EUROPEAN ACADEMIC RESEARCH
Vol. VI, Issue 7/ October 2018

ISSN 2286-4822
www.euacademic.org

Impact Factor: 3.4546 (UIF)
DRJI Value: 5.9 (B+)

Nuclear and Radiological Safety of PWR Reactor at Steady State and Transient Conditions

F. A. GABER
M. A. EL-SARRAF[1]
Nuclear and Radiological Regulatory Authority, Cairo, Egypt
A. YOUSSEF
Faculty of Science, Cairo University, Egypt
W. A. KANSOUH
Nuclear Research Centre, Atomic Energy Authority, Cairo, Egypt

Abstract:

This study involved the Nuclear and Radiological Safety of the PWR with three different fuel types, a simplified lumped parameter mathematical modeling of the PWR based on lumped parameters technique, this model includes the neutronics, core heat transfer and piping. A computer code of PWR is designed by FORTRAN language to study the dynamic performance in the steady state and transient conditions using different fuels: UO_2, $ThO_2 - UO_2$ *and* $U - ZrH_{1.6}$. *Also, an experimental simulation for a cylindrical reactor (like PWR) was performed using* Cf^{252} *source, and the study of neutrons decay through a water/iron bi-layer assembly.*

Key words: PWR, modeling, dynamic performance, steady state, transient conditions, experimental simulation.

1) INTRODUCTION

The introduction of nuclear energy as competitive source of energy in Egypt underlines the need for scientific and technical

[1] Corresponding author: 3 Ahmed El-Zomor St. Nasr City, 11762 Cairo, Egypt.

base of skilled personnel to handle design, operation and maintenance problems as well as safety analysis review activities.

This paper assesses the dynamic performance of uranium oxide, mixed uranium-thorium oxides and hydride fuels in pressurized water reactor (PWR) cores in steady state and transient conditions.

PWR Mathematical Model

The reactor model is represented by three axial regions; the lower plenum, the core and the upper plenum. The reactor power was modeled using the point kinetics equations with six groups of delayed neutrons [1, 2]. Thus, the precursor concentration for the ith delayed neutron group (c_i) is,

$$\frac{dc_i}{dt} = \left[\frac{\beta_i P}{\Lambda}\right] - \lambda_i c_i \qquad \dots (1)$$

Where,
P = Power,
β_i= Delayed neutron fraction for the ith delayed neutron group,
λ_i= Decay constant for the ith delayed neutron group,
Λ= mean neutron generation times.

The reactor power and reactivity feedback due to changes of fuel temperature, coolant temperature and pressure are as follows,

$$\frac{dP}{dt} = \left[\frac{\rho-\beta}{\Lambda}\right] P + \sum_{i=1}^{6} \lambda_i c_i \qquad \dots (2)$$

Where,
β : Total delayed neutron fraction,
P : reactor thermal nuclear power,
ρ : Total reactivity, ($\rho = \rho_{cr} + \rho_{fb}$)
ρ_{cr}: Reactivity due to control rod movement,
ρ_{fb}: Reactivity feedback, which can be written explicitly as follows:

$$\rho_{fb} = \alpha_F \Delta T_F + \alpha_{co} \Delta T_{co} + \alpha_P \Delta P_P \qquad \dots (3)$$

Where,
ΔT_F: Change in fuel temperature
ΔT_{co}: Change in coolant temperature in core
ΔP_P: Change in coolant pressure
α_P: Pressure coefficient of reactivity

Reactivity coefficients and kinetic coefficients are important safety parameters. Table (1) shows the fuel Doppler coefficient (α_F), coolant temperature coefficient of reactivity (α_{co}) and delayed neutron fraction (β) parameters of different fuels [3, 4].

Table 1: parameters for UO_2, ThO_2-UO_2 and U-$ZrH_{1.6}$ fueled PWRs

Fuel	U-$ZrH_{1.6}$	ThO_2-UO_2	UO_2
Fuel Doppler coefficient of reactivity α_F	-3.3 pcm/K	-2.36 pcm/K	-2 pcm/K
Coolant temperature coefficient of reactivity α_{co}	-3.2 pcm/K	-5.63 pcm/K	-10.7 pcm/K
Delayed neutron fraction β	0.00742	0.0085	0.0069

Negative coefficients bring negative feedback which is important for safe reactor operation. The lumped parameters core heat transfer model used in this study utilizes one fuel lump and two coolant lump.

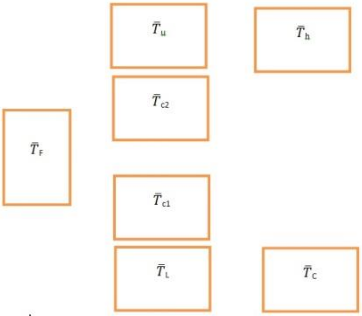

Figure (1) Schematic diagram of reactor thermal model lumps.

