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Comparison of Neutronic and Thermal-Hydraulic Performance of KOMODO, NODAL3, and COBRA for Steady-State Operation of APR1400

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Highlights

- Steady State Operation of APR1400 has been done with KOMODO, NODAL3, and COBRA-EN.
- APR-1400 core model is in good agreement between KOMODO and NODAL3.
- The calculated thermal-hydraulics parameters of COBRA-EN show some consistency to KOMODO.
- KOMODO could be a promising tool for analyzing PWR performance.

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Abstract

The important challenge in preparing nuclear reactor safety parameters is in the coupled neutronic and thermohydraulic calculation since the thermohydraulic parameters of the coolant could affect the neutron flux and power distribution. For this reason, it is necessary to carry out accurate calculations for these two parameters. KOMODO is one of the open-source coupled neutronic and thermal-hydraulic (N/TH) calculation codes, while NODAL3 is an in-house coupled N/TH code owned by BRIN. On the other hand, COBRA-EN has also been widely used for steady-state thermohydraulic calculations and could be used to compare the thermal hydraulics solver performance. The APR1400 reactor core was selected under steady-state conditions, either hot zero power (HZP) or critical hot full power (HFP). The calculated core multiplication factor of our HZP model to the reference data is 100 pcm. The radial and axial power distribution calculated by KOMODO and NODAL3 agree with a maximum of 4.78% difference. For steady-state thermal-hydraulics calculations, there was a difference of 1.84% at maximum between KOMODO and COBRA-EN. The KOMODO, NODAL3, and COBRA-EN codes show consistent neutronic and thermohydraulic performance on the APR1400 steady-state calculation model.

1. INTRODUCTION

Indonesia's primary energy comes mostly from coal, natural gas, and petroleum, while industrialization requires sustainable energy sources that must be available in large quantities. Nuclear power plants (NPP) with their high-power density are one possible solution to this problem, which also diversify Indonesia's energy mix. The Nuclear Energy Research Organization of the National Research and Innovation Agency (ORTN-BRIN) as an R&D institution in the nuclear energy sector has carried out several studies on the safety of nuclear power plant reactors while also operating research reactors. One of the NPP candidates that has the potential to be built in Indonesia is the PWR type because it constitutes most nuclear power plants in the world. Several studies have been carried out by ORTN-BRIN to determine the performance of SRAC-2006 and NODAL3 for typical PWR reactors and MOX/UO_2 benchmark cases [1-4]. Other studies consider coupled neutronic and thermal-hydraulic calculations (N/TH) for either steady state or transient calculation regarding the operation of nuclear reactors [5-7].

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An important part of building a nuclear energy community in a nation is to develop tools for analyzing the safety of reactor operations. It could be used to validate the understanding of physics phenomena in nuclear reactors while also improving human resource capacity. ORTN-BRIN developed NODAL3 to evaluate PWR cores' performance such as transient analysis and PWR-FUEL for PWR fuel management. These programs have been used to evaluate the design of several PWR and benchmark cases [8-11]. The programs were developed in early 2000 at the National Nuclear Energy Agency (BATAN) before being merged with other Indonesian research agencies into BRIN.

Neutronic parameters such as core reactivity (criticality) and power distribution either radial or axial became important neutronic parameters in reactor safety operations. On the other hand, thermal-hydraulic analysis of nuclear reactors was used to calculate the core components' temperature, such as the coolant temperature, fuel cladding, and fuel pellet. Analysis of operational safety parameters for nuclear power reactors was needed to predict safety parameters in normal and abnormal conditions. Hence, such computer code must be verified and validated extensively while maintained continuously to improve its performance and reliability. This paper aims to compare the performance of the open source coupled N/TH code KOMODO to our inhouse code NODAL3 while the COBRA-EN code was used to verify the thermal-hydraulic performance of KOMODO by using the power distribution from KOMODO as part of its input. The Advanced Power Reactor - APR1400, a 1400MWe two loops PWR developed by Korea Electric Power Corporation (KEPCO) was chosen as an object to this study as a representation of a modern pressurized water reactor (Gen III+).

2. METHODOLOGY

2.1. APR1400 Initial Core Configuration

The APR1400 initial core consists of 241 fuel assemblies in a rectangular grid. Each fuel assembly consists of 236 fuel rods or fuel rods with burnable absorbers, 4 guide tubes, and 1 center tube, all arranged in a rectangular grid with size of 16×16. Nine types of fuel assembly were used: 1.71 wt% UO_2, 2.00 wt% UO_2, 2.64 wt% UO_2, 3.14 wt% UO_2, 3.64 wt% UO_2, and its variation with gadolinia burnable absorber. The APR-1400 reactor generates a thermal power of 3983 MWth with light water (H_2O) as coolant and moderator [12]. The quarter core configuration and axial fuel region of APR1400 are shown in Figure 1 with the specification of each fuel assembly shown in Table 1.

