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Prediction of AP1000's Nuclear Reactor Pressure Vessel Temperature During Normal Operation

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A B S T R A C T

Modeling of thermal-hydraulic calculations for the AP1000 core to predict the reactor pressure vessel (RPV) temperature has been carried out. The reactor's primary coolant system transfers the heat produced in the reactor fuel during reactor operation to the steam generator. Part of the heat will also be transferred from the coolant to the reactor vessel and the pipe. This paper presents the calculation result of the RPV temperature prediction during AP1000 normal operation. Calculations were performed using COBRA-EN code for analyzing the core thermal hydraulics and using analytics for predicting the RPV temperature. These methods were carried out with the aim to predict the RPV temperature as well as at steady state nominal power conditions, at the function of flow, and at power fluctuation conditions. The calculation results at nominal power 3400 MWt (100% heat generated in fuel was assumed) and thermal design flow with 10% tube plugging (TDF2) of 48,443.7 ton/hr, for the minimum system pressure of 15.1 MPa, nominal system pressure of 15.513 MPa, and design system pressure of 17.133 MPa, show that the core outlet coolant temperature is 326.96°C, 327.01°C, and 327.22°C, and the RPV temperature is 303.65°C, 303.87°C, and 306.67°C, and the minimum departure from nucleate boiling ratio (MDNBR) is 3.21, 3.29, and 3.01, respectively. During reactor operation at a fixed nominal power of 3400 MWt, nominal system pressure, and under the condition of flow fluctuation, the maximum RPV temperature is shown to be 303.87°C.

1. INTRODUCTION

A pressure vessel is an integral part of many manufacturing facilities and processing plants, enabling the safe storage of pressurized liquids or gases. In nuclear power reactors, the reactor pressure vessel (RPV) is one of the most important components. It contains the reactor core and other reactor internals (such as control rods, etc.), and is also an integral part of the reactor coolant pressure boundary. It is designed, fabricated, erected, and tested to a quality standard with certain requirements [1, 2]. The main safety design bases and performance of the reactor vessel are providing a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products.

Most of the nuclear power plants (NPP) operating in the world are light water reactors (pressurized water reactors and boiling water reactors), which are cooled and moderated by high-pressure water. Therefore, the RPV must be able to withstand high pressure. BWR operates at a

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primary system pressure of around 7 MPa, whereas PWR operates at a primary system pressure of around 15.5 MPa. This research will be focused on the RPV of the AP1000 reactor. Due to the operating condition of the AP1000, the RPV is operated under nominal system pressure of 15.513 MPa [1], and the primary coolant remains in a liquid state. In NPP operation, it must always be ensured that the NPP can operate safely in both steady-state and transient conditions. Design and fabrication of the PWR pressure vessels is carried out based on the ASME Code, Section III, Class 1 requirements. The reactor vessel is the primary pressure boundary for the reactor coolant (see Fig. 1). It is also the secondary barrier against the release of radioactive fission products after the fuel matrix, cladding, and reactor coolant. The reactor core, where the fission reaction happened, is placed within the reactor vessel. At operating conditions, the heat generated in the core is transferred by the primary cooling system which is circulated by forced convection to the steam generator. However, some of the heat carried by the primary coolant will be absorbed by the reactor vessel, so the temperature of the reactor vessel will also be considerably high.

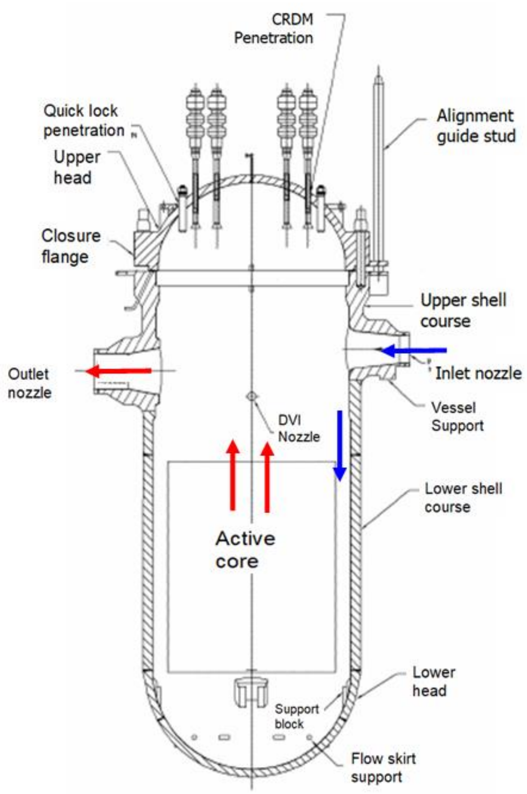

Fig. 1. Reactor pressure vessel of AP1000 [1]