The Schematic diagram of reactor thermal model lumps is shown in Figure (1). Applying the energy conservation equations on each of these three lumps, the coolant in the upper and lower plenum and the coolant in the hot and cold leg piping, and assuming the average temperature in each lump equal to the outlet temperature, the following sets of equations are obtained:

For the fuel lump:

$$M_F C_{PF} \frac{dT_F}{dt} = -U_{fc} A_{fc} \left(T_F - T_{c_1}\right) + P$$

For coolant lumps:

$$M_{c_1} C_{P_1} \frac{dT_{c_1}}{dt} = \frac{1}{2} U_{fc} A_{fc} \left(T_F - T_{c_1}\right) + w_{pi} \left(T_L - T_{c_1}\right) C_{P_1}$$

$$M_{c_2} C_{P1} \frac{dT_{c_2}}{dt} = \frac{1}{2} U_{fc} A_{fc} \left(T_F - T_{c_1}\right) + w_{pi} \left(T_{c_1} - T_{c_2}\right) C_{P1}$$

For upper and lower plenums:

$$M_u \frac{dT_u}{dt} = w_{pi} \left(T_{c_2} - T_u\right)$$

$$M_L \frac{dT_L}{dt} = w_{pi} (T_C - T_L)$$

For the hot and cold leg piping:

$$M_C \frac{dT_C}{dt} = w_{pi} \left(T_{P_0} - T_C\right)$$

$$M_h \frac{dT_h}{dt} = w_{pi} (T_u - T_h)$$

Where :

P = reactor power,
A_{fc} = heat transfer area between fuel and coolant,
W_{pi} = primary coolant flow rate ,
U_{fc} = overall heat transfer coeff. between fuel & coolant ,
T_{c1}, T_{c2} = temp. of coolant in the first and second core node respectively,
M_{c1}, M_{c2} = mass of coolant in the first and second core node respectively,
T_u, T_L = coolant temp. in the upper and lower plenum respectively,
M_u, M_L= mass of coolant in the upper and lower plenum respectively,
T_c, T_h= temp. of coolant in the cold and hot leg piping respectively,
M_c, M_h = mass of coolant in cold and hot leg piping,
T_{po} = primary coolant temp. leaving the steam generator.

2) Program description

The "PWR FORTRAN" program calculates the transient behavior of the state variables using the forcing and steady state parameters as an input. This program consists of three sections, the first is the main program that includes the forcing values, the main basic data, the geometrical parameters and the steady state values as an input to the PWR model. The second section is the subroutine "DIFF" which includes the state equations, the heat transfer correlations, algebraic variables and physical coolant properties as time dependent for the PWR model. The third section is the subroutine "Merson" – numerical method for solving a set of coupled non-linear differential equations – which includes the algorithm for solving the system equations. The reactor model developed consists of fourteen first order differential equations.

2.1. Steady State Calculations

The steady state conditions (normal operation) are required as an input to the dynamic model algorithm. The steady state results are determined by using the previously reactor model state equations by letting the time derivatives equal to zero and solving the resulting set of equations. Table (2) show the steady state results of (600 MW) PWR .

Table (2) : Steady state results of UO_2, ThO_2-UO_2 and U-$ZrH_{1.6}$ fueled PWR of 600 MW_e [5, 6]

Rated thermal powerP_0	1882 MW
Average fuel temperature T_F	837.778 $°C$
Temperature of the coolant in the 1st core node T_{C_1}	306.11 $°C$
Temperature of the coolant in the node T_{C_2}	324.5 $°C$
coolant temperature in the 2nd core lower plenum T_L	287.72 $°C$
coolant temperature in the upper plenum T_U	324.5 $°C$
temperature in the hot leg piping T_h	324.5 $°C$
temperature in the cold leg piping T_C	287.72 $°C$
primary temperature leaving steam generator T_{P_0}	287.72 $°C$

3) Experimental study

Since Cf-252 spontaneous fission source produce neutrons of energies that follow a maxwellian distribution as those emitted from nuclear fission reactors and their associated gamma ray source term resemble prompt – fission γ-ray spectrum for thermal neutron fission of ^{235}U to reasonable extent; we would utilize this source for a comparative study [7].

The (Cf-252 of 100 μgm – june 2001) source was used. It decays by α particles with an effective half life time of 2.65 years. The source spectrum and its specifications are found elsewhere [7].