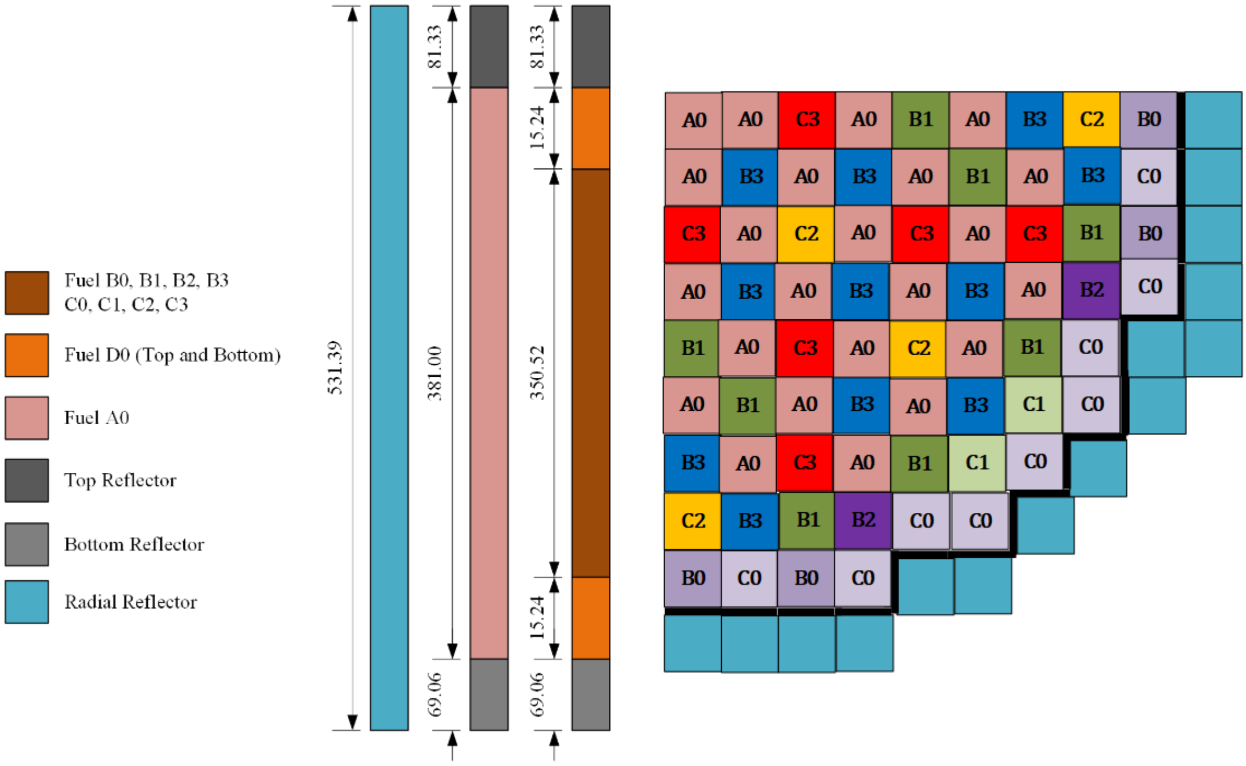

Figure 1. *APR1400 initial core loading pattern and axial fuel zoning*

Table 1. *APR1400 fuel assembly type*

Type	Number of assemblies in core	Enrichment (wt.%)	Number of fuel rods per assembly	Number of Gd_2O_3 fuel rods per assembly	Gd_2O_3 contents (wt.%)
A0	77	1.71	236	-	-
B0	12	3.14	236	-	-
B1	28	3.14/2.64	172/56	12	8
B2	8	3.14/2.64	124/100	12	8
B3	40	3.14/2.64	168/52	16	8
C0	36	3.64/3.14	184/52	-	-
C1	8	3.64/3.14	172/52	12	8
C2	12	3.64/3.14	168/52	16	8
C3	20	3.64/3.14	120/100	16	8
D0	164	2.00	236	-	-

The APR1400 core reactivity is controlled using seven control rod banks, five regular banks labeled 1-5 for power maneuvering, and two shutdown banks labeled as SD (A and B)[13]. The control rod bank configuration is shown in Figure 2.

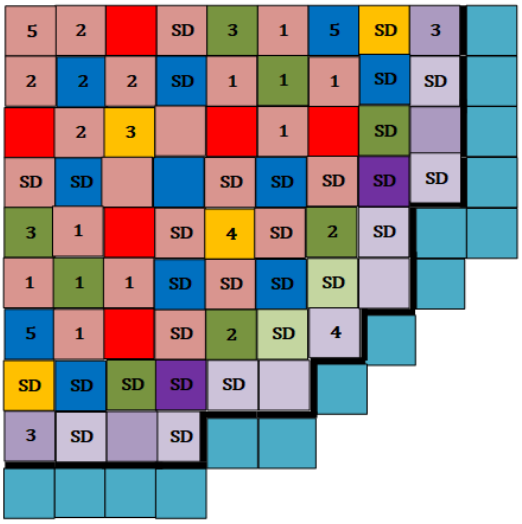

***Figure 2.** APR1400 simplified control bank configuration*

2.2. Neutronic and Thermohydraulics Codes

2.2.1. KOMODO

Komodo is an open-source nuclear reactor calculation program that solve either static or transient calculations using neutron diffusion coupled with simple heat transfer. Komodo solve the diffusion equation using either the nodal method or finite difference method while for transient calculation, the solution in the time domain was solved by discretization using the Crank-Nicholson method [14].

In this study, KOMODO uses the macroscopic cross-section set for APR-1400 fuel assemblies generated using the Serpent program [13]. Semi-Analytic Nodal Method (SANM) approach was used to solve the neutronic calculation of core geometry without the use of the ADF (Assembly Discontinuity Factors) feature. The reactor core was modeled in 3D full core geometry at various conditions such as steady state without boron or critical with a certain boron concentration. Parameters related to heat transfer such as water and steam properties are from the industrial formulation IAPWS-IF97 while thermal conductivity and heat capacity of the UO_2 fuel used are derived from the following correlation [15, 16]

$$\lambda_{UO_2} = 1.05 + 2150/(T[K] - 73.15)\ (W/m.K)$$
$$Cp_{UO_2} = 162.3 + 0.3038\,T[K] - 2.391.10^{-4}T^2 + 6.404.10^{-8}T^3.$$

2.2.2. NODAL3

The NODAL3 program is a coupled neutronic and thermohydraulic program for static and transient parameter analysis of PWR-type reactors. NODAL3 code uses the multigroup neutron diffusion equations with the polynomial nodal method (PNM) to solve 3-dimensional geometry [17]. The thermal-hydraulics module is developed specifically for light water reactor calculations with single-phase coolant flow. Heat conduction equations in fuel rods are discretized in time and space using the finite-difference method. Heat conduction is considered only in radial direction. Fluid dynamics is modelled under single-phase flow conditions. NODAL3 code required a complete set macroscopic cross section for diffusion, scattering, absorption and fission. For determine transient NODAL3 code besides a complete set macroscopic cross section required derivative constant with respect to fuel temperature, moderator temperature, moderator density and boron density.

2.2.3. COBRA-EN

The COBRA-EN code is a tool to analyze nuclear core thermohydraulic by calculating enthalpy distribution, mass flow rate, coolant temperature, fuel meat temperature and cladding temperature, moderator density, heat flux, and departure from nucleate boiling ratio (DNBR). The COBRA-EN program uses the basic concept of subchannel analysis, with the fuel assembly or reactor core divided into several coolant channels with heat generated by its corresponding fuel pin or fuel assembly surface. The thermal-hydraulic model of COBRA-EN is based on partial differential equations corresponding to subchannel approximation and describes the conservation of mass, energy, and momentum either in axial or radial directions for the water liquid/vapor mixture (two-phase coolant) also its interaction with the system structures [18].

The channel is axially divided into several discrete control volumes that solve the equations for the conservation of mass, energy, and momentum. The coolant mass flow rate, pressure, enthalpy, and density were defined as volume-averaged values. The coolant thermophysical properties came from The Electric Power Research Institute (EPRI) correlation.

The COBRA-EN calculated parameter includes the distribution of pressure drops within the coolant channel, temperature distribution for coolant, cladding, and fuel, heat flux, heat conduction, and DNBR. COBRA-EN code has been used to calculate radial and axial power fluctuation of an NPP [19], validate the SIMBAT-PWR core under transient conditions [20], calculate the VVER-1000 reactor [21, 22], and predict AP1000's reactor pressure vessel temperature under normal operation condition [23].

