International Journal of Engineering, Mathematical and Physical Sciences
ISSN: 2517-9934
Vol:5, No:1, 2011

The Comparative Investigation and Calculation of Thermo-Neutronic Parameters on Two Gens II and III Nuclear Reactors with Same Powers

Mousavi Shirazi, Seyed Alireza [1*]; Rastayesh, Sima [2]

***Abstract*—**Whereas in the third generation nuclear reactors, dimensions of core and also the kind of coolant and enrichment percent of fuel have significantly changed than the second generation, therefore in this article the aim is based on a comparative investigation between two same power reactors of second and third generations, that the neutronic parameters of both reactors such as: K_∞, K_{eff} and its details and thermal hydraulic parameters such as: power density, specific power, volumetric heat rate, released power per fuel volume unit, volume and mass of clad and fuel (consisting fissile and fertile fuels), be calculated and compared together. By this comparing the efficiency and modification of third generation nuclear reactors than second generation which have same power can be distinguished.

In order to calculate the cited parameters, some information such as: core dimensions, the pitch of lattice, the fuel matter, the percent of enrichment and the kind of coolant are used. For calculating the neutronic parameters, a neutronic program entitled: SIXFAC and also related formulas have been used. Meantime for calculating the thermal hydraulic and other parameters, analytical method and related formulas have been applied.

***Keywords*—**Nuclear reactor, second generation, third generation, thermo-neutronics parameters.

Nomenclature

η : Thermal fission coefficient
f: Thermal absorption coefficient
p: Resonance escape probability
ε: Fast fission coefficient
P_{FNL}: Fast neutrons non leakage probability
P_{THNL}: Thermal neutrons non leakage probability
P_{TNL}: Total neutrons non leakage probability
$\dot{q}_{th}$: Released heat rate in volume of reactor core
$\dot{q}'$: Linear heat rate
r: Enrichment of fuel
M_{ff}: Mass number of fissile
M_{nf}: Mass number of fertile
σ_f : Fission cross section
R_{fuel} : Fuel rod radius
dz : Fuel rod height
ρ_{pellet} : Fuel pellet density
n: Fuel rods numbers

Authors are with Islamic Azad University, South Tehran Branch,Department of Physics,Tehran, Iran.(e-mail:a_moosavi@azad.ac.ir)

I. INTRODUCTION

IN view of the great advancements in the nuclear reactors technology, the phenomenal and significant changes in the evolution of nuclear reactors is observed. Since the first nuclear reactor made in 1948 until modern reactors, great changes are obvious. These major changes are: the kind of reactor design, the percent of fuel enrichment, the kind of coolant and neutron moderator, the dimensions of core and safety are referred [1]-[2].

In this paper, the reactors considered here are Boiling Water Reactor (BWR): Kashiwazaki-Kariwa (Unit-6) Japan, and Forsmark (Unit-3) Sweden, respectively, that have same thermal power: $\dot{q}_{th}$=3700MWt =1235MWe [3]-[4]. Both Kashiwazaki and Forsmark reactors are enumerated as second generation nuclear reactors.

II. METHODOLOGY

The related information to these reactors is shown in Table I:

TABLE I
THE RELATED SPECIFICATIONS TO BOTH UNDER STUDY REACTORS [3]-[4]

Forsmark reactor	Kashiwazaki reactor	Various pararmeters
Pellet of UO_2	Pellet of UO_2	The kind of fuel
Zr-2	Zr-2	The kind of clad
2.25 %	2.6 %	The percentage of fuel enrichment
8.7 mm	10.3 mm	The height of fuel pellet
8.2 mm	10.4 mm	The diameter of fuel pellet
0.63 mm	0.86 mm	The thickness of fuel pellet
0.08 mm	0.09 mm	The thickness of fuel external gap (δ_g)
3914 mm	4066 mm	The height of fuel rod
9.62 mm	12.3 mm	The external diameter of fuel rod
8x8	8x8	The model of fuel assembly
100	60	The fuel rods numbers at each fuel assembly
1.27 cm	1.6 cm	The pitch of fuel rods
700	872	The numbers of fuel assembly
15.45 cm	15.5 cm	The pitch of fuel assembly
3.75 m	3.71 m	The height of core
4.6 m	5.16 m	The diameter of core
H2O	H2O	The coolant
Cylindrical	Cylindrical	The model of core vessel
20.8 m	21.0 m	The height of vessel
6.4 m	7.1 m	The inner diameter of vessel

