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INVESTIGATING THE EFFECT OF LOSS-OF-PRESSURE-CONTROL ON THE STABILITY OF WATER-COOLED REACTOR DESIGN MODELS

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ABSTRACT: *To investigate loss-of-pressure-control on the stability of a typical water-cooled reactor design models during operation in terms of normal applicable pressure within reactor core and abnormal applicable pressure within reactor core. Linear Regression Analysis Techniques was applied on a typical water-cooled nuclear reactor design models, viz Water-Cooled Reactor Design with Normal Pressure (WCRD NP) flow rate within the reactor and Water-Cooled Reactor Design with Abnormal Pressure (WCRD AP) flow rate within the reactor. Empirical expressions are obtained for WCRD NP model and WCRD AP model. The results of the statistical analyses on these two types of nuclear reactor models reveals that the WCRD NP promises to be more stable than WCRD AP. The implication of this research effort to Nigeria's nuclear power project drive*.

KEYWORDS: Linear Regression Analysis, Water-Cooled Reactor Design Model with Normal Pressure and Abnormal PRESSURE, Safety Factor, Y, Optimization, Stability Margin in Nuclear Power Reactor Designs.

INTRODUCTION

Most commercial types of nuclear reactor use a pressure vessel to maintain pressure in the reactor plant. This is necessary in a pressurized water reactor (PWR) to prevent boiling in the core, which could lead to a nuclear melt-down [1]. Most water-cooled reactors could be susceptible to pressure or hydrogen buildup when core cooling fails and eventually accidents, for example, the boiling water reactors (BWRs) of Fukushima Daiichi Nuclear Power Reactor number 1, 2, 3 and 4 and the BWR of Chernobyl Nuclear Power Reactor number 4. The PWR at Three Mile Island nuclear power plant in USA melt-down and was destroyed due to cooling malfunction that resulted from pressure-built-up problem [2].

In a loss-of-pressure-control accident, the pressure of the confined coolant falls below specification without the means to restore it. This may form an insulating 'bubble' of steam surrounding the fuel assemblies (for pressurized water reactors) and in others may reduce the heat transfer efficiency (when using an inert gas as a coolant).
Many failures in a reactor plant or its supporting auxiliaries could cause a loss-of- pressure-control, including**:**

- Inadvertent isolation of the pressurizing vessel from the reactor plant, via the closing of an isolation valve or mechanically clogged piping.
- Failure of either the spray nozzles (failing open would inhibit raising pressure as the relatively cool spray collapses the pressurizer vessel bubble) or the heaters of the pressurizing system.

- Thermal Stratification of the liquid portion of the pressurizer.
- A rupture in the pressurizer vessel, which would also be a loss-of-coolant accident.

There have been several reports and analysis on the safety of these water-cooled reactors such as in a loss-of-pressure-control accident we have, ‘Analysis of Severe Accidents in Pressurized Heavy Water Reactors [3], and “Accident Analysis for Nuclear Power Plants with Pressurized Heavy Water Reactors”. In a loss-of-coolant accidents we have BWRs and PWRs taking into account the specific design features of these reactors, these include “Influence of Turbulence on the Deflagrative Flame Propagation in Lean Premixed Hydrogen Air Mixtures”[4], ‘Loss-of-Coolant Accidents (LOCAs) in BWRs and PWRs’[5],"Fuel Failures in Water Cooled Reactors”[6]. Others are ‘Accident Analysis for Nuclear Power Plants with Pressurized Water Reactors”, “Digital Instrumentation and Control Failure Events Derivation and Analysis for Advanced Boiling Water Reactor”[7] and related work on pressure effect is “A general formula for reactant conversion over a single catalyst particle in TAP pulse experiments”[8]. Furthermore, inclusive are recent studies on “Nuclear, Plasma and Radiation Science”[9]. 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.