2.3. Neutronic and Thermohydraulıcs Calculation

The neutronic parameter calculations using KOMODO and NODAL3 were done in a 3D core model facilitating the whole fuel assembly shown in Figure 1 with a total height of 381.0 cm. Each fuel assembly zone was divided into 4 nodes (2×2), with the axial zone divided into 27 layers, one with heights of 69.060 cm, then 20 layers of 17.526 cm, 2 layers of 15.24 cm, and 4 layers of 20.332cm respectively from bottom to top. Neutronic calculations were carried out at Hot Zero Power (HZP) conditions with a fuel temperature of 563.75 K, and a moderator temperature of 564.6 K without boron soluble boron (0 ppm). The Hot Full Power (HFP) conditions were then calculated with a fuel temperature of 900 K and a moderator temperature of 582.2 K. The calculation results for criticality are compared with references [12]. Calculation of steady-state parameters such as fuel temperature distribution, moderator, and density were carried out at HFP critical conditions with a boron concentration of 1098.23 ppm. Either KOMODO and COBRA were using same inlet coolant temperature of 563.75 K (15.5 MPa) with fuel assembly mass flow rate of 81.10565 kg/s. The fuel pellet has radius of 4.09575 mm, gap thickness of 9.155E-02 mm, 0.5625 mm of cladding thickness, and 12.3949 mm of fuel pin pitch.

The thermohydraulic calculation of the APR1400 core using the COBRA-EN code was done with the power distribution information from KOMODO. Each APR-1400 fuel assembly had 5 tubes for the control rod guide and instrumentation tube covering 4 fuel rod lattice areas, so in KOMODO, the simplification of the coolant channel for thermal-hydraulics calculation on each fuel assembly was done by using an equivalent of 20 coolant channels without fuel pins as the heat source.

The COBRA-EN code model used was a 1/8 symmetrical core of APR1400 which consists of 38 fuel assemblies, 30 full fuel assemblies, 16 half fuel assemblies, and one-eight for center fuel assembly, to conserve the 241 fuel assemblies in the core. The COBRA-EN 1/8 core model is shown in Figure 3.

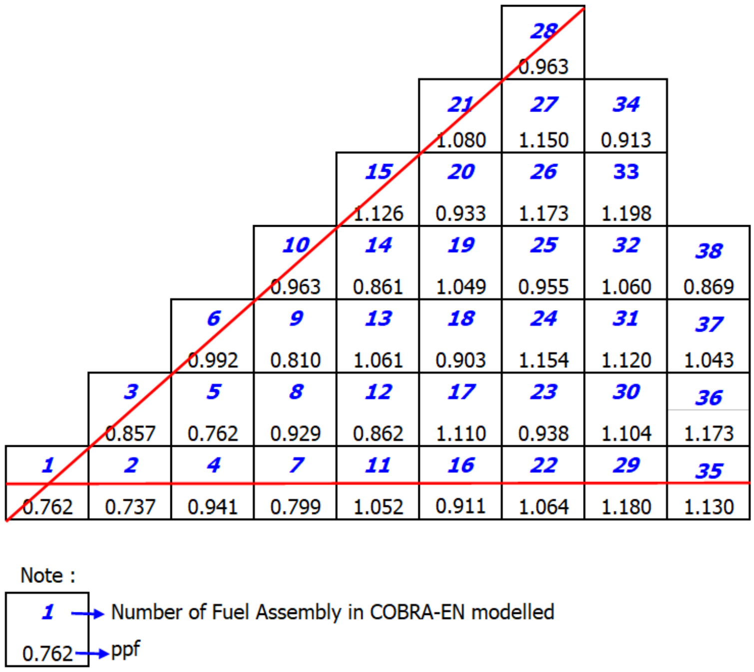

Figure 3. *Normalized Power Density Distribution Near Beginning of Life, Hot Full Power*

There are 2 options for the thermal conductivity of the fuel and cladding materials used in COBRA-EN calculation, one using a fixed thermal conductivity value while the other uses the thermal conductivity as a function of temperature based on the MATPRO fuel analysis package [24]. Since KOMODO uses thermal conductivity values as a function of temperature, then in this calculation, COBRA-EN also uses the same approach. However, the equation for fuel thermal conductivity used by the KOMODO code came from NEACRP 3-D LWR Transient Benchmark [16] while the COBRA-EN use MATRPO [24] so there are some differences as shown in Figure 4.

From Figure 4, the UO_2 temperature of less than 300°C the thermal conductivity used in KOMODO is higher than COBRA-EN, with a maximum difference of 12% at a temperature of 100°C. Meanwhile, at temperatures between 400-1600°C, both codes have less than 3.5% deviation from each other with a maximum difference of 3.2% at 600°C. Above 1700°C, the thermal conductivity used by COBRA-EN from MATPRO is higher than one used by KOMODO.

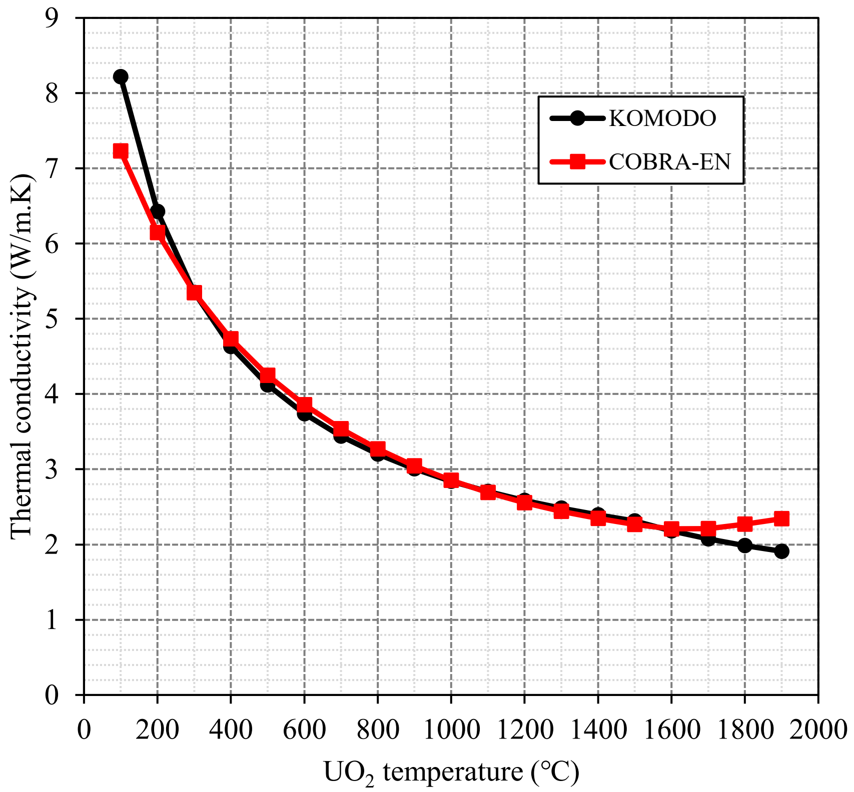

Figure 4. *Thermal conductivity of UO_2 used by KOMODO and COBRA-EN*

3. RESULTS AND DISCUSSION

3.1. Neutronic Parameters

K-eff as the core effective multiplication factor represents the core neutronic capabilities, which in this section will be related to the core excess reactivity at the beginning of the core cycle (BOC). The results of calculated k-eff and design parameters under hot zero power (HZP), boron free (0 ppm) at the BOC are shown in Table 2 which shows a difference of 100 pcm for KOMODO and NODAL3 calculations from design reference.