Research on RPV has been conducted such as the design of cylindrical shells of RPV [3,4,5]. These papers recommended that the RPV should be designed accurately to cope with the operating pressure and temperature. A literature review of the design and fabrication of RPV shows that many pressure vessels are made of steel, especially A516 [6]. A study on the conceptual design of RPV for IPR1000 shows that the RPV is designed for static and dynamic loads at the pressure of 2485 psig or 17.133 MPa, and temperature of 343.33°C [7], study on the RPV of PWR shows that the critical points are on the inlet and outlet nozzles area at design condition of 409 MPa [8], and the simulation on the wall of the PWR pressure vessel gives the result that the thermal stress is 248 MPa at inner and outer RPV wall temperature of 427°C and 250°C [9].

Research on thermal hydraulics analysis of an NPP has been conducted by using COBRA-EN code, such as radial and axial power fluctuation [10], validation of SIMBAT-PWR using standard code [11], comparative analysis using fixed thermal conductivity and temperature function of thermal conductivity [12], and capability for VVER reactor calculation [13, 14].

There are two types of core thermal hydraulics analysis, i.e. channel analysis (or core analysis) and sub-channel analysis. Since this research focuses on the prediction of the RPV temperature, only channel analysis is needed. The model of channel analysis in this research refers to the previous model [10].

This work focuses on heat transfer from the core to the coolant, with part of the heat in the coolant transferred to the reactor pressure vessel. The analysis of thermal hydraulics from core to coolant was conducted using COBRA-EN. Another analysis to predict the RPV temperature is conducted by analytical method. The aim of this work is to investigate RPV temperature during AP1000 reactor operation at steady state nominal power conditions, at the function of flow, and at power fluctuation conditions. The result of RPV temperature prediction is useful to evaluate the material strength of the reactor pressure vessel and to avoid material degradation of the RPV.

2. BRIEF DESCRIPTION OF AP1000

The AP1000 core is composed of 157 fuel assemblies, which are arranged into 8 symmetries. The reactor is operated at a nominal power of 3400 MWt, cooled by conservative flow used for thermal-hydraulic analysis (the thermal design flow) of 48,443.7 ton/hr and at a nominal coolant system pressure of 15.513 MPa, respectively. A brief technical specification of AP1000 is presented in Table 1.

Table 1. Technical Specification of AP1000 [1, 10].

Parameter	Value
NPP Type	AP1000
Thermal Power (MWt)	3400
Electric Power (MW)	1117
Effective Flow (ton/hr)	48,443.7
System Pressure (MPa)	15.513
Inlet Coolant Temperature (°C)	279.4
Outlet Coolant Temperature (°C)	324.7
Pressure drop (kPa)	275.0
MDNBR	2.80
Average Heat Flux (kW/m^2)	628.7
Maximum Heat Flux (kW/m^2)	1634.71
Number of Fuel Assemblies	157
Number of fuel rods per F.A.	264
Number of guide thimble per F.A.	25
Total number of fuel rod	41448
Rod pitch (m)	0.0126
Rod Diameter (m)	0.0095
Guide Thimble Diameter (m)	0.01224

AP1000 RPV design parameters are presented in Table 2.

Table 2. Reactor Vessel Design Parameters [1].

Parameter	Value
Design pressure (MPa)	17.133
Design temperature (°C)	343.33
Overall height of vessel and closure head, bottom head outside diameter to top of control rod mechanism (mm)	13944.6 4775.2
Outside diameter of the closure head flange (mm)	3779.8
Inside diameter of flange (mm)	4470.4
Outside diameter of the shell (mm)	4038.6
Inside diameter of the shell (mm)	558.8
Inlet nozzle inside diameter (mm)	787.4
Outlet nozzle inside diameter (mm)	5.6
Clad thickness nominal (mm)	152.4
Lower head thickness (mm)	203.2
Vessel beltline thickness (mm)	158.7
Closure head thickness (mm)	

The coolant flow is supplied by the reactor coolant pumps to remove heat from the reactor core and transfer it to the steam generators. Reactor coolant flow is established by a design procedure in detail, supported by operating plant performance data and component data of experimental hydraulics.