In This research we will be considering a cylindrical reactor that consists of (1) reactor core (2) thermal shield and (3) pressure vessel. This configuration will be experimentally simulated by considering an experimental setup of (radioactive source + collimator – samples shoulder – detector + collimator) as shown in figure (2). The Cf-252 radioactive source (1) which will be representing the core, the water layer (2) will be representing the thermal shield and iron layer (3) will be representing the pressure vessel. The transmitted thermal neutron flux through the water/iron bi-layer assembly will be measured with a He_3 detector.

The bi-layer assembly will be of fixed thickness 16 cm and area 256 cm^2. In this construction the relative thicknesses of water to iron will be changing and give the same total thickness for all readings. The weight per cm^2 ($g \cdot cm^{-2}$) for each configuration will be calculated and the alternative thicknesses variation is helpful with understanding the priority of either medium in terms of thermal neutron attenuation [8].

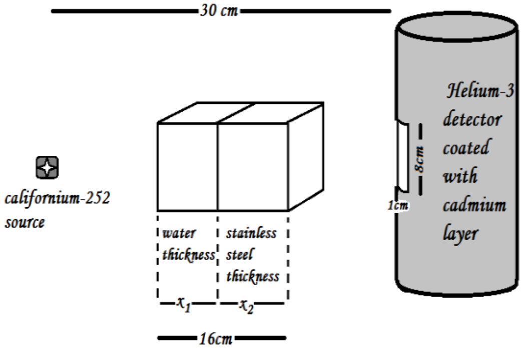

Figure (2): Experimental facility

4) Results and discussion

The results of the PWR system with uranium dioxide (UO_2) fuel are presented and compared with corresponding results of PWR fueled with thoria-urania fuel (ThO_2-UO_2) and the hydride fuel (U-$ZrH_{1.6}$) in the following cases:

a) Reactivity decrease

Inserting the absorber control rods would give rise to a negative reactivity ramp. The relative reactor power will drop and as a consequence also the fuel and coolant temperature would decrease [6]. These negative changes of temperature now produce a positive reactivity feedback. The power decrease is limited by this positive reactivity feedback.

Figure (3) shows the transient response for UO_2, ThO_2-UO_2 and U-$ZrH_{1.6}$ fueled PWR system, for a 0.01 ΔK/K step decrease in the external reactivity.

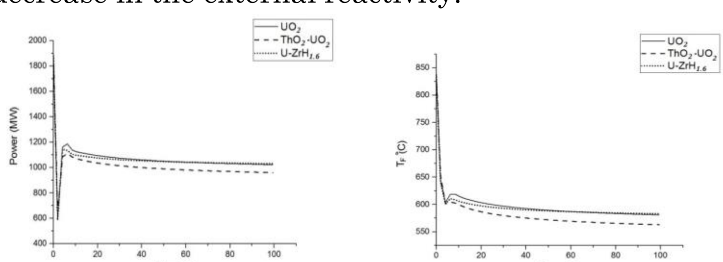

Figure (3): The transient response for UO_2, ThO_2-UO_2 and U-$ZrH_{1.6}$ fueled PWR system, for a 0.01 ΔK/K step decrease in the external reactivity.

b) Loss of coolant flow rate transient

Loss of flow rate accidents would be caused by failure of one or more pumps in the primary coolant system caused by a rupture of the primary coolant line, failure of a primary coolant pump seal, inadvertent opening of a pressure relief or safety valve, and so on. This results in increasing in the temperature of the coolant as shown in figure (4).

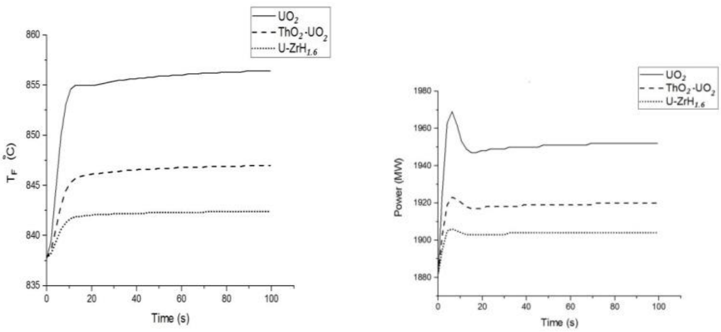

Figure (4): The transient response for UO_2, ThO_2-UO_2 and U-$ZrH_{1.6}$ fueled PWR system relative to a 10% step reduction in coolant flow rate

c) Inlet coolant temperature transient increase

The increase in inlet coolant temperature makes rapid decrease in the reactor power followed by increase in the reactor power and fuel temperature due to negative feedback reactivity. The increase in the coolant temperature is followed by a decrease in the reactor power and the fuel temperature as shown in figure (5).

