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Comparative Analysis of Water-Cooled Reactor Design Models and Gas-Cooled Reactor Design Models

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Abstract
To determined the most stable and probably the safest reactor between water-cooled reactor designs and gas-cooled reactor designs in terms of their coolant, Linear Regression Analysis is applied on four typical nuclear reactor design models, viz water-cooled reactor design I (WCRD I), water-cooled reactor design II (WCRD II), gas-cooled reactor design I (GCRD I) and gas-cooled reactor design II (GCRD II). Empirical expressions are obtained for WCRD I model, WCRD II model, GCRD I model and GCRD II model. The results of the statistical analyses on these four types of nuclear reactor models reveal that the GCRD II promises to be most stable. The implication of this research effort to Nigeria's nuclear power project is discussed.
Keywords: Linear Regression Analysis, Water-Cooled Reactor Design Models, Gas-Cooled Reactor Design Models, Safety Factor, Ŷ, Optimization, Stability Margin in Nuclear Power Reactor Designs

INTRODUCTION

One of the major components of nuclear power reactor design for effective and efficient operation is the coolant. A nuclear reactor is a source of intense heat which is generated through the exothermic fission reactions taking place inside the core. Therefore a coolant is necessary to ensure that this heat is taken away and utilized in a proper manner. However, experience have shown that Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs) could be susceptible to hydrogen buildup when core cooling fails [1] and eventually accidents, for example, the BWRs of Fukushima Daiichi Nuclear Power Reactor number 1, 2, 3 and 4 and the Chernobyl Nuclear Power Reactor number 4 [2]. The PWRs at Three Mile Island Unit 2 (TM1-2) near Pennsylvania in United States of America (USA)[3], the Davis-Besse Nuclear Power Station in Oak Harbor, Ohio, USA[4], the reactor plant shutdown at Palisades, Michigan, USA[5]. Others include three PWR at Oconee Nuclear Station located on Lake Keowee near Seneca, South Carolina, USA [6] and the PWR San Onofre Nuclear Generating Station (SONGS) located on the Pacific coast of California is an inoperative nuclear power plant, now planned to be decommissioned[7]. These are few cases of serious accidents record of BWRs and PWRs

There have been several report and analysis on the safety of these BWRs and PWRs taking into account the specific design features of these reactors, these include 'Status of thermohyraulic research in nuclear safety and new challenges' [8], 'Loss-of-Coolant Accidents (LOCAs) in BWRs and PWRs'[9],'Accident analysis for nuclear power plants with pressurized water reactors'[10] and Investigation of PWR accident situations' [11]. These accidents may perhaps be as a result of design concept process of BWR and PWR(which could involve novel technologies) that have inherent risk of failure in operation and were not well studied/understood [12].

Reactor stability is ability of a reactor to remain unchanged or to remain in operation over time under stated or reasonably expected conditions. Maintaining safety in the design and operation of nuclear power plants (NPPs) is a very important task under the conditions of a challenging environment, affected by the deregulated electricity market and implementation of risk informed regulations. Typically, safety margins are determined with use of computational tools for safety analysis [13].

A gas-cooled reactor (GCR) uses graphite as a neutron moderator, nitrogen, carbon dioxide, sodium, lead, hydrogen or helium as coolant. Although there are many other types of reactor cooled by gas, the terms *GCR* are particularly used to refer to this type of reactor. Carbon dioxide has been used in Magnox and AGR reactors. Helium is extremely inert both chemically and with respect to nuclear reactions but has a low heat capacity, necessitating rapid circulation. The design of a gas–cooled reactor such as Pebble Bed Modular Reactor (PBMR) is characterised by inherently safe features, which mean that no human error or equipment failure can cause an accident that would harm the public [14]. This type of reactor is claimed to be passively safe; that is, it removes the need for redundant, active safety systems [15]. Further literature review reveal that the design of PBMR's uses helium gas as coolant and the design is not as complex as other gas-cooled reactors, it is found to be simple, the PBMR's has lower power density, naturally safe fuel and would have no significant radiation release in accident unlike water-cooled reactors. The moderator used in these types of PBMR's is Graphite which offers the advantage of being stable under conditions of high radiation as well as high temperature or pressure nor when the velocity of the coolant reduced to zero, these are great advantage over water- cooled reactors. The gas-cooled reactors are any more or less economic or reliable than the water-cooled reactors. Why because technology is evolving as in the case of PBMR's.

In safety studies of High Temperature Gas-Cooled Reactor (HTGR), a failure of a standpipe at the top of the reactor vessel or a fuel loading pipe may be one of the most critical design- base accidents. Once the pipe rupture accident occurs, helium blows up through the breach immediately [16].

A loss-of-coolant accident (LOCA) is a mode of failure for a nuclear reactor [17]. LOCAs have occurred in light water and heavy water *reactors* as well as *gas cooled* and liquid metal *cooled* ones. Experience have shown that while nuclear power plants operate under strict safety codes and emergency procedures, there is no way to fully protect them from natural disasters, terrorist attacks or mere human error. Therefore, a study of a loss-of-coolant accident (LOCA) is very significant in the design and operation of any nuclear power reactor, since there have been several report analysis on the safety of the GCR's taking into account the specific design features of these reactors and loss-of-coolant accident in these reactors, these include 'Accident analysis for nuclear power plants with modular High temperature gas cooled reactors' [18], 'Nuclear Plant Risk Studies: Failing the Grade' and Overview Gas-cooled Reactor Problems' [19], 'Evaluation of system reliability with common-cause failures, by a pseudo-environments model'[20]. 'Reliability and Safety Analysis Methodology in the Nuclear Programs[21], 'nuclear power futures, costs and benefits'[22] and several reports analysis on the cost of failure on these GCR's, this include; 'A preliminary assessment of major energy accidents'[23].