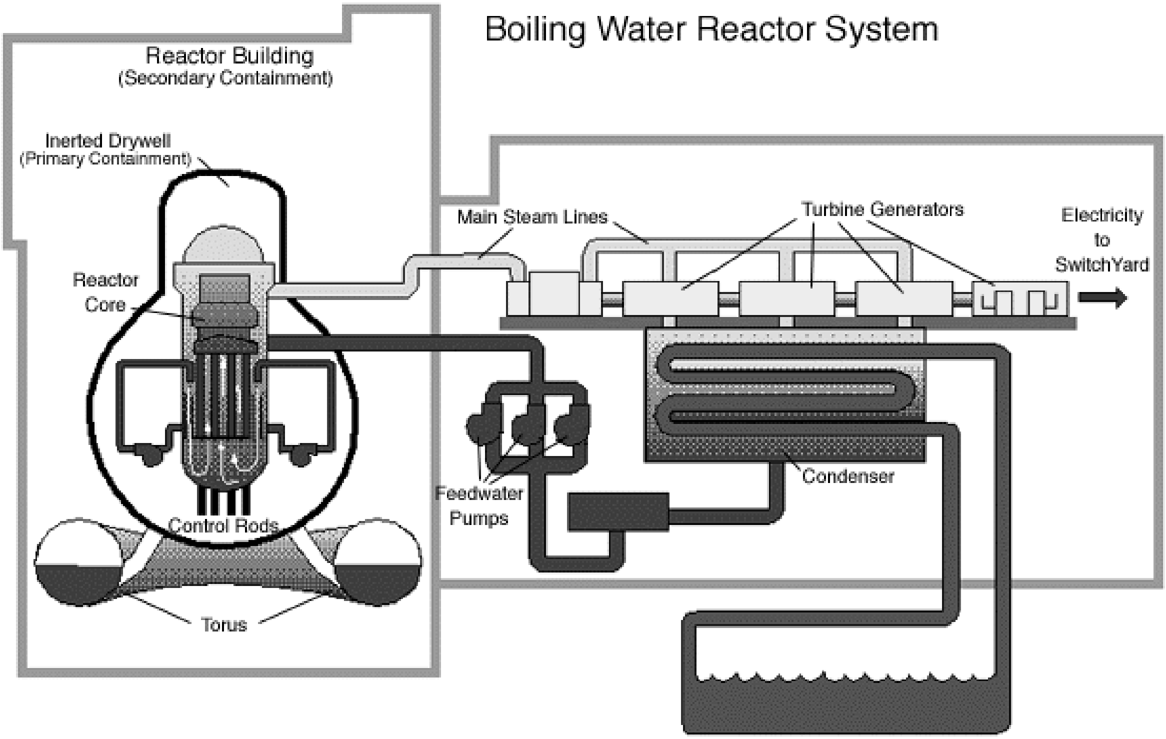

Fig. 1 The view of a BWR reactor [3]

The Neutronic Parameters

In this paper by using the neutronic software titled: SIXFAC and also by applying the related formulas, the values of infinite multiplying coefficient (K_∞), effective multiplying coefficient (K_{eff}) and also the fundamental components of both coefficients for the two mentioned reactors in super critical state have been determined. In addition according to fuel enrichment percent in two cited reactors, the amounts of applied fuel in both reactors have been accurately computed.

The effective multiplying coefficient is: the ratio of generated neutrons in every generation to generated neutrons in last generation. So to operate the nuclear reactor in steady state, this parameter should be: 1 means the generated neutrons in every generation are equal with neutrons which have absorbed or leaked in last generation that means: critical state. The minimum value of K_{eff} is: 0 and maximum of it is: υ namely: *2.43*.

The effective multiplying coefficient is [5]:

$$K_{eff} = \eta . f . p . \varepsilon . P_{FNL} . P_{THNL} = \eta . f . p . \varepsilon . P_{TNL} \quad (1)$$

As the thermal fission coefficient (η) is [5,6]:

$$\eta = \upsilon \frac{\Sigma_f^F}{\Sigma_{fm}^F} = \frac{\upsilon . N^{235} . \sigma_f^{235} . g_f^{235}}{N^{235} . \sigma_a^{235} . g_a^{235} + N^{238} . \sigma_a^{238} . g_a^{238} + N^O . \sigma_a^O} \quad (2)$$

$$N^{235} = \frac{m^{235} . A}{M^{235}} \quad (3)$$

$$N^{238} = \frac{m^{238} . A}{M^{238}} \quad (4)$$

$$N^O = 2 N^U = 2 \frac{m^U . A}{M^U} = 2 \frac{m^U . A}{r M^{235} + (1-r) M^{238}} \quad (5)$$

