The International Journal of Engineering and Science (IJES)
|| Volume || 7 || Issue || 5 Ver. I || Pages || PP 49-56 || 2018 ||
ISSN (e): 2319 – 1813 ISSN (p): 23-19 – 1805

Characterization Of Oxide Fuel Element Temperature Of RSG GAS By Using Forced And Natural Convection Cooling Mode

Sudarmono, Endiah Puji Hastuti

Center for Nuclear Reactor Technology and Safety, Bld 80 PUSPIPTEK Area, Serpong, South Tangerang, 15310, Indonesia

Center for Nuclear Reactor Technology and Safety, Bld 80 PUSPIPTEK Area, Serpong, South Tangerang, 15310, Indonesia

Corresponding Author: aksel@batan.go.id

ABSTRACT

The RSG-GAS reactor is a pool type research reactor which is operated at a power level of 30 MW using LEU MTR-type fuel. RSG-GAS uses both of forced convection cooling mode and natural convection mode. An assessment of the steady state and natural convection thermal hydraulic capabilities of the computer HEATHYDE and NATCON codes was made using model for a RSG GAS reactor core. The measurement results of temperature distributions for fuel, clad and coolant core thermo-hydraulics are compared to the results of HEATHYDE and NATCON codes at steady state and natural convection condition. The heat transfer characteristics of down flow in the single-phase forced-convection will be investigated by Temperature measurement is done using two instrumented fuel elements equipped with thermocouple. The assessment has included both steady-state and natural convection analyses of safety margins and limits. The measurement results of core thermo-hydraulics at steady state, are compared to the results of these codes. The maximum fuel temperature obtained from direct measurement was 121 ℃ at steady state of the working core operating at 30 MW. The deviation between measurement results and calculation of each core is less than 13.14%. Under normal operation, core safety margins towards DNB and OFI were 3.56 and 2.6, respectively. The results obtained by NATCON for typical core cooled by natural convection mode indicate that onset of nucleate boiling occurs at power level of 826 kW (2.8% of nominal power)..

KEYWORDS *- oxide fuel element, instrumented fuel element, RSG GAS, HEATHYDE, NATCON.*

Date of Submission: 28-04-2018 Date of acceptance: 14-05-2018

I INTRODUCTION

The G.A. Siwabessy Multi Purpose Nuclear Reactor (RSG-GAS) which is built in the area of the Center for Research of Science and Technology (PUSPIPTEK) Serpong is one of the facilities possessed by BATAN. RSG-GAS is managed and operated by the Center of the Multi Purpose Reactor (PRSG). The RSG-GAS reactor is a pool type research reactor which is operated at a power level of 30 MW using LEU MTR-type fuel. The operation cycle length is 25 full power days (or 750 MWD). The reactor core is placed at the bottom of the reactor pool, around 12.5 m below the surface of the pool water. This core consists of 40 *Fuel Elements (FE), 8 Control Elements (CE)*, on large central irradiation position (CIP) containing 2x2 grid plate position, and 4 small in-core in irradiation position (IP) and several irradiation holes and is surrounded by beryllium elements and blocks as a reflector. The other two sides of the active core cross section are surrounded by a beryllium block reflector. Demineralized light water is used as moderator and coolant, see Fig 1.

K	BS+59	B-29	B-30	PRTF	B-20	B-13	B-8	BS+10	B-5	B-2
J	B-28	BS+58	B-22	PRTF	B-21	B-23	B-24	B-4	BS+52	B-15
H	B-26	**FE**	**FE**	FE	FE	FE	FE	B-19	B-17	BS+51
G	B-16	FE	**FE**	AL-4	CR	FE	FE	B-40	BS+57	B-14
F	**FE**	FE	CR	FE	FE	CR	FE	**FE**	B-32	PNRA
E	FE	CR	FE	AL-6	AL-3	FE	AL-8	FE	B-34	HYRA
D	FE	AL-2	FE	AL-5	AL-7	FE	CR	**FE**	B-36	HYRA
C	**FE**	FE	CR	FE	FE	CR	**FE**	**FE**	B-37	HYRA
B	BS+54 NS	FE	FE	CR	AL-1	**FE**	FE	B-06	B-11	HYRA
A	B-10	**FE**	FE	FE	FE	FE	FE	B-03	BS+56	B-1
	10	9	8	7	6	5	4	3	2	1

BERYLLIUM BLOCK REFLEKTOR

Note:
B = Beryllium, BS+ = Beryllium stopper with stopper, Al = Aluminum stopper without stopper, RI = Fuel Element, CR = Control Element, NS = neutron source

Fig. 1. RSG-GAS Core Configuration.