There are four reactor coolant flows that are applied in various plant design considerations, as presented below [1]:

1. Mechanical Design Flow (MDF). The MDF is a conservatively high flow that is used as the basis for the internal mechanical design of fuel assemblies, reactor vessel internals, and other system components. The value of MDF is 104% × BEF.
2. Best Estimate Flow (BEF). The BEF is the flow value that is most likely used for normal full-power operating conditions. This flow is based on the best estimate of the reactor vessel, fuel, steam generator, and pipeline resistance, and on the best estimate of the reactor coolant pump head and flow capability.
3. Minimum Measured Flow (MMF). The MMF is the flow that must be confirmed by the flow measurement obtained during generator startup. This flow is used in the core boiling release (DNB) analysis for the thermal design procedure. The value of MMF is 97% × BEF.
4. Thermal Design Flow (TDF). The TDF is a low conservative value used for thermal-hydraulic analysis where design and measurement uncertainties are not statistically combined. The value of TDF is 95.5% × BEF. Whereas the value of TDF with 10% tube plugging (referred to in this paper as TDF2) is 94.3%.

Table 3 presents the value of four reactor coolant flows in the AP1000 reactor [1].

Table 3. Types of reactor coolant flow in AP1000 [1].

No.	Reactor Coolant Flow	m^3/hr
1	Mechanical Design Flow (MDF)	74406.06
2	Best Estimate Flow (BEF)	71544.28
3	Minimum Measured Flow (MMF)	69397.96
4	Thermal Design Flow (TDF)	68110.16
5	Thermal Design Flow With 10% Tube Plugging (TDF2)	67228.92

3. THEORY

The reactor primary coolant system transfers the heat generated in the reactor fuel to the steam and power conversion system during power operation (normal and transients, including the transition from forced to natural circulation), to maintain fuel condition within the operating limits permitted by the reactor control and protection systems. Part of the heat will also be transferred from the coolant to the reactor vessel and the pipes. The coolant flow from the reactor is generally a strong turbulent flow in order to obtain better heat transfer than natural flow, so the medium and large-scale reactors always use turbulent flow as the coolant. In calculation, the heat transfer coefficient is estimated using an empirical formula obtained from the experiment [1, 15]. The heat transfer coefficient formula used extensively in the thermal design of water-cooled reactors follows the Dittus-

Boelter correlation [15] as shown in Eqs. 1 and 2, when the coolant is heated,

$$Nu = 0{,}023\ \mathrm{Re}^{0{,}8}\ \mathrm{Pr}^{0{,}4} \tag{1}$$

and when the coolant is cooled,

$$Nu = 0{,}023\ \mathrm{Re}^{0{,}8}\ \mathrm{Pr}^{0{,}3} \tag{2}$$

where,

Nu: Nusselt number,

$$Nu = \frac{h\,D_e}{k} \tag{3}$$

Re: Reynold number

$$Re = \frac{\rho\,v\,D_e}{\mu} \tag{4}$$

Pr: Prandtl number,

$$Pr = \frac{\mu c_p}{k} \tag{5}$$

h: heat transfer coefficient [W/m^2 K]
De: equivalent diameter [m]
k: thermal conductivity [W/m K]
ρ: density [kg/m^3]
v: average flow velocity [m/s]
μ: viscosity [kg/m s]
C_p: specific heat at constant pressure [J/kg K]

The heated coolant condition occurs when the coolant receives heat from the fission reaction that occurs in the fuel rod. Meanwhile, the cooled coolant condition has occurred when the heat carried by the coolant is partially transferred to the reactor pressure vessel or to the primary cooling pipe.