Failure may be recognized by measures of risks which include performance, design fault, obsolete components, human errors and accident. These risks can be defined and quantified as the product of the probability of an occurrence of failure and a measure of the consequence of that failure. Since the objective of engineering is to design and build things to meet requirements, apart from cost implication, it is important to consider risk along with performance, and technology selections made during concept design. Engineering council guidance on risk for the engineering profession defined "Engineering Risk" as "the chance of incurring a loss or gain by investing in an engineering project". Similar definitions are given by Modarres [24], Molak. [25] and Blanchard [26], that risk is a measure of the potential loss occurred due to natural or human activities.

In this work, Ordinary Least Square (OLS) methodology, which is largely used in nuclear industry for modeling safety, is employed. Some related previous works on the application of regression analysis technique include: "Stochastic Modeling of Deterioration in Nuclear Power Plants Components"[27], "Regression Approach to a Simple Physics Problem" [28], "Japan raises nuclear crisis severity to highest level" [29]. Others are, "Advanced Power Plant Modeling with Applications to the Advanced Boiling Water Reactor and the Heat Exchanger"[30], 'Investigation of Fundamental Thermal- Hydraulic Phenomena in Advanced Gas-Cooled Reactors' [31], 'Quantitative functional failure analysis of a thermal-hydraulic passive system by means of bootstrapped Artificial Neural Networks'[32],'Fuel cycle studies on minor Actinide burning in gas cooled Fast reactors'[33], 'Counter-current flow limitations during hot leg injection in Gas-cooled reactors with a multiple *linear regression* model'[34] and "Optimization of The Stability Margin for Nuclear Power Reactor Design Models Using Regression Analyses Techniques"[35], where the effective of Regression Analyses Techniques 'RAT' in the Optimization of the Safety Factor in Nuclear Reactor Design Model was established.

This work provides a mathematical expression for predicting "Safety Factor", Ỳ, (dependent variables) given the values of independent variables or input parameters for a typical reactor design model. Furthermore, the mathematical expression can be used to determine the contribution of coolant flow rates (which is the independent variables) to the nuclear reactor stability, given the value of dependent variable. In conclusion a comparative analysis of the design models via the use of RAT was carried out.

THE RESEARCH OBJECTIVES:

To apply the linear regression technique on Boiling Water Reactors and Pressurized Water Reactors design models for the determination of their Safety Factor in terms of their coolant and to carry out a comparative analysis of the different design models.

The implication of this research effort to Nigeria nuclear power project will be discussed.

RESEARCH DESIGN/APPROACH

Theory and experience has shown that for nuclear power plants, coolants plays significant role in the safety of the reactor during operation in preventing reactor damage during accident. Hence, in this work, in assessment of some typical boiling/pressurized water reactor designs, the input parameter considered is the coolant (which is water in this case study). The typical nuclear reactor designs are coded as BWRD, and PWRD which stands for Boiling Water Reactor Design, Pressurized Water Reactor Design, Gas-Cooled Reactor Design I and Gas-Cooled Reactor Design II. The Tables 1 2,3 and 4, presented the values of design input parameters.

Table 1: Design Input Parameters of A Typical Boiling Water Reactor (BWR) Similar to Fukushima Daiichi damaged reactor 1-3, in Japan

Nos. of trial (j)	Safety factor	Coolant (water) flow rate in kg/s BWRD
1	1.30	100
2	1.35	200
3	1.40	300
4	1.45	400
5	1.50	500
6	1.55	600
7	1.60	650
8	1.65	700
9	1.70	800
10	1.75	900
11	1.80	1000
12	1.30	100
13	1.35	200

Source : [36]

Table 2: Design Input Parameters of a Typical Pressurized Water Reactor (PWR) Similar to Three Mile Island Unit 2 (TM1-2) damaged reactor near Pennsylvania in USA

Nos. of trial (j)	Safety factor	Coolant (water) flow rate in kg/s PWRD
1	1.30	100
2	1.40	200
3	1.40	300
4	1.50	400
5	1.45	500
6	1.60	600
7	1.55	700
8	1.70	800
9	1.72	900
10	1.55	1000
11	1.70	1100
12	1.72	1200
13	1.80	1300

Source: [37]

Table 3: Design Input Parameters of A Typical Gas-cooled Reactor (GCR)Similar to MIT in USA, Chinese PBMRs (HTR-10) and PBR in South Africa

Nos. of trial (j)	Safety factor	Coolant (gas) flow rate in kg/s GCRD I
1	1.30	100
2	1.35	105
3	1.40	110
4	1.45	115
5	1.50	120
6	1.55	125
7	1.60	130
8	1.65	135
9	1.71	140
10	1.73	145
11	1.75	150
12	1.77	155
13	1.79	160