RSG-GAS uses forced convection cooling mode through the primary cooling pump, under normal operation. The pump is completed with fly wheel so that the primary coolant flow rate does not drop immediately when the pump is failed and the reactor power is still sufficiently high. For particular application such as foil irradiation at fuel element position, reactor operation with maximum power of 1% of nominal power is allowed without the primary pump in operation[1]. Reactor core cooling by natural convection occurs when primary coolant flow rate decreases since the pump is failed so that natural circulation valve is open. Natural convection cooling mode also might occur because the primary and secondary cooling system do not operate due to low power generated in the reactor core. The main purpose of reactor safety is not to permit any release of radioactivity from the reactor core to the environment. To ensure reactor safety, the reactor must have sufficient margins during normal operation as well as in all possible accidental events, such force convection, natural convection and loss-of-flow accidents (LOFAs), etc.

Many studies have been performed regarding research reactor analysis by researchers. Ahmad Lashkari (2015) performs LOFA experimental and numerical studies in Tehran Research Reactor (TRR). The main purpose of this study is the use of simple calculation methods and benchmarks it with experimental data [2]. While Moh. Faiz Salim et al. (2016) presenting part of the overall development nodalization description for RTP-TRIGA research reactor [3]. Badrun Nahar Hamid (2015) study the LOFA analysis of the research reactor, the aim is to examine the impact of LOFA on the safety of the reactor. The analysis is to ensure sufficient margin during normal operation as well as in all possible accident events [4]. Pavel Zitek (2014) summarizes the basis for the solution heat removal by natural convection of conventional nuclear reactors and reactors with coolant flow through the fuel [5]. Deepak Sharma (2015) analyzed the determination of the optimal coolant flow distribution and pressure drop across the core [6].

Safety analysis of many various reactors also performed, using several computer codes. A. Hussain (2015) perform an assessment of the ability of a steady state and transient thermohidraulik of computer programs COBRA 3C / RERTR made using the model for PHWR reactor core [7]. Meanwhile F. M. Bsebsu (2000) investigated the characteristic of heat transfer with downward flow in single-phase conventional-force convection, as theoretically with narrow annular vertical sub-canals (WWR-M2 channel), using THMOD2 program [8]. W. L. Woodruff et al. (1994), are developing a computer code to analyze natural convection characteristic of research reactor, namely NATCON [9]. Beside that they also performed Transient analyses and Thermal-hydraulic Safety Margins for The Greek Research Reactor (GRRI). The assessment has included both steady-state and transient analyses of safety margins and limits [10]. NABBI, R. (1989) performed Steady-State Thermal Hydraulic Analysis of Low-Enriched U Fuel Reactor using HEATHYD code (11). In all previous studies, the results of this model are rarely compared to the experimental values and are usually compared with computational results with other programs. Therefore this paper explained the comparison between calculation method and measurement using instrumented fuel element, for steady state and natural convection.

An assessment of the steady state and natural convection thermal hydraulic capabilities of the computer HEATHYDE and NATCON codes was made using model for a RSG_GAS reactor core. The measurement results of temperature distributions for fuel, clad and coolant core thermo-hydraulics are compared to the results of HEATHYDE and NATCON codes at steady state and natural convection condition[12]. The heat transfer characteristics of down flow in the single-phase forced-convection will be investigated by temperature measurement, was done by using two instrumented fuel elements equipped with thermocouple. Evaluation on measurement results is carried out by computation codes such as HEATHYDE, NATCON and the results are then compared to criteria of safety margins. *This study focuses on t*emperature fuel element distribution measurement, clad material, on-set of critical heat flux phenomenon analysis of research reactors with the aim of investigating the impact of an unprotected on reactor safety. Major parameters such as fuel cladding temperature and bulk coolant temperature are evaluated to investigate whether these parameters exceed their safety limit or if any nucleate boiling occurs in the bulk coolant. [13]