Newton's law of cooling description of the heat transfer from a solid to a moving fluid (coolant) is expressed in fundamental relation as:

$$q = hA(t_w - t_b) \tag{6}$$

Where:
q : heat [W]
h: heat transfer coefficient [W/m^2 K]
A: flow area [m^2]
T_w: outer temperature of a wall (cladding) [K]
T_b: temperature of coolant [K]

4. METHODOLOGY

In the core thermal hydraulics analysis using COBRA-EN code, the core used 1/8 of the core consisting of 26 fuel assemblies, as shown in Fig. 2.

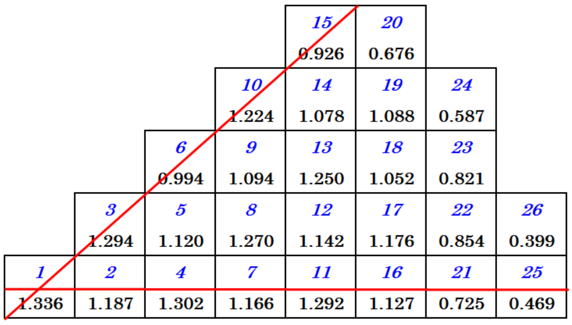

Fig. 2. Normalized Power Density Distribution Near Beginning of Life, Hot Full Power, Equilibrium Xenon [1, 10].

In predicting RPV temperature, it was assumed that the heat is only produced from the fuel and no heat is generated in the RPV, but the RPV absorbs the heat carried by the coolant. Hence, the maximum temperature of RPV is predicted to occur around the upper part of the core i.e. at the coolant outlet from the core. The core coolant temperature could be obtained from core thermal hydraulics analysis or channel analysis that was conducted using the COBRA-EN code. However, there is insufficient data on the material specification and its geometry on the part above the core, so the prediction of RPV temperature was carried out at the vessel outlet or at the hot leg, by referring temperature difference between the core outlet and the vessel outlet about 2.22°C [1]. Predicting the RPV temperature was performed through several steps as explained below.

1. Core thermal hydraulics analysis

The calculation model of 1/8 of AP1000s core, referring to the previous model [10], was conducted using COBRA-EN code for nominal reactor power of 3400 MWt (assumed 100% heat generated in fuel) and thermal design flow with 10% tube plugging (TDF2) of 48,443.7 ton/hr, for the minimum system pressure of 15.1 MPa, nominal system pressure of 15.513 MPa, and design system pressure of 17.133 MPa. The core calculation outputs are the distribution of pressure drop, enthalpy, coolant temperature, coolant density, void fraction, and flow of the core. Whereas, the channel calculation outputs are the distribution of pressure drop, enthalpy, coolant temperature, coolant density, void fraction and flow, critical heat flux, momentum, heat flux, a departure from nucleate boiling ratio (DNBR), temperatures of average fuel, center meat, and outer meat, temperatures of inner and outer cladding, heat transfer coefficient, heat transfer mode, and channel coolant temperature.

2. Prediction of the RPV temperature for minimum, nominal and design system pressure

From the core output temperature (step 1) for each system pressure, the water properties at each temperature and system pressure, such as density, specific heat, viscosity, and thermal conductivity, the analytical calculation was conducted using Eqs. 4, 5, 2, 3, and 6 in sequence. Then, the RPV temperature will be obtained.

3. Prediction of the RPV temperature as a function of flow fluctuation

The flow fluctuation is generally used at the hydraulic test (function test of the pump) or flow adjustment at cold conditions (reactor shutdown), to understand the influence when the flow decreased from MDF to TDF, or on the contrary, increased from TDF to MDF. At the moment of the hydraulic test, it was occasionally found that the cavity in the system is too high when the reactor coolant system uses MDF or BEF, so flow adjustment was needed. In this work, the calculation was assumed on reactor operation conditions while flow adjustment is in process of decreasing flow from MDF to TDF. The MDF is the highest flow used as a criterion value so the flow fraction of BEF, MMF, TDF, and TDF2 against the MDF is 96.16%, 93.47%, 91.73%, and 90.64%, respectively. The model of the core thermal hydraulics calculation under the condition of flow fluctuation was carried out at a nominal power condition of 3400 MWt and primary system operating pressure of 15.513 MPa.

Steps no (1) and (2) were also used to predict the RPV temperature as a function of flow fluctuation. However, the model of core thermal hydraulics must be changed using the time step as shown in Table 4 for flow fluctuation.