Table 2. *k-eff value for APR1400 calculated by KOMODO and NODAL3*

Condition	Effective multiplication factors, k-eff		
	Design [12], [13]	KOMODO	NODAL3
Hot zero power (HZP), boron-free (0 ppm) at BOC	1.154	1.153 (-100 pcm)	1.155 (+100 pcm)

Other APR1400 neutronic parameters calculated are core radial and axial power distribution with power fraction defined as the local power density divided by the average core power density. The hottest local power density of a nuclear fuel assembly must be estimated to ensure that the fuel does not melt during normal operation or accident. The radial power distribution calculated under HZP, boron-free conditions at BOC are shown in Figure 5. The maximum relative error was around 0.94% between KOMODO and NODAL3.

1.234	1.111	1.242	1.030	1.234	1.009	0.990	0.945	0.949
1.228	1.104	1.233	1.024	1.229	1.006	0.987	0.946	0.956
-0.486%	-0.630%	-0.725%	-0.583%	-0.405%	-0.297%	-0.303%	0.106%	0.738%
1.111	1.192	1.027	1.141	1.026	1.179	0.906	0.902	0.987
1.104	1.183	1.020	1.133	1.022	1.176	0.905	0.902	0.994
-0.630%	-0.755%	-0.682%	-0.701%	-0.390%	-0.254%	-0.110%	0.000%	0.709%
1.242	1.027	1.228	0.991	1.174	0.960	1.057	0.925	0.894
1.233	1.020	1.220	0.985	1.169	0.958	1.056	0.927	0.902
-0.725%	-0.682%	-0.651%	-0.605%	-0.426%	-0.208%	-0.095%	0.216%	0.895%
1.030	1.141	0.991	1.087	0.946	1.040	0.903	0.907	0.793
1.024	1.133	0.985	1.081	0.943	1.038	0.905	0.912	0.800
-0.583%	-0.701%	-0.605%	-0.552%	-0.317%	-0.192%	0.221%	0.551%	0.883%
1.234	1.026	1.174	0.946	1.122	0.902	1.024	1.064	
1.229	1.022	1.169	0.943	1.120	0.903	1.028	1.074	
-0.405%	-0.390%	-0.426%	-0.317%	-0.178%	0.111%	0.391%	0.940%	
1.009	1.179	0.960	1.040	0.902	0.927	0.939	0.823	
1.006	1.176	0.958	1.038	0.903	0.929	0.945	0.832	
-0.297%	-0.254%	-0.208%	-0.192%	0.111%	0.216%	0.639%	1.094%	
0.990	0.906	1.057	0.903	1.024	0.939	0.852		
0.987	0.905	1.056	0.905	1.028	0.945	0.861		
-0.303%	-0.110%	-0.095%	0.221%	0.391%	0.639%	1.056%		
0.945	0.902	0.925	0.907	1.064	0.823			
0.946	0.902	0.927	0.912	1.074	0.832			
0.106%	0.000%	0.216%	0.551%	0.940%	1.094%			
0.949	0.987	0.894	0.793			KOMODO		
0.956	0.994	0.902	0.800			COBRA-EN		
0.738%	0.709%	0.895%	0.883%			%COBRA EN to KOMODO		

Figure 5. *Radial power distribution under HZP condition, boron-free, at BOC*

Figure 6 shows the average axial power distribution showing the average relative power fraction produced at each axial node of the core under the HZP condition, boron-free, at BOC between KOMODO and NODAL3. The maximum difference between KOMODO and NODAL3 calculations is 4.78%.

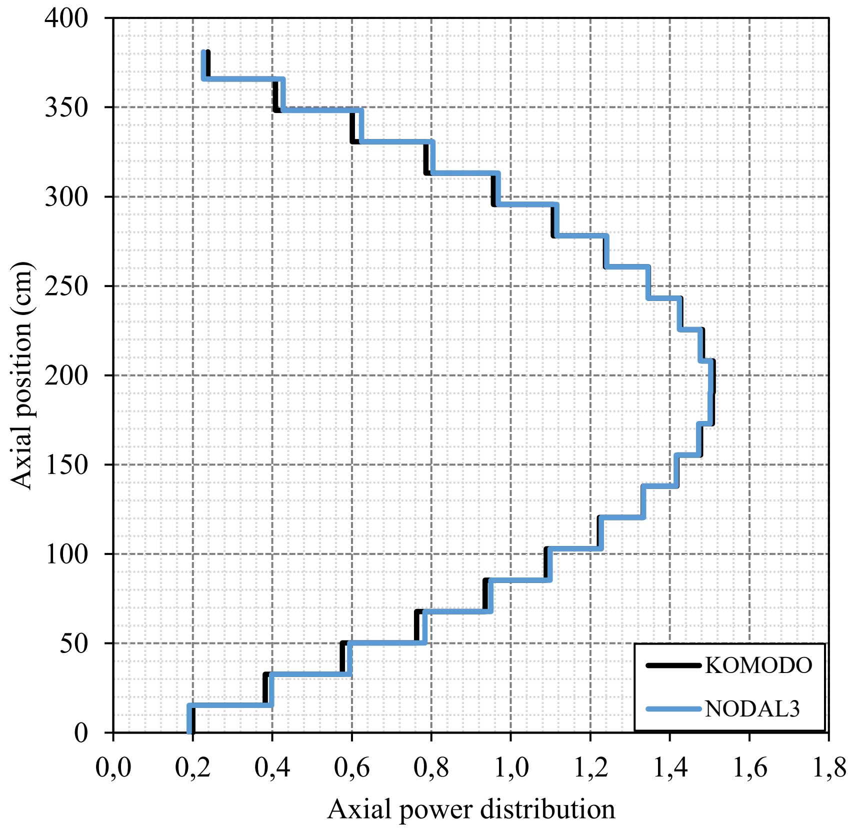

Figure 6. *Axial power distribution under HZP condition, boron-free, at BOC*

3.2. Steady State Thermal Hydraulic Analysis

Thermal-hydraulic analysis for steady-state conditions has been carried out under Hot Full Power (HFP) critical condition with 1098.23 ppm soluble boron, with calculated parameters such as core coolant temperature, peak fuel centerline temperature, and average moderator density. As shown in Table 3, there are no significant differences between KOMODO and COBRA-EN with the maximum difference of 1.84% at the maximum fuel centerline temperature. It can be concluded that the KOMODO and COBRA-EN codes are consistent for calculating thermal-hydraulic parameters under steady-state conditions for APR-1400.