II METHODOLOGY

2.1. HEATHYDE Code

HEATHYDE is a code developed to analyze thermo-hydraulics of MTR-type reactor core in steady state. HEATHYDE computing code employs a model of heat transfer calculation and one-dimensional cooling rate for fuel element with axial and radial power distribution. HEATHYD is a code for the steady-state heat transfer calculation of research nuclear reactors with forced convection. It models heat transfer and coolant flow for assemblies of parallel fuel plates of MTR type with any axial power distribution The thermodynamic model accounts for single phase cooling and sub- cooled boiling condition using the transition criterion of Bergeles-Rohsenow. In addition to the calculation of the channel flow velocities and coolant pressure drops, HEATHYD calculates axial distribution of the coolant and clad-surface temperatures. Applying the finite difference method, HEATHYD solves the equations of heat conduction and heat transfer to the coolant. For the physical properties of the coolant as a function of the coolant temperature polynomials of degree 6 are used. Depending on the coolant condition, different correlations for the heat transfer coefficient can be applied. The analysis of the critical cooling conditions resulting in burnout or flow instability, is performed according to the correlations developed by Mirshak/ Labuntsov and Forgan/Whittle. [14]

This code is completed with equations to compute meat temperature, plate temperature, coolant temperature, pressure drop, coolant flow rate, heat flux, and safety margin towards ONB, OFIR, and DNBR.

The heat flux at ONB is given by Bergles-Rohsenow correlation:

$$q_{ONB} = P_z^{1.156}/9.23\,[1.8(T_s - T_{sat})]^{\left(0.463\,P_z^{0.0234}\right)^{-1}} \quad (1)$$

The safety limits for critical heat conditions causing departure from nucleate boiling (DNB) are determined iteratively using Mirshak and Labunsov correlations [15]. Mirshak correlation has a range of parameters of coolant velocity, subcoolant temperature and critical hot flux more limited than Labunsov correlation. According to Mirshak critical heatflux is given by:

$$q_c = 151(1 + 0.119U)(1 + 0.00914\Delta T_{sub})(1 + 0.019P) \quad (2)$$

where: $\Delta T_{sub} = T_{sat} - T_{in} - T_c$

while the *Labunsov* correlation is :

$$q_c = 145.4\,\theta(P)[1 + 2.5U^2/\theta(P)]^{1/4}\left(1 + 15.1C_p.\Delta T_{sub}/\lambda P^{1/2}\right) \quad (3)$$

where :

$$\theta(P) = 0.99531P^{1/3}\left(1 - \left({}^{P}/_{P_C}\right)\right)^{4/3}$$

Bubble release parameters can be derived from experiments performed on various test sections on narrow channels, ie,:

$$\eta(z) = \left[Ts_{(z)} - Tc_{(z)}\right]v_{(z)}/q"_{(z)} \quad (4)$$

The parameter η controls when the bubbles remain attached to the wall or not. While the heat flux at onset of flow instability can be expressed as::

$$q_{OFI} = 0.05[R\,\rho\,Cp(W\,t_w/W_H L_H)\,U\,(T_{sat} - T_{inlet})] \quad (5)$$

where:

P_z	≡	pressure at coolant at any point 'z'	bar abs.
T_s	≡	clad surface temperature	ºC
T_{sat}	≡	saturation temperature of water	ºC
T_{inlet}	≡	water temperatur at core inlet	ºC