Source: [38]

Table 4: Design Input Parameters of A Typical Gas-cooled Reactor (GCR)
Similar to the MIT in USA, Chinese PBMRs(HTR-10) and the PBM in South Africa

Nos. of trial (j)	Safety factor	Coolant (gas) flow rate in kg/s GCRD II
1	1.32	100
2	1.35	105
3	1.41	110
4	1.45	115
5	1.50	120
6	1.55	125
7	1.60	130
8	1.63	135
9	1.67	140
10	1.69	145
11	1.73	150
12	1.76	155
13	1.78	160
14	1.80	165

Source: [39]

RESULTS AND ANALYSES

1. Boiling Water Reactor Design (BWRD)

The results of the application of the linear regression analysis of the data in Table 1 and 2 for BWR are presented as follows: These regression analyses were carried out on two different water-cooled nuclear reactor designs(BWR and PWR) with the use of statistical software known as Number Cruncher Statistics Software (NCSS).

(i) Empirical Expression for Safety Factor, Ỳ

The data obtained in Table 1 which represents typical parameter for Boiling Water Reactor Design (BWRD) was modified in order to obtain the best fit for the model. The new conceptual design reactor model optimizes the performance of the Japan, Fukushima Daiichi Nuclear power reactors (1,2 and 3) the three old reactors dating from 1971-75.

The linear regression model equation to be solved is given by:

$$\grave{Y} = B_0 + B_1 X_j + e_j \quad (1)$$

where, B_0 is an intercept, B_1 is the slope and
X_j is the rate of flow of coolant and e_j = error or residual.
The model empirical expression for the Safety Factor Ỳ is obtained, as:

$$\grave{Y} = (1.3239) + (0.0004)*(X_j) + e_j \quad (2)$$

where,
1.3239 is an intercept, 0.0004 is a slope, X is the rate of flow of water coolant,
e = error or residual and j = 1,2,3,…,13.

- The equation (2) is the model empirical expression that could be applied to make predictions of the Safety Factor, Ỳ, on this type of (BWRD) model.

Note:

- The linear regression equation is a mathematical model describing the relationship between Safety Factor, Ỳ, and the coolant (input parameter, X).
- That the linear regression equation predicts Safety Factor based on their value. The value of Safety Factor depends on the values of design water coolant flow rate.
- The influence of all other variables on the value of Safety Factor is lumped into the residual (error - e_j).

The Linear Regression Plot Section on BWRD is shown in Figure 1

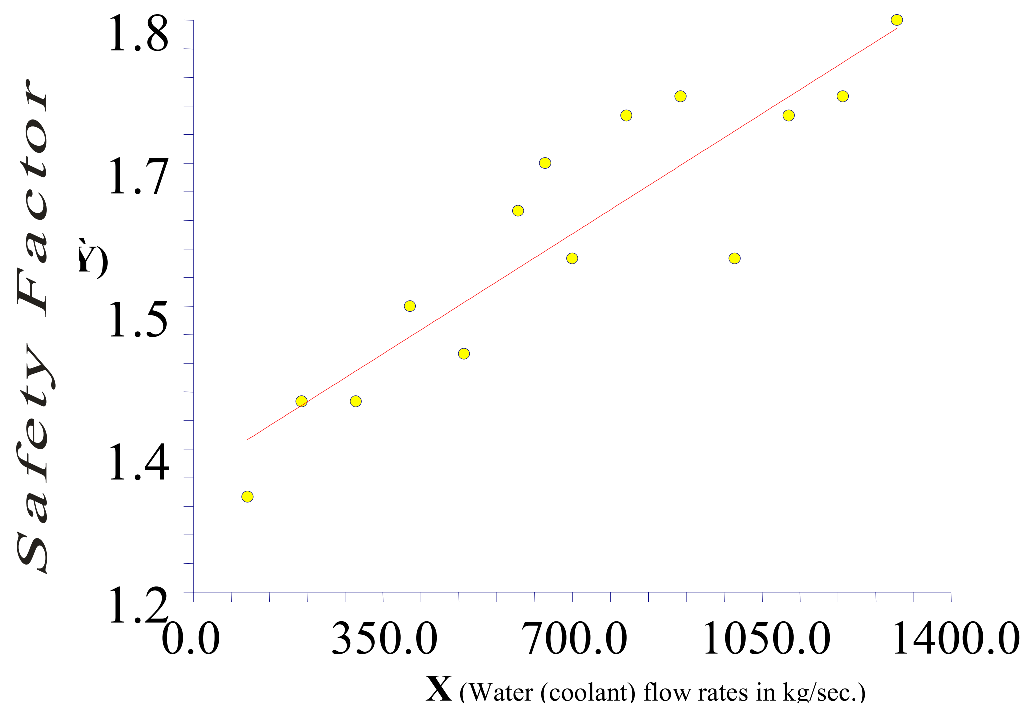

Figure 1: Safety factor (Ỳ) a function of water (coolant) flow rate (X)

The Figures 1 shows the relationship between Safety Factor, Ỳ and the Water (coolant) flow rates, X. The straight line shows that there is a linear relationship and the closeness of the points to the line indicates that the relationship is strong.

Next is the summary of the **F** -test result on BWRD as shown in Table 5.