Table 3. *Calculated temperature properties under HFP critical conditions at BOC*

Parameters	KOMODO	COBRA-ER	Deviation
Average fuel temperature (°C)	535.1	528.95	1.15%
Maximum fuel centerline temperature (°C)	1470.8	1443.65	1.84%
Average moderator temperature (°C)	309.5	310.69	-0.38%
Average moderator density (kg/m^3)	703.4	699.75	0.52%
Maximum moderator temperature (°C)	335.7	333.51	0.65%
Outlet moderator temperature (°C)	325.8	326.73	-0.29%
Outlet moderator density (kg/m^3)	663.1	656.74	0.96%

The radial temperature distribution for the fuel, coolant, and coolant density came from each zone's average value. The fuel temperature distribution is shown in Figure 7 while coolant temperature and coolant density distribution are shown in Figures 8 and 9, respectively. Both codes show a good agreement on all parameters, which max out to 4% higher calculated average fuel temperature by COBRA-EN compared to KOMODO. This higher average fuel temperature is also followed by higher coolant temperature (below 0.5%), while coolant density was inverse of the coolant temperature trend.

468.784 486.000 3.673%	462.344 479.440 3.698%	517.078 536.320 3.721%	478.567 497.090 3.871%	548.419 568.960 3.745%	509.001 529.020 3.933%	552.311 573.150 3.773%	586.676 607.360 3.526%	573.180 593.110 3.477%
462.344 479.440 3.698%	494.093 512.580 3.742%	468.957 486.540 3.749%	513.912 533.300 3.773%	495.545 514.820 3.890%	565.265 586.580 3.771%	516.828 536.750 3.855%	564.092 585.360 3.770%	585.960 605.400 3.318%
517.078 536.320 3.721%	468.957 486.540 3.749%	531.318 551.410 3.781%	481.523 500.060 3.850%	551.217 571.910 3.754%	507.101 526.660 3.857%	578.504 599.680 3.660%	569.106 589.940 3.661%	548.028 567.160 3.491%
478.567 497.090 3.871%	513.912 533.300 3.773%	481.523 500.060 3.850%	523.422 543.180 3.775%	495.544 514.600 3.846%	547.761 568.550 3.795%	521.774 541.660 3.811%	551.937 572.150 3.662%	499.948 516.740 3.359%
548.419 568.960 3.745%	495.545 514.820 3.890%	551.217 571.910 3.754%	495.544 514.600 3.846%	570.144 591.270 3.705%	515.371 535.300 3.867%	584.599 605.460 3.568%	593.159 614.810 3.650%	
509.001 529.020 3.933%	565.265 586.580 3.771%	507.101 526.660 3.857%	547.761 568.550 3.795%	515.371 535.300 3.867%	557.142 578.110 3.763%	578.167 598.700 3.551%	512.172 529.430 3.370%	
552.311 573.150 3.773%	516.828 536.750 3.855%	578.504 599.680 3.660%	521.774 541.660 3.811%	584.599 605.460 3.568%	578.167 598.700 3.551%	525.593 543.920 3.487%		
586.676 607.360 3.526%	564.092 585.360 3.770%	569.106 589.940 3.661%	551.937 572.150 3.662%	593.159 614.810 3.650%	512.172 529.430 3.370%			
573.180 593.110 3.477%	585.960 605.400 3.318%	548.028 567.160 3.491%	499.948 516.740 3.359%					

KOMODO
COBRA-EN
%COBRA EN to KOMODO

Figure 7. *Average fuel temperature distribution (^oC) under HFP critical conditions at BOC*

304.834 305.710 0.287%	304.416 305.600 0.389%	308.074 308.460 0.125%	305.634 307.040 0.460%	310.160 310.480 0.103%	307.800 309.120 0.429%	310.673 311.240 0.183%	312.963 313.250 0.092%	312.166 312.780 0.197%
304.416 305.600 0.389%	306.575 307.300 0.237%	304.926 306.190 0.415%	307.943 308.650 0.230%	306.843 308.100 0.410%	311.297 311.760 0.149%	308.450 309.550 0.357%	311.601 312.200 0.192%	312.897 313.250 0.113%
308.074 308.460 0.125%	304.926 306.190 0.415%	309.021 309.560 0.174%	305.868 307.150 0.419%	310.374 310.870 0.160%	307.716 308.950 0.401%	312.284 312.680 0.127%	311.904 312.350 0.143%	310.592 311.090 0.160%
305.634 307.040 0.460%	307.943 308.650 0.230%	305.868 307.150 0.419%	308.627 309.260 0.205%	306.914 308.140 0.399%	310.317 310.850 0.172%	308.819 309.830 0.327%	310.844 311.270 0.137%	307.385 308.170 0.255%
310.160 310.480 0.103%	306.843 308.100 0.410%	310.374 310.870 0.160%	306.914 308.140 0.399%	311.648 312.000 0.113%	308.366 309.470 0.358%	312.783 313.220 0.140%	313.290 313.530 0.077%	
307.800 309.120 0.429%	311.297 311.760 0.149%	307.716 308.950 0.401%	310.317 310.850 0.172%	308.366 309.470 0.358%	311.105 311.670 0.182%	312.474 312.830 0.114%	308.212 308.920 0.230%	
310.673 311.240 0.183%	308.450 309.550 0.357%	312.284 312.680 0.127%	308.819 309.830 0.327%	312.783 313.220 0.140%	312.474 312.830 0.114%	309.104 309.920 0.264%		
312.963 313.250 0.092%	311.601 312.200 0.192%	311.904 312.350 0.143%	310.844 311.270 0.137%	313.290 313.530 0.077%	308.212 308.920 0.230%			
312.166 312.780 0.197%	312.897 313.250 0.113%	310.592 311.090 0.160%	307.385 308.170 0.255%					

KOMODO
COBRA-EN
%COBRA EN to KOMODO

Figure 8. *Average coolant temperature distribution (^{o}C) under HFP critical conditions at BOC*

714.977 713.450 -0.214%	716.000 713.710 -0.320%	707.161 706.200 -0.136%	713.135 709.930 -0.449%	701.908 700.580 -0.189%	707.836 704.290 -0.501%	700.563 698.390 -0.310%	694.488 692.460 -0.292%	696.505 693.810 -0.387%
716.000 713.710 -0.320%	710.911 709.290 -0.228%	714.842 712.170 -0.374%	707.523 705.660 -0.263%	710.247 707.100 -0.443%	698.839 696.900 -0.277%	706.313 703.110 -0.453%	698.146 695.540 -0.373%	694.413 692.450 -0.283%
707.161 706.200 -0.136%	714.842 712.170 -0.374%	704.845 703.160 -0.239%	712.681 709.650 -0.425%	701.339 699.480 -0.265%	708.109 704.760 -0.473%	696.290 694.170 -0.304%	697.373 695.130 -0.322%	700.712 698.780 -0.276%
713.135 709.930 -0.449%	707.523 705.660 -0.263%	712.681 709.650 -0.425%	705.839 703.970 -0.265%	710.109 706.990 -0.439%	701.451 699.510 -0.277%	705.356 702.320 -0.430%	700.215 698.270 -0.278%	708.735 706.900 -0.259%
701.908 700.580 -0.189%	710.247 707.100 -0.443%	701.339 699.480 -0.265%	710.109 706.990 -0.439%	697.951 696.200 -0.251%	706.494 703.330 -0.448%	694.968 692.540 -0.349%	693.416 691.620 -0.259%	
707.836 704.290 -0.501%	698.839 696.900 -0.277%	708.109 704.760 -0.473%	701.451 699.510 -0.277%	706.494 703.330 -0.448%	699.468 697.100 -0.339%	695.758 693.700 -0.296%	706.620 704.870 -0.248%	
700.563 698.390 -0.310%	706.313 703.110 -0.453%	696.290 694.170 -0.304%	705.356 702.320 -0.430%	694.968 692.540 -0.349%	695.758 693.700 -0.296%	704.485 702.060 -0.344%		
694.488 692.460 -0.292%	698.146 695.540 -0.373%	697.373 695.130 -0.322%	700.215 698.270 -0.278%	693.416 691.620 -0.259%	706.620 704.870 -0.248%			
696.505 693.810 -0.387%	694.413 692.450 -0.283%	700.712 698.780 -0.276%	708.735 706.900 -0.259%					