Failure may be recognized by measures of risks which include performance, design fault, obsolete components, wrong application, 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, Molak and Blanchard 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: “Optimization of The Stability Margin for Nuclear Power Reactor Design Models Using Regression Analyses Techniques”[10],“Modeling and Simulation of an Industrial Trickle-Bed Reactor for Benzene Hydrogeneration**:** Model Validation against Plant Data”[11],“Stochastic Modeling of Deterioration in Nuclear Power Plants Components”[12],“Regression Approach to a Simple Physics Problem”, "Japan raises nuclear crisis severity to highest level". Others are, “Advanced Power Plant Modeling with Applications to the Advanced Boiling Water Reactor and the Heat Exchanger”, ‘Investigation of Fundamental Thermal- Hydraulic Phenomena in Advanced Gas-Cooled Reactors’, ‘Quantitative functional failure analysis of a thermal-hydraulic passive system by means of bootstrapped Artificial Neural Networks’[13], these are materials where the effective used of Regression Analyses Techniques ‘RAT’ in the Optimization of the Safety in Nuclear Reactor Design Model has been established.

The Research Objectives

The purpose of this work is to assist countries wishing to include nuclear energy for the generation of electricity, like Nigeria, to secure a reactor that is better and safe. The achievement here is to make worldwide contribution to knowledge. The studies intended to provide guidance in developing practical catalytic materials for power generation reactor and to help researchers make appropriate recommendation for Nigeria nuclear energy proposition as one of the solutions to Nigeria energy crisis.

RESEARCH DESIGN/APPROACH

Theory and experiment has shown that for a water-cooled reactor, heat within reactor core plays significant role in the safety of the reactor during operation in preventing reactor damage during accident. Hence, in this work, an assessment of the high raise in pressure in the reactor is considered of a typical boiling/pressurized water reactor designs. More specifically, the studies will concentrate on technical factors that limit reactor stability and achievement of higher efficiency due to loss-of-pressure-control interference, such as the abnormal pressure mechanical interaction. Detailed investigations of loss-of-pressure-control behaviour under reactor accident conditions are also included.

The research approach involves adjusting the parameters of a model function to best fit a data set. A simple data set consists of *n* points (data pairs) (x_i, y_i), *i* = 1, ..., *n*, where x_i is an independent variable and y_i is a dependent variable whose value is found by observation. The model function has the form $f(x,\beta)$, where the *m* adjustable parameters are held in the vector β. The goal is to find the parameter values for the model which "best" fits the data. The least squares method finds its optimum when the sum, *S*, of squared residuals

$$S = \sum_{i=1}^{n} r_i^2$$

is a minimum.

The Tables 1 and 2, presented the values of design pressure design input parameters.

Table 1: Input data for safety factor against normal and abnormal pressure in a typical BWR similar to Fukushima Daiichi damaged reactor 1-4, in Japan and similar to BWR at Chernobyl Nuclear Power Reactor no. 4 in Russia accident meltdown and similar to PWR at Three Mile Island Unit 2 (TM1-2) damaged reactor near Pennsylvania in USA.
Source [14]

Nos. of trial (j)	Safety Factor	Max. scale pressure(bar)	Min. scale pressure(bar)
1	1.30	60	50
2	1.42	120	60
3	1. 45	180	80
4	1.50	240	90
5	1.55	300	100
6	1.60	360	110
7	1.65	420	120
8	1.70	480	130
9	1.73	540	140
10	1.75	600	150

Pressure reading at zero level in the nuclear power plant reactor			
Nos. of trial (j)	**Safety Factor**	**Max. scale pressure(bar)**	**Min. scale pressure(bar)**
1	1.30	150	70
2	1.42	160	80
3	1. 45	180	90
4	1.50	0	0
5	1.55	220	110
6	1.60	240	120
7	1.65	260	130
8	1.70	280	140
9	1.73	300	150
10	1.75	400	160

Table 2: Data for Safety Factor against Normal and Abnormal Pressure Reading at Zero in a Typical Water-Cooled Reactor**. Source** [15]

Pressure reading at zero level in the nuclear power plant reactor			
Nos. of trial (j)	**Safety Factor**	**Max. scale pressure(bar)**	**Min. scale pressure(bar)**
1	1.30	150	70
2	1.42	160	80
3	1. 45	180	90
4	1.50	0	0
5	1.55	220	110
6	1.60	240	120
7	1.65	260	130
8	1.70	280	140
9	1.73	300	150
10	1.75	400	160

RESULTS AND ANALYSES

1. Water-Cooled Reactor Design (WCRD)

For each of these different designs, linear regression analysis technique was applied using software, NCSS. The results obtained in form of model equations for each different design were analysed and used to determine the reactor stability.