(ii) F-test Result

The F-test result on BWRD is shown in Table 5

Table 5: Summary of F-test Statistical Data on BWRD

Parameter	**Value**
Dependent Variable	Ỳ
Independent Variable	X
Frequency Variable	None
Weight Variable	None
Intercept(B_0)	1.3239
Slope(B_1)	0.0004
R^2	**0.8136**
Correlation	**0.9021**
Mean Square Error	**4.496755×10^{-3}**
Coefficient of Variation	0.0426
Square Root of MSE	6.705785×10^{-2}

- The value of correlation at 0.9021 shows that the model is very good and could be of significant practical application, as the relationship between safety factor and coolant is strong.
- The value 4.496755×10^{-3} for the MSE indicates that the error e_j is minimized at optimal.
- The R^2 value of 0.8136 indicates that 81.36% of the variation in Ỳ (Safety Factor) would be accounted for by the water coolant flow rate, X, for PWRD II, this proves that the model is valid and good.

Note:

The correlation coefficient, or simply the correlation, is an index that ranges from -1 to 1.

When the value is near zero, there is no linear relationship. As the correlation gets closer to plus or minus one, the relationship is stronger. A value of one (or negative one) indicates a perfect linear relationship between two variables.

2. Pressurized Water Reactor Design (PWRD)

The result of the experiment on PWRD similar to the Three Mile Island Unit 2 (TM1-2) reactor which was severely damaged and similar to the shutdown *PWRs* at Palisades, Michigan, USA and Nuclear Power Station in Oak Harbor, Ohio, USA.

(iv) Empirical Expression for Safety Factor, Ỳ

The data obtained in Table 2 which represents typical parameter for Pressurized Water Reactor Design (PWRD) was modified in order to obtain the best fit for the model. The new conceptual design reactor model optimizes the performance of the accident PWR at Davis-Besse Nuclear Power Station in Oak Harbor, Ohio, USA and the shutdown *PWR* at Palisades, Michigan, USA.

The model empirical expression for the Safety Factor Ỳ is obtained, as:

$$\grave{Y} = (1.3150) + (0.0004)*(X_j) + e_j \quad (3)$$

Where, 1.3150 is an intercept, 0.0004 is a slope, X is the rate of flow of water coolant,
e = error or residual and j = 1,2,3,…,13.

The equation (3) is the model empirical expression that could be applied to make predictions of the Safety Factor, Ỳ, on this type of (PWRD) model.

The Linear Regression Plot Section on PWRD is shown in Figure 2

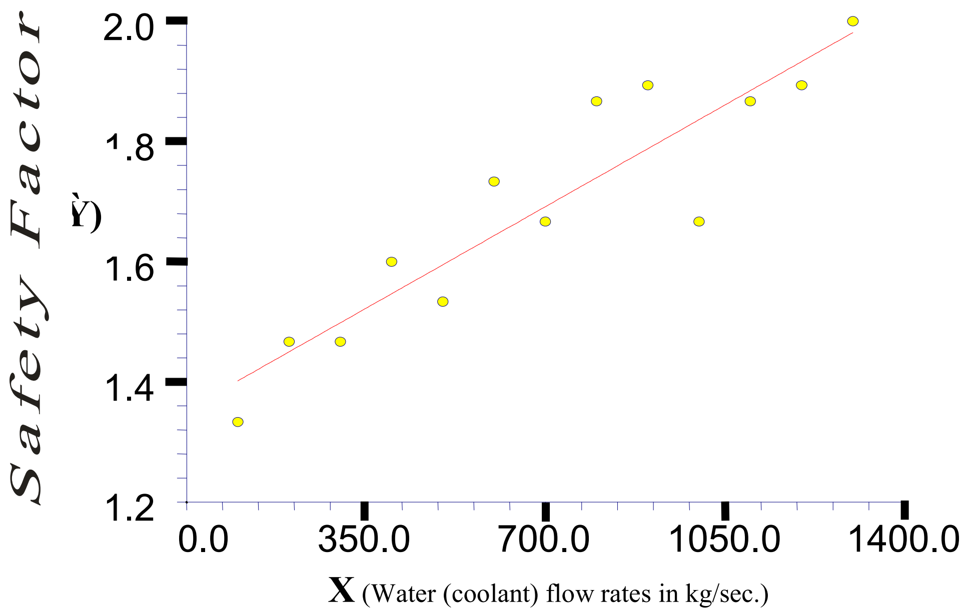

Figure 2. Safety Factor (Ỳ) as a function of water (coolant) flow rate (X)

The Figure 2 shows the relationship between Safety Factor, Ỳ and the Water (coolant) flow rates, X.

The straight line shows that there is a linear relationship and the closeness of the points to the line indicates that the relationship is strong.

(ii) F-test Result

The F-test result on PWRD is shown in Table 6

Table 6: Summary of F-test Statistical Data on PWRD

Parameter	Value
Dependent Variable	Ỳ
Independent Variable	X
Frequency Variable	None
Weight Variable	None
Intercept (B_0)	-3.7293
Slope (B_1)	0.0760
R^2	0.9766
Correlation	0.9882
Mean Square Error (MSE)	3.036871×10^{-3}
Coefficient of Variation	0.0121
Square Root of MSE	1.742662×10^{-1}

- ❖ The value of correlation at 0.9882 shows that the model is very good and could be of significant practical application, as the relationship between safety factor and coolant is strong.
- ❖ The value 3.036871×10^{-3} for the MSE indicates that the error e_j is minimized at optimal.
- ❖ The R^2 value of 0.9766 indicates that 97.66% of the variation in Ỳ (Safety Factor) would be accounted for by the water coolant flow rate, X, for PWRD II, this proves that the model is valid and good.