U	≡	water velocity in the channel	Cm/s
ΔT_{sub}	≡	water subcooling	^oC
P	≡	pressure at channel exit	bar abs.
Cp	≡	spesific heat of water	KJ/Kg ^oC
λ	≡	heat of vaporization	KJ/Kg
P_c	≡	critical pressure of coolant	bar abs.
z	≡	axial location along the channel	cm
q''	≡	heat flux	W/cm^2
q_{ONB}	≡	heat flux at onset of nucleate boiling	W/cm^2
q_{OFI}	≡	heat flux at onset of flow instability	W/cm^2
q_c	≡	burn out critical heat flux	W/cm^2
ρ	≡	density of water	Kg/m^3
W	≡	water channel width	cm
t_w	≡	water channel thickness	cm
W_H	≡	effective fuel plate width for heat transfer	cm
L_H	≡	effective fuel plate length for heat transfer	cm

This one-dimensional calculation code combines heat transfer and coolant hydraulics equations. Fuel element temperature distribution is computed by dividing fuel element into 21 nodes in the axial direction. Coolant channel is heated from two adjacent plate surfaces, with top-bottom flow direction.

2.2 NATCON Code

NATCON is used to analyze natural convection cooling. Implicitly, the equations used in the package code are prepared for calculating heat transfer in plate type fuel element cooled by water in the square channels [16]. Buoyancy force, which is the main parameter in NATCON, is computed based on the difference of water density in coolant channel and in the pool based on water temperature. Coolant steady state rate is carried out by iteration to obtain steady state between buoyancy and friction force. Input data needed are fuel element dimension and specification, nuclear power peak factor, and technical factor, inlet temperature and pressure of coolant channel, and power generation (optional). The computation results are in form of heat flux, temperature and pressure given as axial position function of active fuel element height.
The magnitude of the buoyant force of the water column that is heated is:

$$F_{\mathrm{B}} = g\left(\bar{\rho}_{\mathrm{c}} - \rho_{\mathrm{AMB}}\right) \cdot A_{\mathrm{c}} \cdot L_{\mathrm{c}} \tag{6}$$

where:
ρ_c≡ the average mass density of the heated water column, which is defined as:

$$\bar{\rho}_{\mathrm{c}} = \frac{1}{L_{\mathrm{c}}} \int_0^{L_{\mathrm{c}}} \rho_{\mathrm{c}}(x)\, dx \tag{7}$$

ρ_{AMB} ≡ mass density of the coolant in the reactor tank, |kg/m³|;
A_c ≡ cross-section area of the cooling channel, |m²|;
L_c ≡ length of the heated water column of the cooling channel, |m|.
g ≡ gravity, |m/det^2|.
The buoyant force causes a flow which is inhibited by the friction force that results in a pressure drop. The velocity of the flow will reach a certain price, called the terminal velocity in which the buoyancy of the right is balanced by the frictional force. The magnitude of the frictional force can be expressed as:

$$F_{\mathrm{F}} = \frac{(\rho\, v)_{\mathrm{in}}^2}{2\, g} \cdot A_{\mathrm{c}} \cdot \left[\frac{1}{2\, \rho_{\mathrm{in}}} + \sum_{i=1}^{n} \frac{f\ \Delta z_i}{\rho_i\ D_{\mathrm{H}}} + \frac{1}{\rho_{\mathrm{out}}}\right] \tag{8}$$

where:
ρ ≡ mass density of the coolant at the location indicated, |kg/m³|;
f ≡ friction factor;
v ≡ flow velocity in the inlet, |m/detik|;
g ≡ gravity, g = 9,80665 m/det²;
Δz_i ≡ increment of nodal in the coolant channels, |m|;
D_H ≡ diameter or hydraulic diameter of the coolant channel, |m|.

2.3. Temperaure Measurement

Fuel element temperature measurement is carried out using two instrumented oxide fuel element (IFE) equipped with thermocouple [17]. IFE RI-10 is equipped with 3 thermocouples to measure the fuel element surface temperature of difference height. Meanwhile, IFE RI-11 is equipped with 3 thermocouples, one of which is to measure fuel plate temperature and two of which are to measure inlet and outlet of coolant temperature. During measurement, these IFEs are located in the core position that has the highest radial power density factor.