(i) Empirical Expression for Safety Factor, $\grave{Y}$

Investigating the effect of over-pressurization to the Stability and Safety of the nuclear reactor during operation. The data obtained in Table 1 which represents a typical parameter for Water-Cooled Reactor Design was modified in order to obtain the best fit for the model. The new conceptual design reactor model optimizes the performance of the Fukushima Daiichi damaged reactor 1-4 in Japan, Chernobyl Nuclear Power Reactor no. 4 accident melt-down in Russia and Three Mile Island Unit 2 damaged reactor near Pennsylvania in USA.

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

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

where,

B_0 is an intercept, B_1 is the slope, X_j is the rate of flow of pressure

e_j = error or residual and j = 1,2,3,…,k

Empirical Expression for Safety Factor, Ỳ for Normal Pressure Reading

The model empirical expression is the equation of the straight line relating pressure maximum design limit in the reactor and the pressure minimum design limit in the reactor as a measure of safety, estimated as**:**

$$\grave{Y} = (10.3887) + (1.2608) *(X_j) + e_j \quad (2)$$

Equation (2) is the desire estimated model or predicted

where,

Ỳ = pressure maximum design limit,
10.3887 = intercept,
1.2608 = slope,
X = the value of input parameter
e = error or residual and j = 1,2,3,…,10

Empirical Expression for Safety Factor, Ỳ for Abnormal (High) Pressure Reading

The model empirical expression is the equation of the straight line relating pressure maximum design limit in the reactor and the pressure minimum design limit in the reactor as a measure of safety, estimated as**:**

$$\grave{Y} = (273.9054) + (0.7650)*(X_j) + e_j \quad (3)$$

Equation (3) is the estimated model or predicted for Abnormal (High) Pressure

where,

Ỳ = pressure maximum design limit, 273.9054= intercept, 0.7650 = slope,
X = the value of input parameter,
e = error or residual and j = 1,2,3,…,10

Empirical Expression for Safety Factor, Ỳ for Abnormal (Zero) Pressure Reading

The model empirical expression is the equation of the straight line relating pressure maximum design limit in the reactor and the pressure minimum design limit in the reactor as a measure of safety, estimated as**:**

$$\grave{Y} = (0.0000) + (1.9829)*(X_j) + e_j \quad (4)$$

Equation (4) is the estimated model or predicted for Abnormal (Zero) Pressure

where,

Ỳ = pressure maximum design limit, 0.0000 = intercept, 1.9829 = slope,

X = the value of input parameter,
e = error or residual and j = 1,2,3,…,10

The Figures 1, 2 and 3 shows the linear regression plot on a normal/high/zero pressure reading effect on reactor respectively.

(ii) Comparison of a Normal / Abnormal / Zero Pressure (bar) reading in the operating reactor of a nuclear power plant (Figure 1, Figure 2 and Figure 3) respectively

In Figure 2 the straight line shows that there is a linear relationship between the operating reactor and pressure, but the scattering of the regression points beginning from the middle at point 90 (bar) to the top of the line where the maximum pressure design limit reading is exceeded at 200 (bar) and minimum pressure design limit of is exceeded at 120 (bar) this indicates that the relationship is Not Strong i.e. the relationship could be very weak. Also the regression points in Figure 2 are divergence from the regression line instead of convergence along the regression line as in the case of Figure 1. In Figure 3 the straight line shows that there is a linear relationship, but as the points goes to zero reading this indicates that the relationship is very weak.
Therefore, in Figures 2 and 3 there is likehood of an accident in this kind of operating reactor. Accident can happened especially when the design limit of pressure reading is exceeded.