3. Gas-Cooled Reactor Design I (GCRD I)

We also considered sample from Gas-cooled reactor (GCR) in GCRD I, by performing experiment on GCRD I taken input parameters from reactor Similar to the reactor model of the MIT in USA, Chinese PBMRs (HTR-10) and PBR in South Africa Design Input Parameter Nuclear Power Plants. The data was modified in other to obtain the best fit for the model.

(i) Empirical Expression for Safety Factor, Ỳ

The data obtained in Table 2 which represents typical parameter for Gas-Cooled Reactor Design I (GCRD I) was modified in order to obtain the best fit for the model. The new conceptual design reactor model optimizes the performance of the Design Input Parameters of Typical Pebble Bed Modular Reactor (PBMR) Specifications.
The model empirical expression for the Safety Factor Ỳ is obtained, as:

$$\grave{Y} = (0.5732) + (0.0077)*(X_j) + e_j \qquad (3)$$

where, 0.5732 is an intercept, 0.0077 is a slope, X is the rate of flow of gas coolant and,
e = error or residual and j = 1,2,3,…,13.

- ➢ The equation (3) is the model empirical expression that could be applied to make predictions of the Safety Factor, Ỳ, on this type of (GCRD I) model.

The Linear Regression Plot Section on GCRD I is shown in Figure 2.

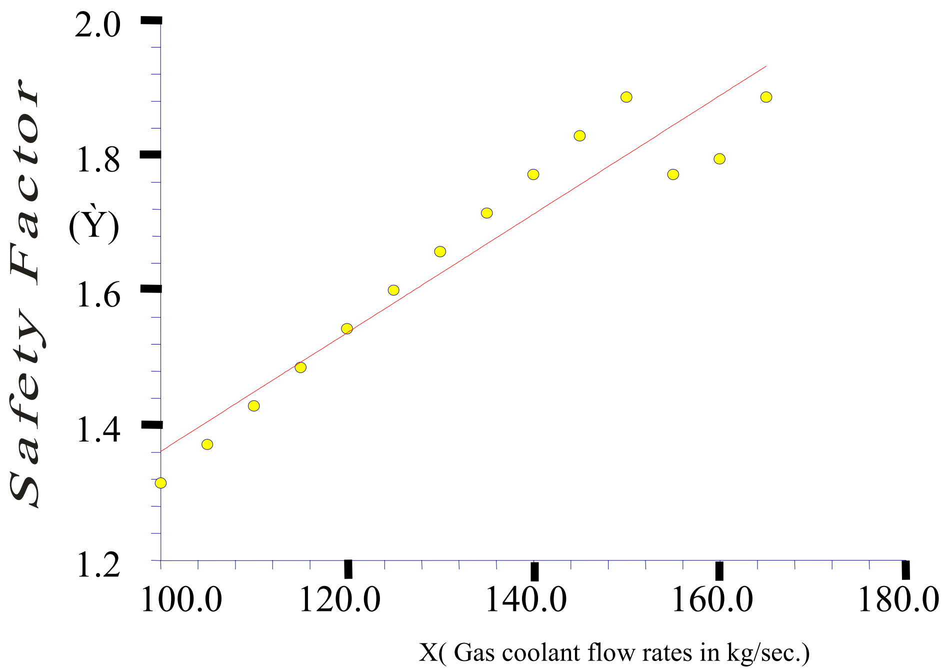

Figure 2: Safety factor (Ỳ) a function of gas (coolant) flow rate (X)

Next is the summary of the **F** -test result on GCRD I as shown in Table 5.

(ii) F-test Result

The F-test result on GCRD I is shown in Table 5

Table 7: Summary of F-test Statistical Data on GCRD I

Parameter	**Value**
Dependent Variable	Ỳ
Independent Variable	X
Frequency Variable	None
Weight Variable	None
Intercept (B_0)	0.5732
Slope (B_1)	0.0077
R^2	0.9571
Correlation	0.9877
Mean Square Error (MSE)	2.525604 x 10^{-3}
Coefficient of Variation	0.2109
Square Root of MSE	5.025539 x 10^{-2}

- The value of correlation at 0.9877 shows that the model is very good and could be of significant practical application.
- The value 2.525604 x 10^{-3} for the MSE indicates that the error e_j is minimized at optimal.
- The R^2 value of 0.9571 indicates that 95.71% of the variation in Ỳ would be accounted for by the gas coolant flow rate, X, this further proves that the model GCRD II is good and valid.

3. Gas-Cooled Reactor Design II (GCRD II)

We also considered sample from Gas-cooled reactor (GCR) in GCRD II, by performing experiment on GCRD II taken input parameters from reactor Similar to the MIT in USA, Chinese PBMRs (HTR-10) and PBR in South Africa Design Input Parameter Nuclear Power Plants. The data was modified in other to obtain the best fit for the model.