Table 4. Time step for flow fluctuation

Time (s)	Flow (%)	Flow Type
0	100.00	MDF
21600	100.00	MDF
23400	96.16	BEF
37800	96.16	BEF
39600	93.47	MMF
54000	93.47	MMF
55800	91.73	TDF1
70200	91.73	TDF1
72000	90.64	TDF2
86400	90.64	TDF2

4. Prediction of the RPV temperature as a function of power fluctuation

The power fluctuation was assumed on the prediction of daily energy demand in a future developed city in Indonesia. Until midnight (at 00.00 a.m.), energy demand is still high, so the reactor is operated at full power (100%). After midnight, energy demand is decreased, and the reactor is operated at 90% of full power. At 5.00 a.m., energy demand is increased again to full power. At 8.30 a.m., the industrial and office activities began, and the energy demand increased to 110% full power (overpower) for 12 hours. At 9.00 p.m., part of the office activities ends and is changed to at-home activities, so the energy demand is decreased to 100% full power until midnight. The time required for increasing or decreasing the reactor power is 30 minutes. The model of the core thermal hydraulics calculation as a function of power fluctuation conditions is carried out at the thermal design flow conditions with 10% tube plugging (TDF2) of 48,443.7 tons/hr.

Steps no (1) and (2) were also used to predict the RPV temperature as a function of power fluctuation. However, the model of core thermal hydraulics must be changed using the time step as shown in Table 5 for power fluctuation.

Table 5. Time step for power fluctuation.

Time (s)	Power (%)
0	100
1800	90
16200	90
18000	100
28800	100
30600	110
73800	110
75600	100
86400	100

4. RESULTS AND DISCUSSION

Prediction of RPV temperature at nominal power for minimum, nominal and design system pressures

The results of thermal hydraulics and RPV temperature prediction of the AP1000 reactor core are shown in Table 6. The result was obtained from the conditions of the nominal power of 3400 MW, TDF2 of 48,443.7 ton/hr, and a coolant inlet temperature of 279.44°C, each for the minimum reactor coolant system pressure of 15.10 MPa, nominal reactor coolant system pressure of 15.513 MPa, and design reactor coolant system pressure of 17.133 MPa.

Table 6. Comparison of calculation results of thermal-hydraulics and RPV temperature based on minimum, nominal, and design system pressures in AP1000 reactor

	System Pressure [MPa]		
	15.10	15.513	17.133
Core Temp [°C] :			
▪ inlet coolant	279.44	279.44	279.44
▪ outlet coolant	326.96	327.01	327.22
▪ average rise	47.52	47.57	47.78
Pressure drop [kPa]	277.56	277.27	276.17
Hot channel Max. Temp [°C]:			
▪ average meat	952.55	952.75	952.85
▪ center meat	1358.25	1358.35	1358.55
▪ outer meat	514.65	515.15	517.05
▪ inner clad	385.25	387.05	393.25
▪ clad surface	346.05	347.95	355.15
▪ outlet coolant	340.01	340.07	340.31
Heat transfer coeff. at the core exit, h [W/m^2 K]	67205.08	49568.91	37852.22
MDNBR [-]	3.21	3.29	3.01
Axial position of MDNBR [m]	3.124	1.753	1.753
Prediction RPV Temp [°C]	303.65	303.87	306.67

Table 6 presents the comparison of thermal-hydraulics analysis and prediction of maximum RPV temperature for minimum, nominal, and design reactor coolant system pressures. The core outlet coolant temperatures for minimum, nominal and design system pressures are 326.96°C, 327.01°C, and 327.22°C, respectively. Whereas the hottest channel outlet coolant temperatures are 340.01°C, 340.07°C, and 340.31°C, respectively. It is shown that at the core coolant temperatures and hot channel temperatures in all 3 cases are almost unchanged against the difference in reactor coolant system pressure. However, at the minimum, nominal and design reactor coolant system pressures for outer cladding surface temperatures were obtained at 346.05°C, 347.95°C, and 355.15°C, and for inner cladding surface temperatures were 385.25°C, 387.05°C, and 393.25°C, respectively. The difference in clad temperatures is caused by the difference in saturation temperature at the three reactor coolant system pressures. At minimum system pressure of 15.10 MPa, the coolant saturation temperature is 343.36°C. Then, at the nominal reactor coolant system pressure of 15.513 MPa, the coolant saturation temperature is 345.60°C. Meanwhile, at the design reactor coolant system pressure of 17.133 MPa, the coolant saturation temperature is 354.04°C. Different saturation temperatures affect water properties, for example, water density, viscosity, and thermal conductivity will be smaller. Thereby, the Reynolds (Eq. 4), Prandtl (Eq. 5), and Nusselt numbers (Eq. 1) will change into smaller numbers, so that the heat transfer coefficient *h* (Eq. 5) at the exit hot channel becomes smaller as well, i.e., 67205.08 W/m^2.K, 49568.91 W/m^2.K, and 37852.22 W/m^2.K, respectively.