The Figure 1 highlights normal pressure reading in the operating reactor before loss-of-pressure-control occurred.

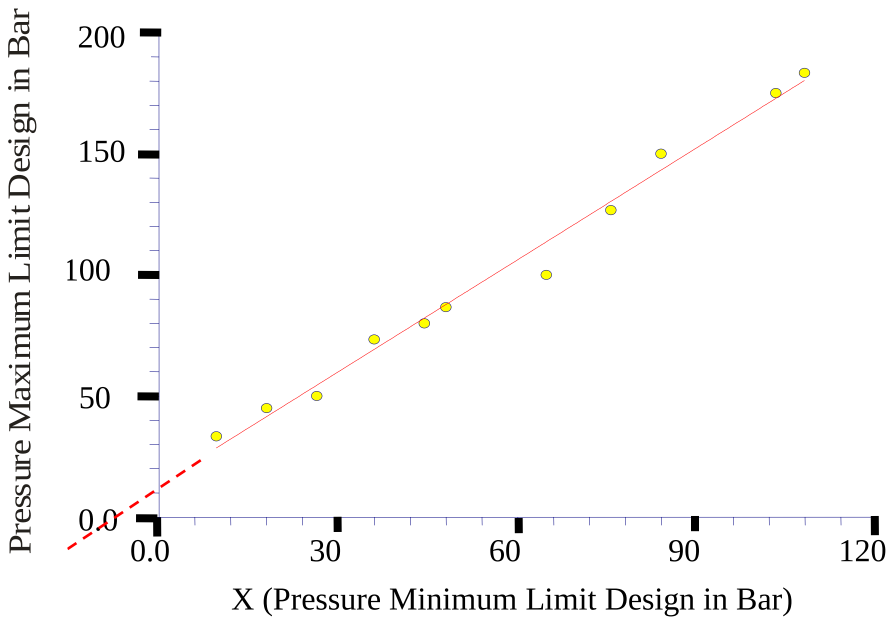

Figure 1: Normal pressure reading in the operating reactor

The Figure 2 is an illustration of high or abnormal pressure reading (bar) in the operating reactor as loss-of-pressure-control occurred.

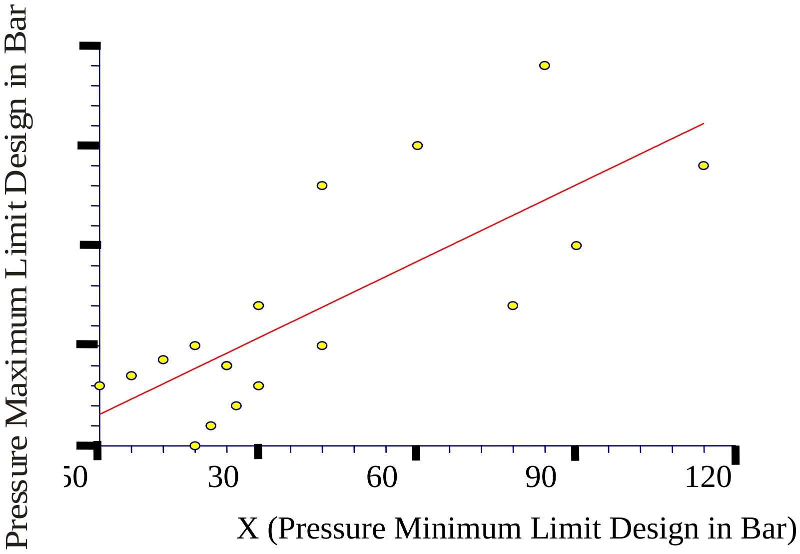

Figure 2: High Pressure in the operating reactor

The Figure 3 present abnormal pressure reading in the operating reactor where the reading suddenly fall to zero level as loss-of-pressure-control occurred.