(i) Empirical Expression for Safety Factor, Ỳ

The data obtained in Table 3 which represents typical parameter for Gas-Cooled Reactor Design II (GCRD II) was modified in order to obtain the best fit for the model. The new conceptual design reactor model optimizes the performance of the Design Input Parameters of the similar to the Massachusetts Institutes of Technology (MIT) Pebble Bed Reactor and the Pebble Bed Nuclear Power Plant in South Africa.

The model empirical expression for the Safety Factor Ỳ is obtained, as:

$$\grave{Y} = (-139.3887) + (110.9289)*(X_j) + e_j \quad (4)$$

where, -139.3887 is an intercept, 110.9289 is a slope, X is the rate of flow of gas coolant, e = error or residual and j = 1,2,3,…,13.

- The equation (4) is the model empirical expression that could be applied to make predictions of the Safety Factor, Ỳ, on this type of (GCRD II) model.

The Linear Regression Plot Section on GCRD II is shown in Figure 2.

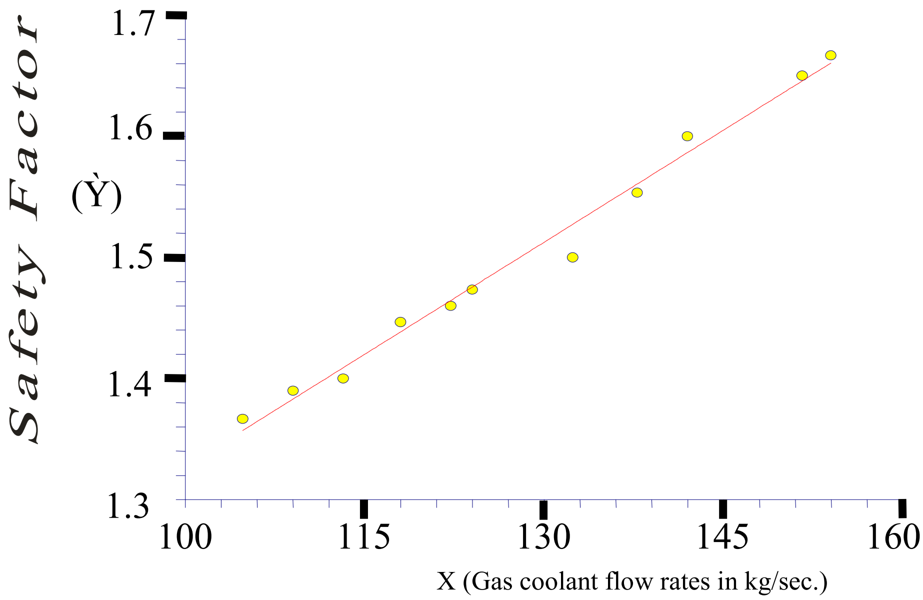

Figure 2: Safety factor (Ỳ) a function of gas (coolant) flow rate (X)

The plot in Figures 2 shows the relationship between Safety Factor, Ỳ and the Gas (coolant) flow rates, X. The straight line shows that there is a linear relationship and the closeness of the points to the line indicates that the relationship is strong.

Next is the summary of the **F** -test result on GCRD II as shown in Table 6.

(ii) F-test Result

The F-test result on GCRD II is shown in Table 6

Table 8: Summary of F-test Statistical Data on GCRD II

Parameter	**Value**
Dependent Variable	Ỳ
Independent Variable	X
Frequency Variable	None
Weight Variable	None
Intercept (B_0)	-139.3887
Slope (B_1)	110.9289
R^2	0.9875
Correlation	0.9938
Mean Square Error (MSE)	1.53471×10^{-3}
Coefficient of Variation	0.0610
Square Root of MSE	4.070336×10^{-2}

- ❖ The value of correlation at 0.9938 shows that the model is very good and could be of significant practical application.
- ❖ The value 1.53471×10^{-3} for MSE indicates that the error e_j is minimized at optimal.
- ❖ The R^2 indicates that 98.75% of the variation in Ỳ would be accounted for by the gas coolant flow rate, X, for GCRD II, this further proves that the model GCRD II is valid and good.

3. SUMMARY/CONCLUSION

This work focus on two typical boiling water reactors designs viz BWRD and PWRD and two typical pressurized water reactors designs viz GCRD I and GCRD II.
The empirical expressions for the optimization of nuclear reactor Safety Factor (Ỳ) as functions of coolant flow rate for boiling water nuclear reactor design models (BWNRDM), pressurized water nuclear reactor design models (PWNRDM) and gas-cooled nuclear reactor design models (GCNRDM) are obtained as**:**

(i) $\grave{Y} = (1.3239) + (0.0004)*(X_j) + e_j$, for BWRD
(ii) $\grave{Y} = (1.3150) + (0.0004)*(X_j) + e_j$, for PWRD
(iii) $\grave{Y} = (0.5732) + (0.0077)*(X_j) + e_j$, for GCRD I
(iv) $\grave{Y} = (-139.3887) + (110.9289)*(X_j) + e_j$, for GCRD II

These are the model equations that could be applied to make predictions of the safety factor, Ỳ, on these types of reactor design models.