KOMODO
COBRA-EN
%COBRA EN to KOMODO

***Figure 9.** Average coolant density distribution (kg/m^3) under HFP critical conditions at BOC*

As previously mentioned in the Calculation method, the thermal conductivity of UO_2 used by KOMODO and COBRA-EN are below 3.5% and deviate at the range of 300-1500°C, so the deviation on both codes might rooted in the slight differences in heat transfer properties of fuel to cladding and cladding to coolant temperature regarding its position, which will be described later.

The axial distribution of average fuel temperature shown in Figure 10 shows a cosine-shaped profile with peaks that are slightly lower than the middle of fuel height on both codes. This was not caused by control rod insertion since there was no control rod being inserted in our case; soluble boron was used to make the core critical instead. It is caused by coolant flow upwards, so there are cooler coolants with high density at the lower height of the core and the coolant temperature increases on the upper side of the fuel height. The axial distribution of average coolant temperature can be seen in Figure 11 with a slight change in average coolant density seen in Figure 12.

The maximum difference between KOMODO and COBRA-EN on all axial distributions was around -4.79 % and as mentioned before, the fuel temperature calculated by COBRA-EN is higher than that of KOMODO. The maximum differences in average temperature and coolant density between KOMODO and COBRA-EN are 1.76% and 2.23%, respectively.