Also the thermal absorption coefficient (f) is [5]:

$$f = \frac{\Sigma_a^F}{\Sigma_a^F + \Sigma_a^M} = \frac{N^{235} . \sigma_a^{235} . g_a^{235} + N^{238} . \sigma_a^{238} . g_a^{238} + N^O . \sigma_a^O}{N^{235} . \sigma_a^{235} . g_a^{235} + N^{238} . \sigma_a^{238} . g_a^{238} + N^O . \sigma_a^O + N^H . \sigma_a^H + N^{O_M} . \sigma_a^O} \quad (6)$$

$M_{U^{235}} = M_{ff} = 235.040$ gr/mol ,

$M_{U^{238}} = M_{nf} = 238.029$ gr/mol

$M_{O_2} = 31.998 \ gr / mol$

$\sigma_f^{235} = 570 \, barn$, $\sigma_a^{235} = 594 \, barn$, $\sigma_f^{238} = 2.4 \, barn$

$g_f^{235} = 0.8956$, $g_a^{235} = 0.9118$, $g_a^{238} = 1.0198$

The resonance escape probability for fast neutrons (p) also is calculated as follows [5]:

$$\xi = \frac{A}{A + \frac{2}{3}} \quad (7)$$

$$I_{eff} = 3.9 \times \left(\frac{\Sigma_s}{N^{238}} \right)^{0.415} \quad (8)$$

$$p = e^{-\left(\frac{N^{238}}{\zeta . \Sigma_s} \right) \times I_{eff}} \quad (9)$$

$$\Sigma_s = N^{235} . \sigma_s^{235} + N^{238} . \sigma_s^{238} + N^O . \sigma^O + N^H . \sigma^H + N^{O_M} . \sigma^O + N^{Zr} . \sigma^{Zr} \quad (10)$$

If the enrichment of fuel is 100% then the resonance escape probability for fast neutrons (p) will be maximum value.

The Thermal Parameters and The Rest Parameters

According to the enrichment of applied fuel and its mass can write [7]:

$$r = \frac{m_{ff}}{m_f} \tag{11}$$

$$f_{fm} = \frac{m_f}{m_{fm}} = \frac{rM_{ff} + (1-r)M_{nf}}{rM_{ff} + (1-r)M_{nf} + M_{O_2}} \tag{12}$$

Where [8]:

$$m_{UO_2} = m_{fm} = \frac{N_{fm}.M_{fm}}{A} = \rho_{fm}.V_{fm} \tag{13}$$

$$m_U = m_f = m_{fm}.f_{fm} = \rho_{fm}.V_{fm}.\frac{rM_{ff} + (1-r)M_{nf}}{rM_{ff} + (1-r)M_{nf} + M_{O_2}} \tag{14}$$

$$m_{U^{235}} = m_{ff} = m_{fm}.f_{fm}.r = \rho_{fm}.V_{fm}.\frac{rM_{ff} + (1-r)M_{nf}}{rM_{ff} + (1-r)M_{nf} + M_{O_2}}.r \tag{15}$$

$$m_{U^{238}} = m_{nf} = m_f - m_{ff} \tag{16}$$

$$\rho_{UO_2} = \rho_{fm} = 10960\ kg/m^3$$

For calculation of specific power, power density and the rate of released heat in reactor volume unit [7]:

S.P= the total power to mass of heavy atoms =

$$\frac{\dot q_{th}}{n.(\pi R_{fuel}^2 dz).\rho_{pellet}.f_{fm}} = \frac{\dot q'}{n\pi(R_f + \delta_{gap})^2.\rho_{smeared}.f_{fm}} \tag{17}$$

Where:

$$\dot q' = \frac{\dot q_{th}}{l} \tag{18}$$

$$\rho_{smeared} = \frac{\pi R_f^2.\rho_{pellet}}{\pi(R_f + \delta_{gap})^2} \tag{19}$$

The Fig. 2 shows modeling the fuel rods in each applied fuel assembly in boiling water reactor that has square fuel networks.