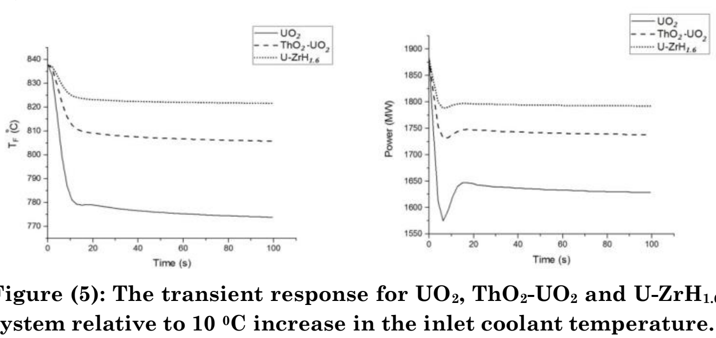

Figure (5): The transient response for UO_2, ThO_2-UO_2 and U-$ZrH_{1.6}$ fueled PWR system relative to 10 ^{0}C increase in the inlet coolant temperature.

d) Experimental simulation

From the experimental study and as shown in figure (6) it is observed that thermal neutron flux decreases as relative iron thickness increases which agrees with the fact that thermal neutron reaction cross-section is greater for Fe.

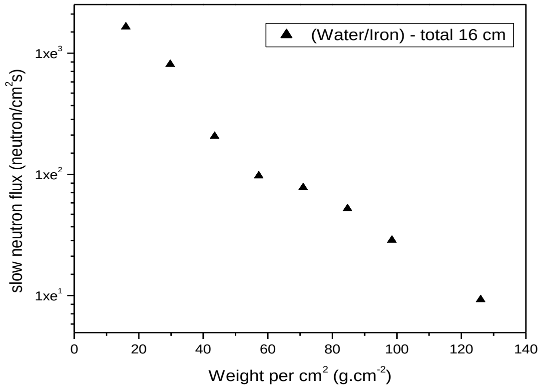

Figure (6): Measured slow neutron flux behind (Water/Iron) bi-layer assembly

5) Conclusion

Comparisons are carried out between the result of the three reactor models subject to the same type and value of perturbations, and the following conclusions are obtained:

PureUO$_2$, ThO$_2$-UO$_2$ and U-ZrH$_{1.6}$ fueled PWR reactors are stable at small reactivity change.

For the same reactivity change, the changes in both UO$_2$ and U-ZrH$_{1.6}$ fueled PWRs power and temperature of fuel and coolant are slightly similar and larger than those of mixed ThO$_2$-UO$_2$ fueled PWR, this can be interpreted as a result of large value of delayed neutron fraction of mixed ThO$_2$-UO$_2$ fueled PWR than that in pure UO$_2$ and hydride U-ZrH$_{1.6}$ fueled PWRs .The hydride fueled PWR shows more stability than other systems due to its larger negative temperature coefficient of reactivity.

From the experimental study; although, transmitted slow neutron flux for full range Fe is 9.5 times less than for full range water, the reactor design should include both layers in order to attenuate fast neutrons and gamma rays.

REFERENCES

1) Ethrington. Nuclear engineering handbook. McGraw Hill Book company, Inc., 1958.
2) Youssef, Akram. "Transient study of pressurized water reactor with different fuel types." M. Sc. Thesis., Cairo University, 2017.
3) Mohammed, Mghar, Abdelouahed, Chetaine and Abdelaziz, Darif. "Calculation of Kinetic Parameters of the Moroccan TRIGA Mark-II Reactor Using the Monte Carlo Code MCNP" Physics Department, Laboratory of nuclear physics, University Mohamed V, Faculty of Science, Rabat, Morocco, 2015.
4) Bjork, K. I. "Thorium Fuels for Light Water Reactors" Department of applied physics, Chalmers University of technology, Göteborg, Sweden, 2015.
5) Mahmoud, N. H. "Effect of steam generator type on the dynamic behavior of PWR system" Ph.D. Thesis, Cairo University, 1982.
6) Hend M. Saad, M. Aziz and H. M.Mansour. "Analysis of reactivity accident for control rods withdrawal the thermal reactor" International Journal of Recent advances in Physics (IJRAP) Vol.3, No.2, (May-2014).
7) Profio, A.E. "Radiation Shielding and Dosimetry" Awidely-Interscience Publication, a Division of John Wiley and Sons, New York, USA., 1979.
8) El-Sayed Abdo and Megahid, R.M. "Homogeneous and Multilayered Shields for Neutron and Gamma-Rays" Nuclear Research Centre, Atomic Energy Authority, Cairo, Egypt, Japanese journal of applied physics, Volume 40 (2001): pp. 2460 – 2464.