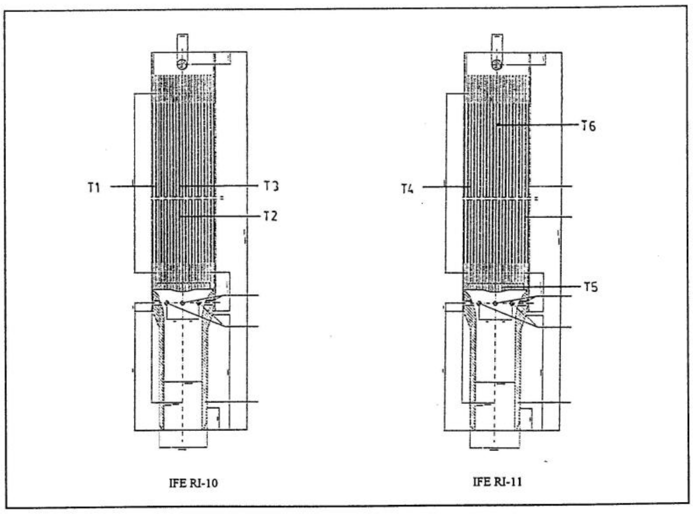

Fig. 2. Positions of Thermocouples in IFEs RI-10 and RI-11

Fuel plate temperature measurement is carried out by placing IFE at the hottest position. Then, the reactor is operated at a particular power using forced convection by operating two primary pumps. Reactor power and fuel plate temperature are measured every two hours, after the power is relatively steady. Data recorded are steady state data.

Temperature measurement at natural convection mode is performed by recording the IFE temperature indicator using a continuous data recorder. Data needed are recorded before natural convection mode is applied. Then, the two primary pumps are turned off and at the same time the primary isolation valve is closed. Data recording is carried out starting from steady state to reactor's scram and the pumps are turned back on. The assessment of the recorded data is conducted to examine the temperature characteristics as a function of time and to study the phenomena occurred.

As explained above, the measurement results are compared to the calculation analysis for many power level core according to its nominal power. The calculation analysis of forced convection mode is carried out by using HEATHYDE code. The analysis is performed for each power level with inlet coolant temperature, coolant flow rate, and measured power as input. Fuel element temperature and safety margin as a function of power in natural convection cooling mode is analyzed using NATCON code.

III RESULTS AND DISCUSSION

3.1. Forced Convection Cooling Mode

The hottest temperature of fuel element is indicated by the hottest fuel plate at the axially hottest point, as a result of heat transfer between product fission heat generated by the related fuel plate element and the amount of coolant flow rate flowing through these channels. The results of the hottest fuel plate temperature by forced convection cooling mode at nominal power of each transitional core are shown in Table 1. This table shows that the measurement point T_1 located at the most outer plate (plate number 1) has higher temperature than that of point T_3, 30 cm from the upper boundary of active fuel. This difference factor is caused by the presence of the difference of radial factor and the effect of different cooling channel. The measurement results of temperature T2 and T3, which are located at the same radial position, show that the axial temperature distribution is located below the center point of active fuel. Considering the temperature measurement result, the real condition in the transitional core, the maximum plate temperature is estimated at the most outer fuel element. This maximum plate temperature at each power level indicates an increase corresponding to nominal power of each transitional core. As shown in Core with nominal power of 10.7 MW, the highest fuel element temperature is 79.9°C. Meanwhile, Core with nominal power of 30 MW indicates 119.3°C.

The calculation results of heat transfer using forced convection cooling mode is carried out by using HEATHYDE code, by which one of the output is fuel element temperature as shown in Table 1. The maximum temperature of fuel plate as calculated using HEATHYDE code at the transitional core can be explained as follows. The maximum temperature of fuel plate with power of 10.7 MW is 82.3°C and in power of 30 MW is 127°C. The maximum plate temperature at power level is shown in Table 1. At maximum power level 30 MW, in which the maximum plate temperature reaches 127°C, experiences sub-cooled nucleate boiling. The safety margins toward DNBR and OFI are 3.56 and 2.60, respectively.