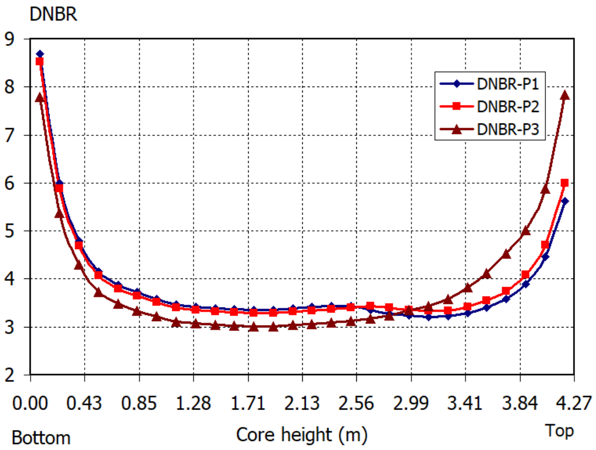

Fig. 3. Distribution of DNBR along the height of the hot channel of the AP1000 reactor.

Fig. 3 presents the distribution of DNBR along the height of the hot channel for minimum (P1), nominal (P2), and design (P3) system pressure conditions. The minimum DNBR (MDNBR) also responds to the change in reactor coolant system pressure. The graphs of P1 and P2 have the same trends, i.e. lower at the position near the bottom and near the top, whereas the graph of P3 is lower at the position near the bottom. The MDNBRs for minimum, nominal, and design reactor coolant system pressures are 3.21 (at z = 3.124 m), 3.29 (at z = 1.753), and 3.01 (at z = 1.753 m), respectively. The MDNBR tends to decrease at the same position for nominal system pressure and design system pressure. At z = 1.753 m, the MDNBR is 3.38 for minimum system pressure. The MDNBR for the condition of the limit design pressure of 17.133 MPa (P3) is lower than the MDNBR for minimum and nominal system pressure. However, all DNBR in these cases is higher than the MDNBR of the AP1000 design of 2.80.

The core outlet coolant temperature obtained from analysis using COBRA-EN will be used to predict the temperature of the AP1000 reactor pressure vessel analytically. Using Eqs. 2, 3, and 6, it was predicted that the RPV temperature for minimum, nominal, and design system pressure is

303.65°C, 303.87°C, and 306.67°C, respectively. Compared to the minimum system pressure and design system pressure conditions, the nominal system pressure condition has moderate meat, clad, coolant, and RPV temperatures, apart from a better safety margin in terms of DNBR.

For the next step, the core outlet coolant temperature is used to predict the RPV temperature at the vessel outlet for flow fluctuation and power fluctuation conditions. Temperature calculation was carried out only for a nominal system pressure of 15.513 MPa.

Prediction of RPV temperature under flow fluctuation condition

In this step, core thermal hydraulics calculation was carried out as a function of flow fluctuation. Calculations were carried out based on reactor power of 3400 MWt (100% heat generated in fuel), nominal system pressure of 15.513 MPa, and coolant inlet temperature of 279.44°C.

Fig. 4 presents the graph of core outlet and RPV temperatures as a function of flow fluctuation. The core outlet temperature as a function of flow fluctuation from MDF, BEF, MMF, TDF, and TDF2 is 323.09°C, 324.62°C, 325.83°C, 326.46°C, and 327.01°C, respectively. Meanwhile, the prediction of RPV temperature is 299.94°C, 301.47°C, 302.68°C, 303.32°C, and 303.87°C, respectively.