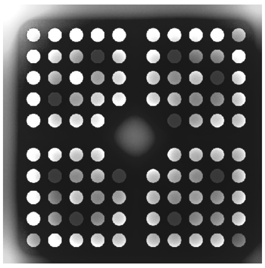

Fig. 2 The sample fuel assembly in a boiling water reactor [9]

In a fuel assembly with square modeling, the power density [7]:

$$P.D = \frac{\dot q'}{pitch^2} \tag{20}$$

Also for calculating the power in volume unit of reactor core can write [8]:

$$\dot q''' = \frac{\dot q_{th}}{V} \tag{21}$$

In addition to calculate the power in volume unit of fuel ($\dot q_c'''$) for a cylindrical nuclear reactor [8]:

$$\dot q''' = \dot q_c''' J_0(\frac{2.405\ r}{R_e})\cos(\frac{\pi.z}{He}) \tag{22}$$

$$\dot q_{th} = \int_{-l/2}^{+l/2} \dot q''' A_s.dz = \frac{2}{\pi}\dot q'''.A_s.H_f \tag{23}$$

Thus for calculating $\dot q_c'''$ the equation (24) is outcome:

$$\dot q_c''' = \frac{\dot q_{th}}{0.275\ n.A_s.H_f} \tag{24}$$

III. RESULTS AND CONCLUSION

By using the information of Table 1 and also by application the mentioned formulas and by using the SIXFAC neutronic program, the thermo-neutronic parameters and other parameters of both reactors (Kashiwazaki Kariwa and Forsmark) according to the Figs:3,4 and also Tables:2,3 have been derived.

According to the obtained values for observable neutronic parameters in Table 2 is deduced that the neutronic parameters including K_{∞}, K_{eff} and their components in the Kashiwazaki reactor (third generation reactor) for finite numbers of neutrons in neutron cycle are more than the Forsmark reactor (second generation reactor). Thus in the super critical state (without consideration the effect of negative reactivity injection derived of control rod) is concluded that the neutronic parameters amounts in the third generation reactor are more than the second generation reactor in the same power. According to the obtained values of thermo-hydraulic parameters which are shown in Table 2 is perceived that the Kashiwazaki reactor has more volume for core, fuel and fuel mass than the Forsmark reactor. Also the powers in volume unit of core and fuel of Kashiwazaki reactor are less than the Forsmark reactor. Therefore, it can finally be concluded that the performance of the third generation nuclear reactors is safer than the second generation nuclear reactors from the operation aspect.

REFERENCES

[1] Y. Tachibana, H. Sawahata, T. Iyoku, T. Nakazawa., Reactivity control system of the high temperature engineering test reactor. Nuclear Engineering and Design 233, 2004, pp.89-101.

[2] F. Zhao, K. Cheung, R. Yeung, Optimal power control system of a research nuclear reactor. Nuclear Engineering and Design 219, 2003, pp.247-252.

[3] http://www.insc.anl.gov/cgibin/rperl/sql_interface?view=newrx_data&qvar=oracle_id&qval=213#ref_opstr_data

[4] http://www.insc.anl.gov/cgibin/rperl/sql_interface?view=newrx_data&qvar=oracle_id&qval=310#fuel_data

[5] J. Lamarsh, *"Introduction to Nuclear Engineering (3 rd Ed)* "; Addison, Wesley, pp.35-80, 2001.

[6] J. Stacy, *"Nuclear Reactor Physic"*; JW, chapter (2, 3), 1999.

[7] E. Neil and M. Kazimi, *"Nuclear Systems 1 Thermal Hydraulic Fundamentals*"; Hemisphere Publishing Corporation, pp.9-50, 1990.

[8] E. Wakil, *"Nuclear Heat Transfer"*; Hemisphere Publishing Corporation, pp.103-109, 1991.

[9] J. Benítez, P. Martínez, H. Pérez, International cooperation on control for safe operation of nuclear research reactors. Progress in Nuclear Energy 46, 2005, pp.321-327.