Table 1 . The highest fuel plate temperature comparation between calculation and measurement

Power (MW)	HEATHYD	Experiment	Deviation (%)
10.7	82.3	79.95	2.94
13.3	87.1	84.4	3.20
18.2	90	87.5	2.86
22.1	93.4	85.1	9.75
25.4	97.3	86	13.14
30	127	119.3	6.45

The calculation results at each transitional core using HEATHYDE code are mostly consistent with the measurement results. The deviation between calculation and measurement results is in the range of 2.86% – 13.14%. This difference is caused by the different burn-up and the assumptions used in the calculation. The comparison of results obtained by the measurement and calculation and the design data provided by the Supplier is shown in Figure 1 indicating that the criteria of safety margins of maximum fuel temperature is sufficient (<145°C) [17].

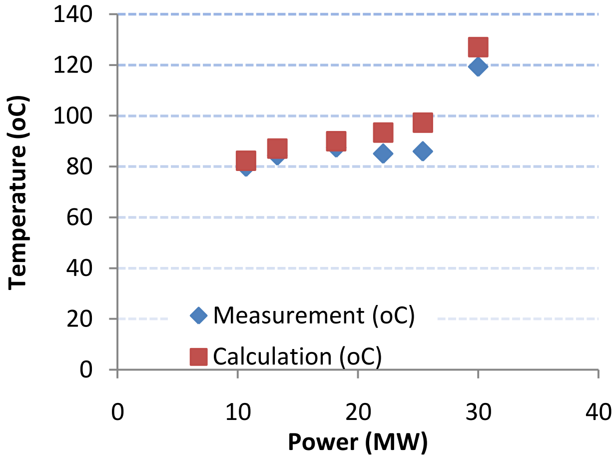

Fig. 3. Graph of maximum fuel plate temperature as a function of power, measurement results, and calculation results

3.2. Natural Convection Cooling Mode

The mechanism of natural convection in RSG-GAS core occurs through two natural circulation valves installed on the wall of a space below the core. The maximum fuel plate temperature obtained from the measurement of each power is shown in Table 2. The measurement results indicate the presence of an increase in temperature corresponding to nominal power of each power. The maximum fuel plate temperature is 63.4°C on nominal power of 10.7 MW, while on power of 30 MW the maximum fuel plate temperature is 121°C. This shows that the fuel temperature after scram depends on the reactor power and the operation period before reactor scram.

Table 2. The measured maximum fuel plate temperature right after scram (natural convection cooling mode)

Power before scram (MW)	Maximum fuel plate temperature
10.7	63.36
13.3	69.70
18.2	74.81
22.1	76.24
25.4	-
30	121

The computation results of natural convection cooling mode using NATCON code are shown in Table 3. In addition to fuel element, ΔT_{ONB}, T_{ONB} and DNBR are also shown in those table. The computation results indicate that the fuel element temperature increase in accordance with the rise of reactor power. Figure 2 shows that the onset of nucleate boiling (ONB) occurs at power 826 kW or about 2.8% of the nominal power of typical working core. DNBR at this power level is 4.69.

Table 3. Analysis results of RSG-GAS transitional core with natural convection cooling mode using NATCON code

REACTOR POWER (kW)	$T_{max.}$ FUEL PLATE TEM (°C)	ΔT_{ONB} (°C)	T_{ONB} (°C)	DNBR
100	58.18	66.75	125.23	19.72
200	66.92	58.7	125.56	12.34
300	76.21	49.91	125.82	9.36
400	85.71	40.93	126.03	7.65
500	94.47	32.02	126.22	6.61
600	104.04	23.27	126.38	5.84
700	112.98	14.64	126.54	5.25
800	121.8	6.11	126.68	5
826	128.13	0	126.77	4.69