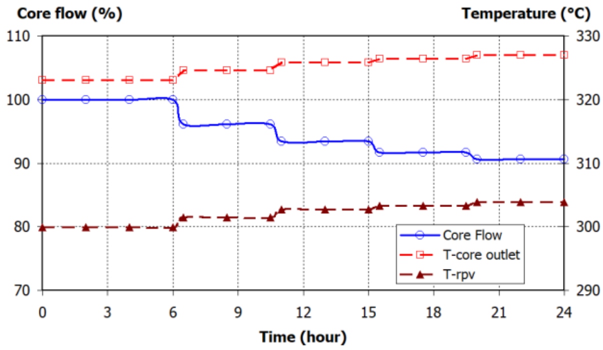

Fig. 4. Graph of the core outlet coolant temperature and prediction of RPV temperature as a function of the flow fluctuation in the AP1000 reactor.

From this result, it is obtained that the maximum RPV temperature under flow fluctuation conditions and fixed nominal power of 3400 MWt is 303.87°C. It was obtained when the reactor is cooled by TDF2.

Prediction of RPV temperature under reactor power fluctuation condition

In this step, core thermal hydraulics calculation was carried out as a function of power fluctuation. Calculations were carried out using TDF2 of 48,443.7 ton/hr at a nominal system pressure of 15.513 MPa and coolant inlet temperature of 279.44°C.

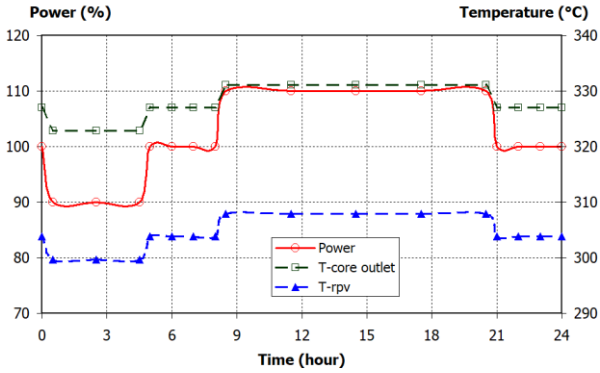

Fig. 5. Graph of core outlet coolant temperature and prediction of RPV temperature as a function of the power fluctuation in AP1000 reactor.

Fig. 5 presents the graph of core outlet and RPV temperatures as a function of reactor power fluctuation. Core outlet temperature as a function of reactor power fluctuation from 100%, 90%, 100%, 110%, and 100% are 327.01°C, 322.81°C, 327.01°C, 331.09°C, and 327.01°C, respectively. Meanwhile, the RPV temperature prediction is 303.87°C, 299.66°C, 303.87°C, 307.94°C, and 303.87°C, respectively.

From the result, it is obtained that the maximum RPV temperature as a function of reactor power fluctuation at a fixed flow of TDF2 is 307.94°C. This temperature is obtained at a condition of 110% reactor power or overpowered condition.

5. CONCLUSION

Nominal system pressure condition has moderate meat, clad, coolant, and RPV temperatures, nonetheless it possesses a better safety margin of DNBR compared to minimum system pressure and design system pressure. The maximum RPV temperature under the condition of flow fluctuation, at a fixed nominal power of 3400 MWt is 303.87°C. This temperature is obtained at the TDF2 condition. Furthermore, for reactor power fluctuation and at the fixed flow of TDF2, it is shown that the maximum RPV temperature is 307.94°C. This temperature is obtained at the condition of 110% reactor power or overpowered condition.

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AUTHOR CONTRIBUTION

Muhammad Darwis Isnaini carried out core modeling in core thermal hydraulics calculation using COBRA-EN code, Elfrida Saragi and Veronica Indriati Sri Wardhani carried out the prediction of RPV temperature using analytical methods. Muhammad Darwis Isnaini, Elfrida Saragi and Veronica Indriati Sri Wardhani participated together as reviewers and data analysis. Muhammad Darwis Isnaini, Elfrida Saragi and Veronica Iindriati Sri Wardhani equally contributed as the main contributors to this paper. All authors read and approved the final version of the manuscript.

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