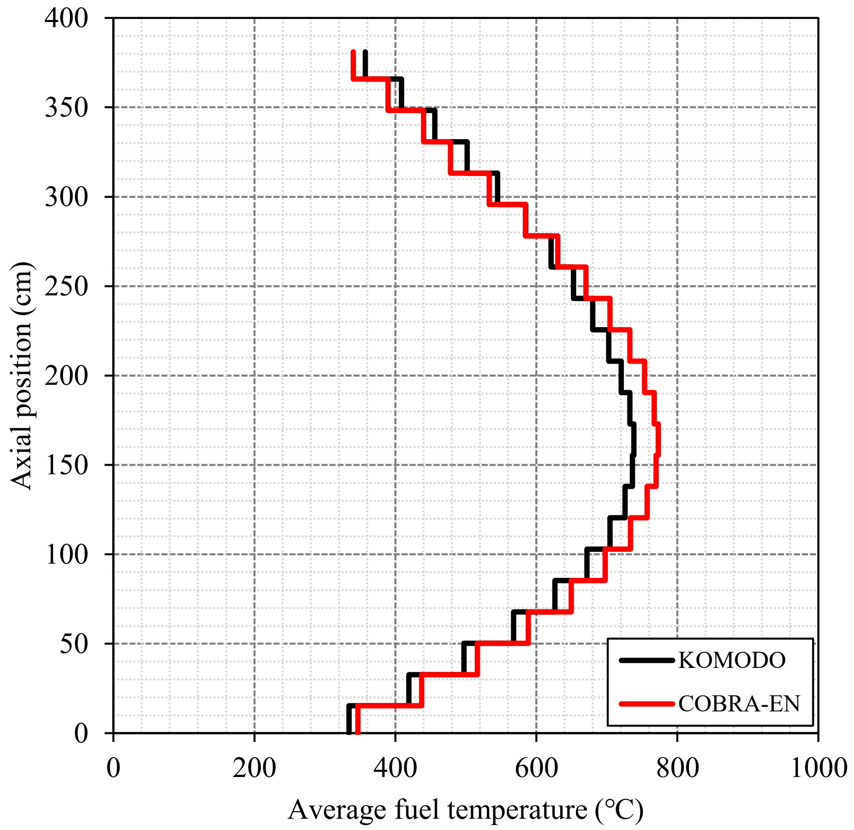

Figure 10. *Axial fuel temperatures distribution*

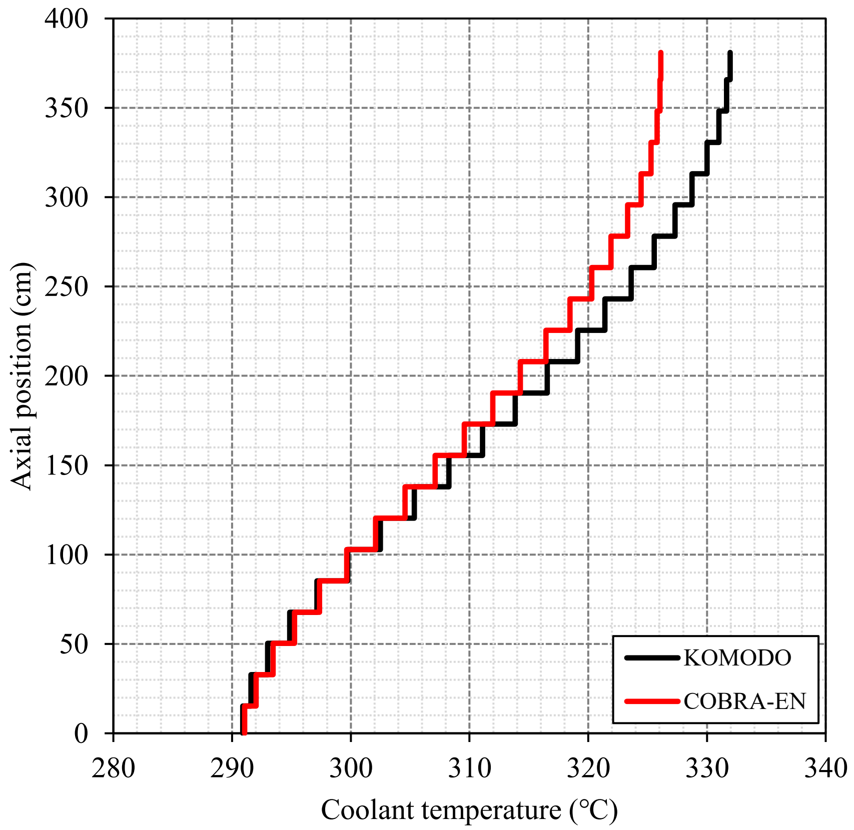

Figure 11. *Axial coolant temperatures distribution*

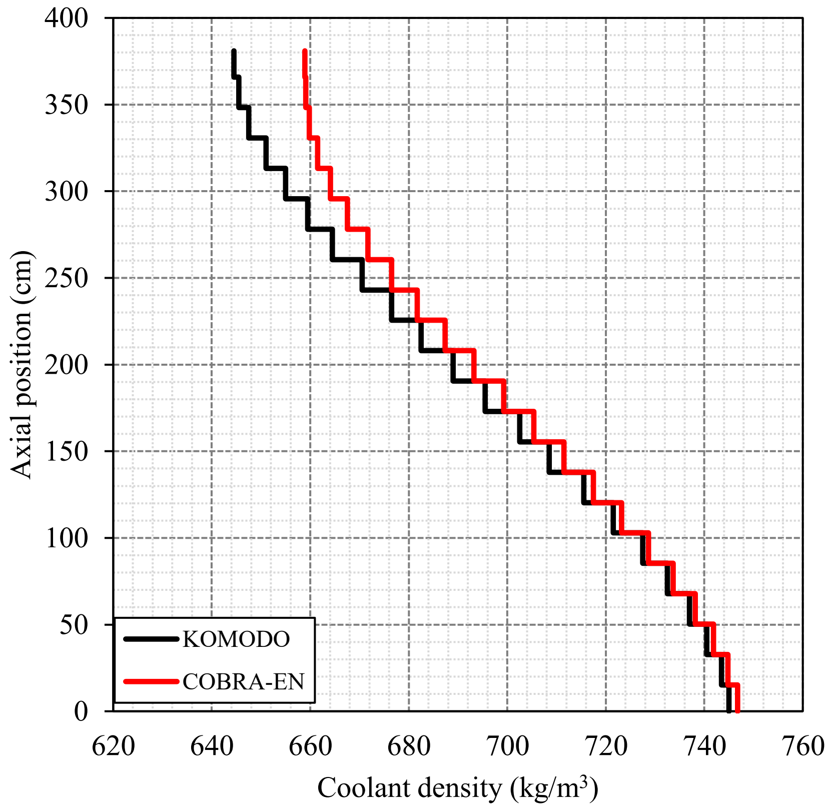

Figure 12. *Axial coolant density distribution*

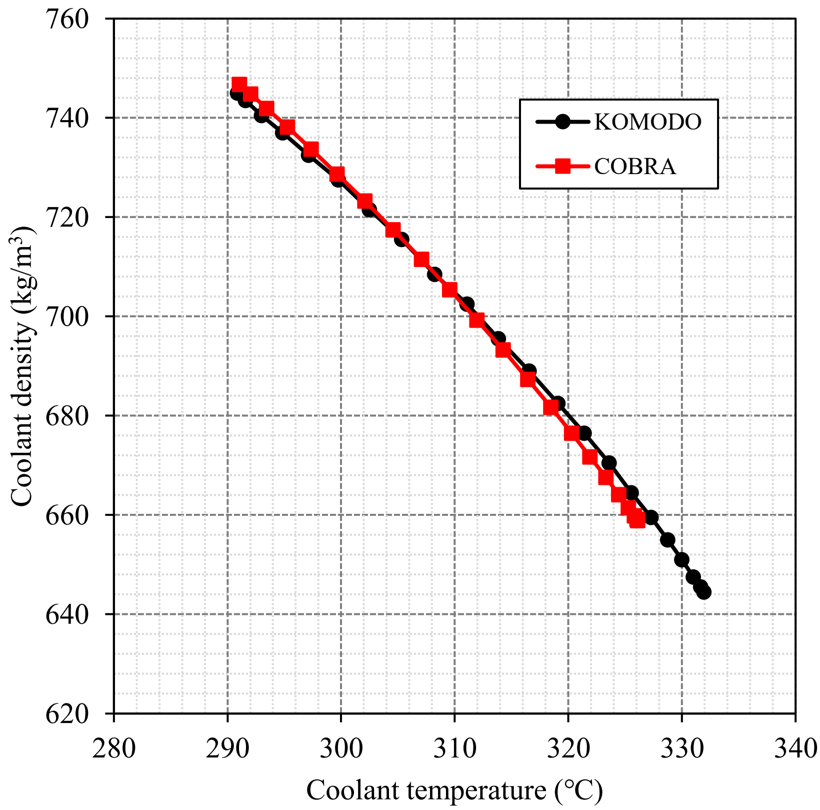

Figure 13. *Coolant density to temperature correlation*

Figure 13 shows the coolant density to temperature correlation from processed axial coolant temperature and coolant density calculated with KOMODO and COBRA. KOMODO used water and steam properties from the industrial formulation IAPWS-IF97 while COBRA used steam correlation based on EPRI. From Figure 13, it can be concluded that both KOMODO and COBRA-EN show some consistency in the water density within a range of 290-320°C besides the difference on the water properties being used. It should be noted that the built-in steam table being implemented in KOMODO has a pressure of 15.5 MPa (155 bar) with a temperature range of 543.15 to 613.15 K, which needs to be updated if other core model has different pressures.

4. CONCLUSION

The neutronic and thermal-hydraulics calculated results of the KOMODO, NODAL3, and COBRA-EN show a good agreement in APR-1400 steady state condition at the beginning of cycle. The KOMODO and NODAL3 as coupled N/TH codes show good agreement to the design parameters under hot zero power conditions. The thermal-hydraulic calculation results between KOMODO and COBRA do not show significant differences in results besides both codes used different approaches to thermal-hydraulics parameters being used such as fuel thermal conductivity and coolant water thermal properties. From this study, the open-source N/TH coupled code KOMODO shows a good agreement with our in-house code NODAL3 while also maintaining some degree of consistency with the industrial standard COBRA-EN. It shows that the KOMODO could be a promising tool for analyzing PWR either for steady-state or transient calculation. Some correction to the thermal hydraulics solver needs to be addressed to model other reactor cores with different pressures or to facilitate another water-cooled reactor such as BWR (Boiling Water Reactor) and SCWR (Supercritical Water Reactor).

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6. CREDIT AUTHORSHIP CONTRIBUTION STATEMENT

Surian Pinem: Conceptualization, Methodology, Software, Validation, Data Curation, Formal Analysis, Writing - Original Draft; **Muhammad Darwis Isnaini**: Software, Formal Analysis, Validation, Data Curation, Writing - Original Draft; **Wahid Luthfi**: Software, Data Curation, Formal Analysis, Writing - Original Draft, Writing - Review & Editing.
All authors have read and agreed to the published version of the manuscript.

CONFLICT OF INTEREST

No conflict of interest was declared by the authors.

