axial node number 41.

The differences in MDNBR results may be explained not only by the grid loss coefficients as explained by the steady state DNBR results. The outlet pressure distribution may have some influence on the cross flow distribution as well and this is the feature which is taken into account only by the THALES code, as referred in Table 3.

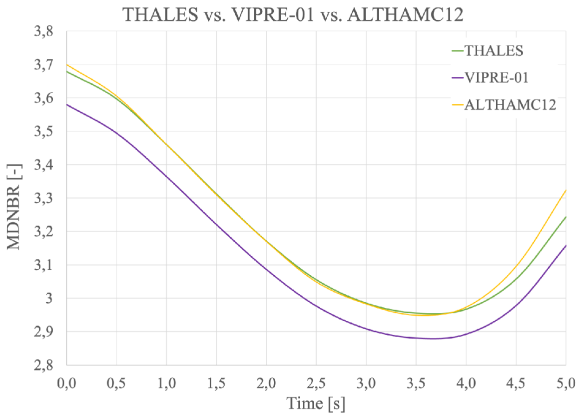

Figure 8. Results of MDNBR calculation during TLOFA.

7. Conclusion

In this study, the APR1400 reactor core model developed in ALTHAMC12 code. The model was validated using the code-to-code comparison. The MDNBR results predicted by the ALTHAMC12 were compared with reference results for steady state and transient conditions. The steady state validation results come out very similar to the results calculated using the THALES and VIPRE-01 codes in the reference study: $\mathrm{MDNBR}_{\mathrm{THALES}} = 2.957$, $\mathrm{MDNBR}_{\mathrm{VIPRE\text{-}01}} = 2.954$, and $\mathrm{MDNBR}_{\mathrm{ALTHAMC12}} = 3.003$. The ALTHAMC12 predicted MDNBR in the most loaded FA meanwhile the other codes predicted MDNBR in the FAs with the second highest relative power.

The scenario for transient validation analysis was the total loss of coolant flow. It can be concluded on the basis of results that the validation for this transient also performs well. VIPRE-01 predicts the lowest MDNBR values during the analyzed period of time. The minimal $\mathrm{MDNBR}_{\mathrm{VIPRE\text{-}01}}$ was 2.88 at the time of 3.5 second. The MDNBR results for ALTHAMC12 and THALES are almost identical with a slight difference occurring at the beginning and at the end of the analyzed transient. The minimal MDNBR of both codes is 2.95 and occurs at the time 3.5 second, which is the same time as predicted by VIPRE-01. THALES predicts the minimal MDNBR in the most loaded FA meanwhile the other codes predicted MDNBR in the FAs with the second highest relative power.

List of symbols

DNBR Departure from Nucleate Boiling Ratio

FA Fuel Assembly

MDNBR Minimum Departure from Nucleate Boiling Ratio

MCP Main Coolant Pump

Acknowledgements

Research described in this paper was supported by the Technology Agency of the Czech Republic, program THETA 4, under grant TK04030168, project "Tolerant Nuclear Fuels for Small Modular Reactors and Safety Evaluation".

References

[1] KEPCO. KCE-1 critical heat flux correlation for PLUS7 thermal design. [2021-12-23]. `https://www.nrc.gov/docs/ML1301/ML13018A147.pdf`

[2] KEPCO. APR1400 Design Control Document Tier 2, Chapter 4: Reactor, rev. 3. [2023-06-06]. `https://www.nrc.gov/docs/ML1822/ML18228A652.pdf`

[3] K. H. Moon, E. Ozdemir, S. J. Oh, Y. Kim. Evaluation of THALES subchannel code behavior for loss of flow and RCP rotor seizure scenarios. [2023-07-10]. `http://glc.ans.org/nureth-16/data/papers/13939.pdf`

[4] S. B. Kim. *Improvement of Subchannel Scale Analysis Capability of CUPID with Grid-directed Cross Flow and Fuel Rod Models*. Diploma thesis, Seoul National University, Gwanak-ro, Gwanak-gu, Seoul, Korea, 2018.

[5] Combustion Engineering. *CETOP-D Code Structure And Modeling Methods For San Onofre Nuclear Generating Station Units 2 And 3*, 1981. [2023-06-06]. `https://www.nrc.gov/docs/ML2000/ML20004D823.pdf`

[6] IAEA. *Accident Analysis for Nuclear Power Plants SRS no. 23*, 2002.

[7] ALVEL. *ALTHAMC12 Dokumentace programu, Document # 17-20A-005-001*, 2018.

[8] KEPCO. APR1400 Design Control Document Tier 2, Chapter 15: Transient and accident analyses, rev. 3. [2023-06-06]. `https://www.nrc.gov/docs/ML1822/ML18228A662.pdf`