The empirical expressions may also be used for the calculation of the Ỳ of the reactors which in turn is a measure of the reactor's stability.
Also, the empirical formulae derived can be used to determine the contribution of coolant to the stability of the reactor. The Table 9 highlights the summary results on coolant effects on water reactors.

Table 9. Summary Results on Coolant Effects on Water Reactors

Types of Nuclear Power Reactor Design Model	**Correlation** values between Safety factor and Coolant	**R^2** Indicating goodness-of -fit	**Mean Square Error** values at which error is minimized at optimal
Boiling Water Reactors Design			
BWRD	0.9021	0.8136	4.496755×10^{-3}
Pressurized Water Reactors Design			
PWRD	0.9176	0.8421	4.068482×10^{-3}
Gas-Cooled Reactors Designs			
GCNRD II	0.9877	0.9571	2.525604×10^{-3}
GCNRD III	0.9938	0.9875	1.53471×10^{-3}

❖ Figure 5 is a graphical representation comparing the correlation values of BWRD PWRD , GCRD I and GCRD II. It is obvious that the gas-cooled reactor design II (GCRD II), is most stable in terms of correlation. It is also understandable that GCRD II with correlation value of 0.9938 is most optimized than any other model.

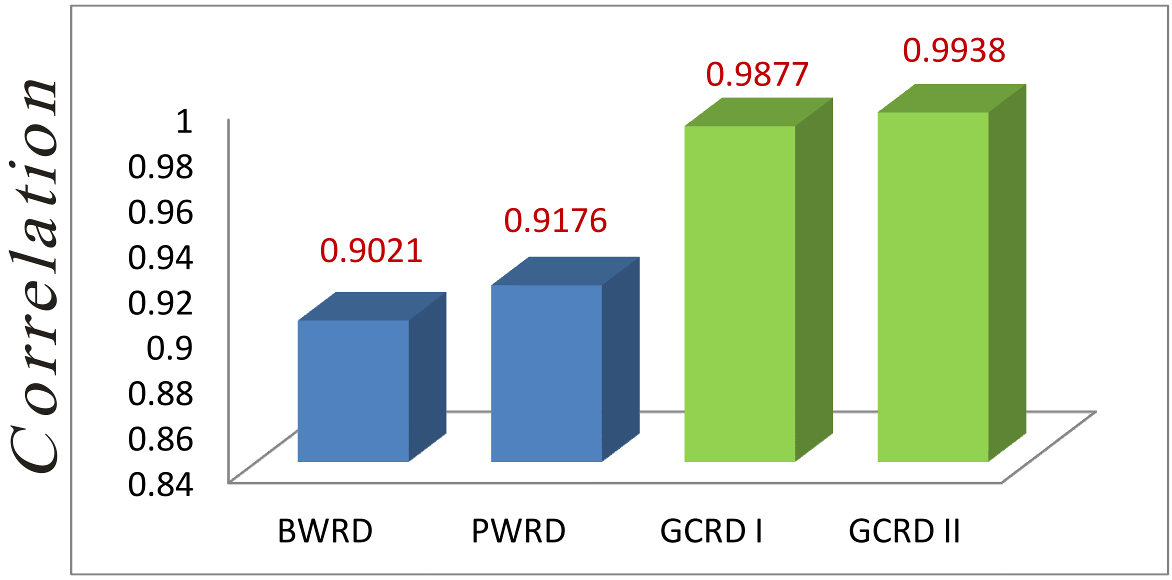

Figure 5. Nuclear Reactor Design Models

❖ In Figure 6 the bar charts reveal that, BWRD, PWRD and GCRD I have lower value of coefficient of determination (R^2), compared to GCRD II. It is also clear from the figure that the values of PWRD II are better optimized than BWRD, PWRD and GCRD I. Therefore, GCRD II with coefficient of determination value of 0.9875 could be said to promise the best stability and possibly the safest when compared with other design models.