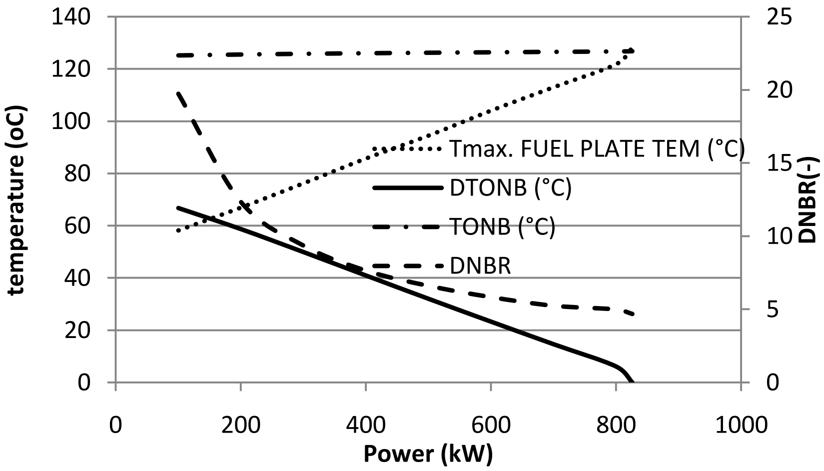

Fig. 4. Safety margin of natural convection by using NATCON code

As the RSG-GAS core cycle is only 25 days, the maximum decay heat right after scram is ± 2.25% of nominal power before scram and decay heat will decreases exponentially. Taking into account the decay heat occurred, RSG-GAS pool is able to cool the decay heat occurred in the reactor core when the power supply to the primary cooling fails.

Reactor protection system is adjusted so that the reactor cannot be operated without primary coolant at power level above 300 kW (1% of nominal power). It means that the setting of the related protection system has sufficient safety margin. The comparison between the measurement and calculation results and the design data shows that the safety operating margin of fuel element temperature is fulfilled (<145°C).

IV CONCLUSION

Based on the evaluation results of maximum fuel plate temperature characteristics in the transitional core with either forced or natural convection cooling mode, it can be concluded that: the measured maximum temperature of the fuel plate element of 121°C indicates that the maximum operating margin is still fulfilled by the permitted fuel plate element temperature margin, which is <145°C. The evaluation results on the steady state core with forced convection cooling mode by using HEATHYDE code show that, because the maximum fuel plate temperature is still lower than the boiling temperature, the coolant flow is still one phase, i.e. liquid. In addition, the safety margin towards DNBR is still sufficient, i.e. >1.5. Meanwhile the evaluation results on steady state core with natural convection cooling mode by using NATCON code show that, in the cooling by natural convection, the residual flow is able to cool decay heat occurring in the reactor core after the pump is turned off. From the thermo-hydraulics aspect point of view, the results of the measurement and evaluation carried out show that RSG-GAS is able to operate to nominal power of 30 MW safely.

ACKNOWLEDGEMENT

The author would like to appreciate to the Management of PTKRN and PRSG, BATAN. The author also appreciate to the operator and supervisor of the RSG GAS for data taking of reactor operation.

REFERENCES

[1]. Badan Tenaga Nuklir Nasional, *Safety Analysis Report*, Vol. 8, Jakarta, October 1998.

[2]. Ahmad Lashkari, Loss of Flow Accident Analyses in Tehran Research Reactor, *24 International Conference Nuclear Energy for New Europe*, Portorozslovenia,NENE, 2015

[3]. Mohd Faiz Salim, Mohd Rafie Samsudin, Mohd Rizal Mamat, et al., An Approach to Model Reactor Core Nodalization for Deterministic Safety Analysis, the American Institute of Physics, *AIP Conference Proceedings* 1704, 020007, 2016.

[4]. Badrun Nahar Hamid, Md. Altaf Hossen, Sheikh Md. Tazul Islam et al., Modelling an Unprotected Loss-of-Flow Accident in Research Reactors using the Eureka-2/Rr Code, *Journal of Physical Science*, Vol. 26(2), 73–87, 2015

[5]. Pavel Zitek1, and Vaclav Valenta11, Solution of heat removal from nuclear reactors by natural convection, Czech Republic, *EDP Science*, 2014

[6]. A. Hussain, F.Abolaban, S.M. Khubaib, S.Mubin and I.Ahmed, Analysis of PHWR using COBRA-3C/RERTR, IJST, *Transactions of Mechanical Engineering*, Vol. 39, No. M1+, 2015, 233-242,