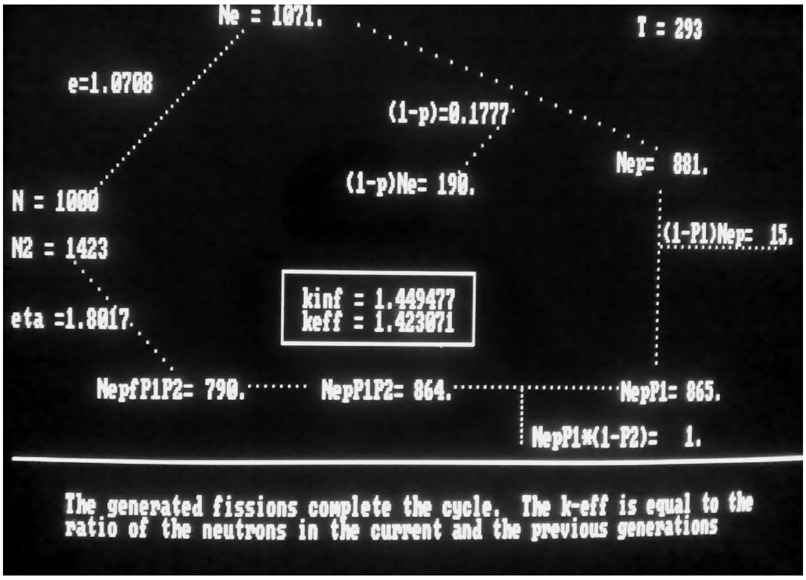

Fig. 3 The neutron cycle defined by the SIXFAC neutronic software for Kashiwazaki reactor

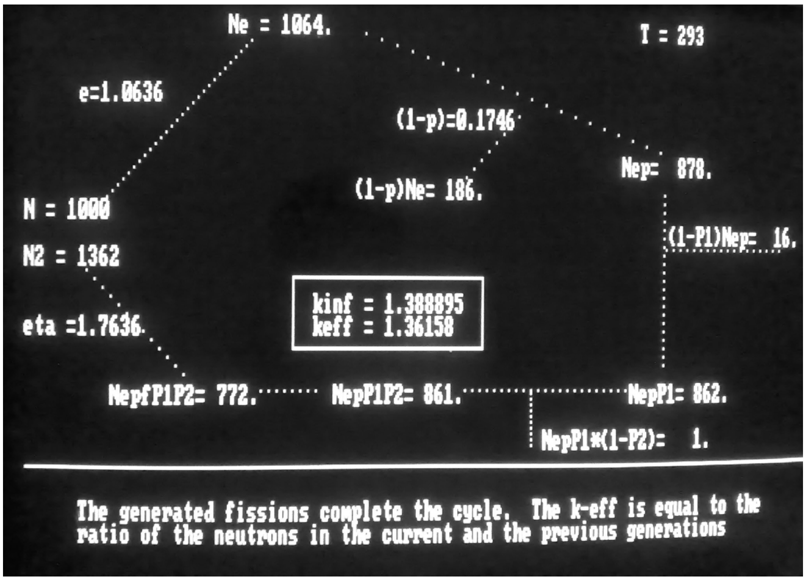

Fig. 4 The neutron cycle defined by the SIXFAC neutronic software for Forsmark reactor

TABLE II
THE CALCULATED AND OBTAINED VALUES FOR NEUTRONIC PARAMETERS

via SIXFAC program for Forsmark reactor	via analytical method for Forsmark reactor	via SIXFAC program for Kashiwazaki reactor	via analytical method for Kashiwazaki reactor	The neutronic parameters in super critical state
1.3615	1.3140	1.4230	1.3868	K_{eff}
1.3888	1.3384	1.4494	1.4039	K_{∞}
1.7636	1.7500	1.8017	1.7902	η
0.8966	0.8945	0.9143	0.9090	f
0.8254	0.8203	0.8223	0.8205	p
1.0636	1.0423	1.0708	1.0515	ε
0.9818	0.9826	0.9823	0.9880	P_{FNL}
0.9989	0.9992	0.9988	0.9998	P_{THNL}
0.9807	0.9818	0.9811	0.9878	P_{TNL}

TABLE III
THE CALCULATED AND OBTAINED VALUES FOR THERMAL PARAMETERS

Forsmark reactor	Kashiwazaki reactor	The thermal parameters and the rest parameters
14.468 m^3	18.071 m^3	The volume of fuel (V_{UO_2})
5.445 m^3	7.206 m^3	The volume of clad (V_{cl})
19.914 m^3	25.277 m^3	The volume of fuel rod
62.321 m^3	77.582 m^3	The volume of core
669.134 m^3	831.430 m^3	The volume of vessel
158569.28 kg	198058.16 kg	The total mass of existent UO_2 in reactor core (m_{fm})
139774.20 kg	174581.72 kg	The total mass of existent U in reactor core (m_f)
136629.28 kg	170042.60 kg	The total mass of existent U^{238} in reactor core (m_{nf})
3144.92 kg	4539.12 kg	The total mass of existent U^{235} in reactor core (m_{ff})
26.469 W/gr	21.194 W/gr	Specific power (S.P)
83.728 kW/lit	67.940 kW/lit	Power density (P.D)
59.370 kW/lit	47.691 kW/lit	Power in volume unit of core $(\dot{q}''')$
675.631 kW/lit	532.272 kW/lit	Power in volume unit of fuel $(\dot{q}_c''')$