Note: Calculations done in this study were performed using KOMODO: an open nuclear reactor simulator and reactor core analysis tool developed by Muhammad Imron https://github.com/imronuke/KOMODO, while NODAL3 is our in-house coupled neutronic and thermal-hydraulics code owned by BRIN, and COBRA-EN was licensed to Muhammad Darwis Isnaini.

REFERENCES

[1] Luthfi, W., Pinem, S., "Validation of SRAC Code System for Neutronic Calculation of the PWR MOX/UO2 Core Benchmark", Urania - Jurnal Ilmu Daur Bahan Bakar Nuklir, 27(1): 47–56, (2021).

[2] Pinem, S., Sembiring, T.M., Surbakti, T., “Neutronic parameters analysis of a PWR fuel element using silicon carbide claddings with SRAC2006/NODAL3 codes”, AIP Conference Proceedings, 2180, (2019).

[3] Luthfi, W., Pinem, S., “Calculation of 2-Dimensional PWR MOX/UO2 Core Benchmark OECD NEA 6048 With SRAC Code”, Jurnal Teknolgi Reaktor Nuklir Tri Dasa Mega, 22(3): 89, (2020). DOI: 10.17146/tdm.2020.22.3.5955

[4] Sembiring, T.M., Susilo, J., Pinem, S., “Evaluation of the AP1000 delayed neutron parameters using MCNP6”, Journal of Physics: Conference Series, 962: 012030, (2018).

[5] Louis, H.K., “Neutronic Analysis of the VVER-1200 under Normal Operating Conditions”, Journal Nuclear Particle Physic, 11(3): 53-66, (2021). DOI: 10.5923/j.jnpp.20211103.01

[6] Zhao, P., Liu, Z., Yu, T., Xie, J., Chen, Z., Shen, C., “Code development on steady-state thermal-hydraulic for small modular natural circulation lead-based fast reactor”, Journal Nuclear Engineering Technology, 52(12): 2789-2802, (2020). DOI: 10.1016/j.net.2020.05.023

[7] Safavi, A., Esteki, M.H., Mirvakili, S.M., Arani, M.K., “Application of a new neutronics/thermal hydraulics coupled code for steady state analysis of light water reactors”, Journal Nuclear Enggineering Technology, 52(8): 1603-1610, (2020). DOI: 10.1016/j.net.2020.01.024

[8] Pinem, S., Sembiring, T.M., Liem, P.H., “NODAL3 Sensitivity Analysis for NEACRP 3D LWR Core Transient Benchmark (PWR)”, Journal of Science Technology Nuclear Installation, 2016: 1-11, (2016). DOI: 10.1155/2016/7538681

[9] Sembiring, T.M., Pinem, S., Liem, P.H., “Analysis of NEA-NSC PWR Uncontrolled Control Rod Withdrawal at Zero Power Benchmark Cases with NODAL3 Code”, Journal Science Technology Nuclear Installation, 2017: 1-8, (2017). DOI: 10.1155/2017/5151890

[10] Sembiring, T.M., Pinem, S., “The validation of the NODAL3 code for static cases of the PWR benchmark core”, Journal Sains Teknologi Nuklir Ganendra, 15(2): 82-92, (2012).

[11] Kuntoro, I., Pinem, S., Sembiring, T.M., “Validation of Pwr-Fuel Code for Static Parameters in the LWR Core Benchmark”, Jurnal Teknologi Reaktor Nuklir Tri Dasa Mega, 20(3): 111, (2018). DOI: 10.17146/tdm.2018.20.3.4650

[12] KEPCO, KHNP, “APR1400 Design Control Document Tier 2: Chapter 4 Reactor: Revision 3”, (2018).

[13] Alnaqbi, J., Hartanto, D., Alnuaimi, R., Imron, M., Gillette, V., “Static and transient analyses of Advanced Power Reactor 1400 (APR1400) initial core using open-source nodal core simulator KOMODO”, Journal Nuclear Engineering Technology, 54(2): 764-769, (2022). DOI: 10.1016/j.net.2021.08.012

[14] Imron, M., “Development and verification of open reactor simulator ADPRES”, Journal Annals Nuclear Energy, 133: 580-588, (2019). DOI: 10.1016/j.anucene.2019.06.049

[15] Wagner, W., Kretzschmar, H.J., International Steam Tables Properties of Water and Steam Based on the Industrial Formulation IAPWS-IF97, Second Edition, Berlin Heidelberg: Springer, (2008). DOI: 10.1007/978-3-540-74234-0

[16] Finnemann H., Galati, A., “NEACRP 3-D LWR Transient Benchmark, Final Specifications”, (1992), [Online]. Available: https://www.oecd-nea.org/jcms/pl_17240

[17] Liem, P.H., Pinem, S., Sembiring, T.M., Tran, H., “Status on development and verification of reactivity initiated accident analysis code for PWR (NODAL3)”, Journal Nuclear Science Technology, 6(1): 1-13, (2016).

[18] NEA, OECD, “COBRA-EN, Thermal-Hydraulic Transient Analysis of Reactor Cores”, OECD NEA code details, (2001). https://www.oecd-nea.org/tools/abstract/detail/nea-1614/. Accessed: 15.01. 2025

[19] Isnaini, M.D., Widodo, S., Subekti, M., “The Thermal-Hydraulics Analysis on Radial and Axial Power Fluctuation for AP1000 Reactor”, Jurnal Teknologi Reaktor Nuklir Tri Dasa Mega, 17(1): 79-86, (2015).

[20] Isnaini, M.D., Subekti, M., “Validation of SIMBAT-PWR using Standard Code of COBRA- EN on Reactor Transient Condition”, Jurnal Teknologi Reaktor Nuklir Tri Dasa Mega, 18(1): 41-50, (2016).

[21] Arshi, S.S., Mirvakili, S.M., Faghihi, F., “Modified COBRA-EN code to investigate thermal-hydraulic analysis of the Iranian VVER-1000 core”, Journal Progress Nuclear Energy, 52(6): 589-595, (2010). DOI: 10.1016/j.pnucene.2010.01.005

[22] Aghaie, M., Zolfaghari, A., Minuchehr, M., Norouzi, A., “Enhancement of COBRA-EN capability for VVER reactors calculations”, Journal Annals Nuclear Energy, 46: 234-243, (2012). DOI: 10.1016/j.anucene.2012.03.026

[23] Isnaini, M.D., Saragi, E., Wardani, V.I.S., “Prediction of Ap1000'S Nuclear Reactor Pressure Vessel Temperature During Normal Operation”, Jurnal Teknologi Reaktor Nuklir Tri Dasa Mega, 24(3): 99, (2022). DOI: 10.17146/tdm.2022.24.3.6684

[24] Todreas, N.E., Kazimi, M.S., Nuclear Systems I: Thermal Hydraulic Fundamentals, Vol. 1, Second Pri. Hemisphere Publishing Corporation: Taylor & Francis, (1990).