[7]. F. M. Bsebsu and G. Bede, *Theoretical Study in Single-Phase Forced-Convection Heat Transfer Characteristics for Narrow Annuli Fuel Coolant Channels*, Department for Energy, Budapest University of Technology and Economics, Hungary, 2000

[8]. Deepak Sharma, K.M.Pandey ' Thermal Hydraulics Problems in Nuclear Reactors: A Review, *IJISET - International Journal of Innovative Science, Engineering & Technology*, Vol. 2 Issue 9, 2015

[9]. C. Housiadas, *Thermal-hydraulic Calculations for the GRR-1 Research Reactor Core Conversion to Low Enriched Uranium Fuel*, National Centre for Scientific Research "Demokritos"Institute of Nuclear Technology and Radiation Protection P.O. Box 60228, 15310 Agia Paraskevi, Athens, Greece, 2000.

[10]. W. L. Woodruff and J. R. Deen, C. Papastergiou, Transient analyses and Thermal-hydraulic Safety Margins for The Greek Research Reactor (GRRI), *International Meeting on Reduced Enrichment for Research and Test Reactors*, Williamsburg, Virginia, USA, 1994

[11]. R.S. Smith and W.L. Woodruff, *A Computer Code, NATCON, for the Analysis of Steady State Thermo-hydraulics and Safety Margin in Plate-Type Research Reactor Cooled by Natural Convections*, Argonne National Laboratory, Argonne Illinois, 1988.

[12]. M.M. EL-WAKIL, *Nuclear Heat Transport*, American Nuclear Society La Grange Park, Illinois, 1978

[13]. Nabbi, R, *HEATHYD, Steady-State Thermal Hydraulic Analysis of Low-Enriched U Fuel Reactor*, Nuclear Energy Agency of the OECD (NEA), Volume 41, Issue 03, 1989.

[14]. L.A. Khan and R. Nabbi, *Heat Transfer Analysis of the Existing and Proposed LEU Cores of Pakistan Research Reactor*, Report 2118, Germany, 1987.

[15]. R.S. Smith and W.L. Woodruff, *A Computer Code, NATCON, for the Analysis of Steady State Thermo-hydraulics and Safety Margin in Plate-Type Research Reactor Cooled by Natural Convections*, Argonne National Laboratory, Argonne Illinois, 1988.

[16]. Sudarmono, et.al., Measurement of Fuel Element Temperature in the Core of RSG-GAS, *Seminar Proceeding of Scientific Meeting and Presentation of Basic Researches on Nuclear Science and Technology*, Yogyakarta, 1988.

[17]. R. Nabbi, Uju Jujuratisbela, Kurnia Putranta, Endiah Puji Hastuti, Thermo-hydraulics of the Indonesian Research Reactor RSG-GAS, *ASSR-2*, Jakarta, September 1989.

Biographies and Photographs

Sudarmono 	Senior researcher with expertise reactor technology. Born in Yogyakarta, June 17, 1957. Graduated from MIPA Physics UGM in 1987. Working for the first time in PRSG-BATAN 1985-2002, has been following commissioning RSG GAS reactor 1987-1989. Work at the Center for Advanced Reactor Systems-BATAN 2002-2005, and graduated majoring in Mathematics Physics master program in 2004. Working in PTKRN-BATAN 2005 until now. Being the chief investigator in the CRP-IAEA with title StudyAnalysis of Transmutation System using Accelerator Driven Subcritical Reactor under Accelerator-Driven System (ADS) Applications and Use of Low Enriched Uranium in ADSs-CRP System 2017-2019.
Endiah Puji Hastuti 	As a researcher with reactor technology expertise. Graduated from UNDIP chemical engineering department in 1985. Graduated from ITB nuclear power engineering department in 1995. Started work in PRSG-BATAN 1987, and followed the commissioning of GA Siwabessy's multipurpose reactor. Worked in nuclear technology and nuclear safety center 2005 up to now.

Sudarmono." Characterization Of Oxide Fuel Element Temperature Of RSG GAS By Using Forced And Natural Convection Cooling Mode. " The International Journal of Engineering and Science (IJES) 7.5 (2018): 49-56