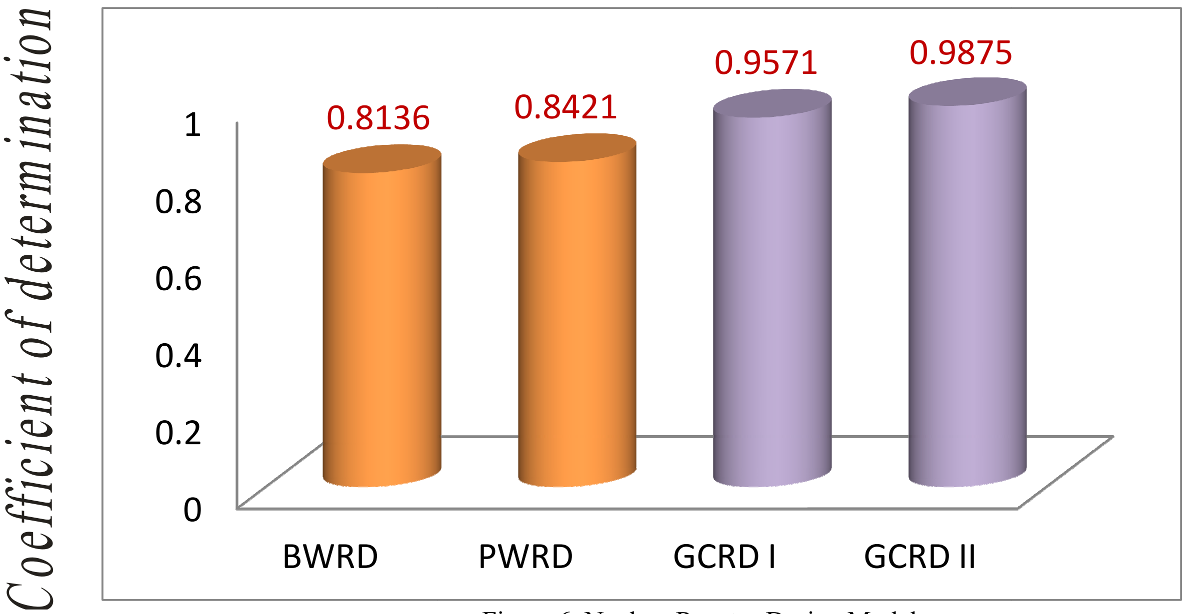

Figure 6. Nuclear Reactor Design Models

❖ In Figure 7, charts reveal that BWRD, PWRD and GCRD I have higher values of the mean square of error (4.496755), (4.068482), (2.525604) respectively than GCRD II with mean square of error (1.53471).
Therefore, since GCRD II have minimal mean square of error (1.53471) it indicates that PWRD models may promises more safety features than BWRD, PWRD and GCRD I models.

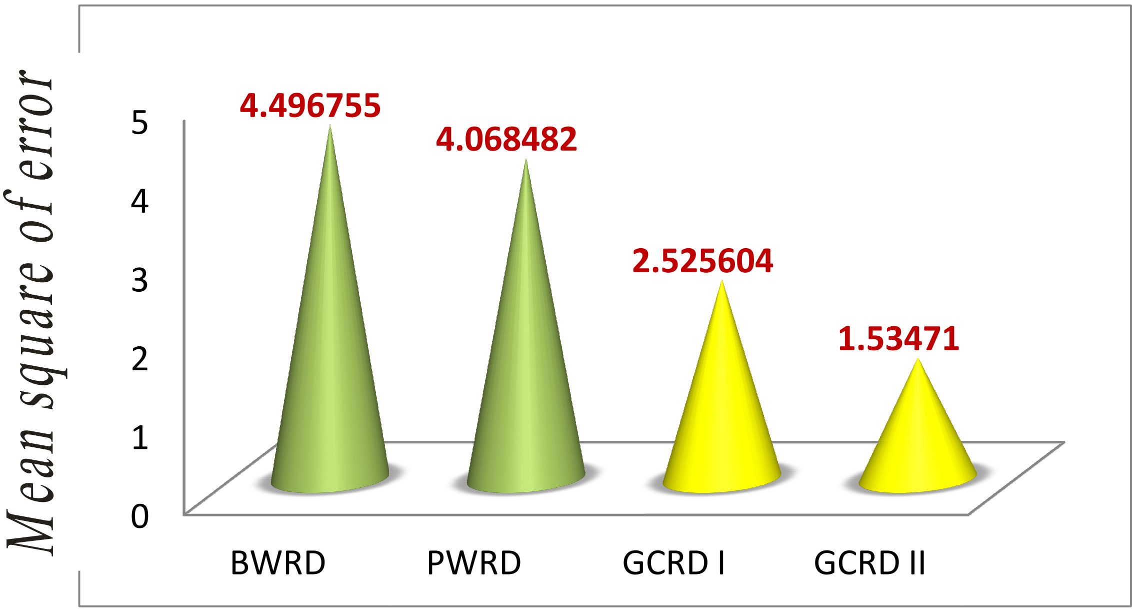

Figure 7. Nuclear Reactor Design Models

In conclusion Linear regression analysis is applied on four typical Water-Cooled Nuclear Reactor design models, viz Boiling Water Reactor Design (BWRD), Pressurized Water Reactor Design (PWRD), Gas-Cooled Reactor Design I (GCRD I) and Gas-Cooled Reactor Design II (GCRD II). Empirical expressions are obtained for BWRD model, PWRD model, GCRD I model and GCRD II model. The results of the statistical analyses on these four types of nuclear reactor models reveal that the GCRD II promises to be most stable.
In this method of regression analysis the safety margin prediction of up to 0.062% has been validated for reactor design models on pressurized water reactor as an advantage over the current 5.1% challenging problem for plant engineers to predict the safety margin limit. According to Xianxun Yuan (2007, P49) in "Stochastic Modeling of Deterioration in Nuclear Power Plants Components" a challenging problem of plant engineers is to predict the end of life of a system safety margin up to 5.1% validation. However, the current design limits for various reactors Safety Factor in a nuclear power plant, defined by the relative increase and decrease in the parametric range at a chosen operating point from its original value, varies from station to station.
Finally, the proposed new method for reactor design concept with the use of coolant as input parameter and the discoveries on water coolant on safety factor shall provides a good, novel approach and method for multi-objective decision-making based on six dissimilar objectives attributes**:** evolving technology, effectiveness, efficiency, cost, safety and failure.
It is therefore suggested that for countries wishing to include nuclear energy for the generation of electricity, like Nigeria, the parameters of the selected nuclear reactor should undergo analysis via RAT for optimization and choice.

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