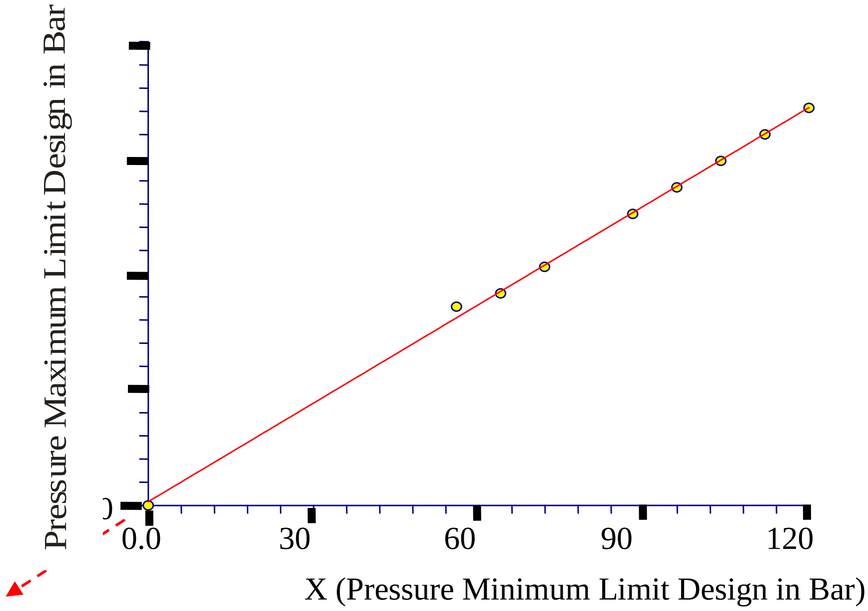

Figure 3: Abnormal pressure reading (zero) in the operating reactor

(iii) F-test Result

Table 3: Comparison of the Summary of F-test Statistical Data on Application of Normal Pressure (NP) and Abnormal Pressure (AP) on WCRD

Parameter	Value of Pressure Reading (Normal)	Value of Pressure Reading (High)
Dependent Variable	Pressure Max. Design Limit	Pressure Max. Design Limit
Independent Variable	Pressure Man. Design Limit	Pressure Man. Design Limit
Intercept(B_0)	10.3887	273.9054
Slope(B_1)	1.2608	0.7650
R-Squared	0.9683	0.5852
Correlation	0.9012	0.5651
Mean Square Error (MSE)	14927.28 x 10^{-3}	116351.9 x 10^{3}
Coefficient of Variation	0.1196	0.3050
Square Root of MSE	1.18855	355.4602

F-test Result

Table 4: Comparison/Analysis of Summary of F-test Statistical Data on Pressure Reading at Normal / Abnormal (Zero) on WCRD

Parameter	Value of Pressure Reading (Normal)	Value of Pressure Reading (Zero)
Dependent Variable	Pressure Max. Design Limit	Pressure Max. Design Limit
Independent Variable	Pressure Man. Design Limit	Pressure Man. Design Limit
Intercept(B_0)	10.3887	0.0000
Slope(B_1)	1.2608	1.9829
R-Squared	0.9683	0.6087
Correlation	0.9012	0.6024
Mean Square Error (MSE)	14927.28 x 10^{-3}	126351.9 x 10^{-3}
Coefficient of Variation	0.1196	0.0174
Square Root of MSE	1.18855	3.463016

2. Result on Abnormal Pressure Application to the Safety Factor

The results of the experiment carried out on the effect of high temperature in the reactor as a function of Safety Factor.

(i) Empirical Expression for Safety Factor, $\grave{Y}$

Using the input parameter in Tables 1 and 2, for Safety Factor, $\grave{Y}$
The model empirical expression for the Safety Factor is obtained, as:

$$\boxed{\grave{Y} = (0.3006) + (0.0011)*(X_j) + e_j} \quad (5)$$

where,
Ỳ = Safety Factor, 0.3006 is the intercept, 0.0011 is a slope, X is the flow rate of high or low pressure, e = error or residual and j = 1,2,3,…,10.

Equation (5) is the model empirical expression that could be applied to make predictions of the Safety Factor on this type of reactor design model.

Linear Regression Plot

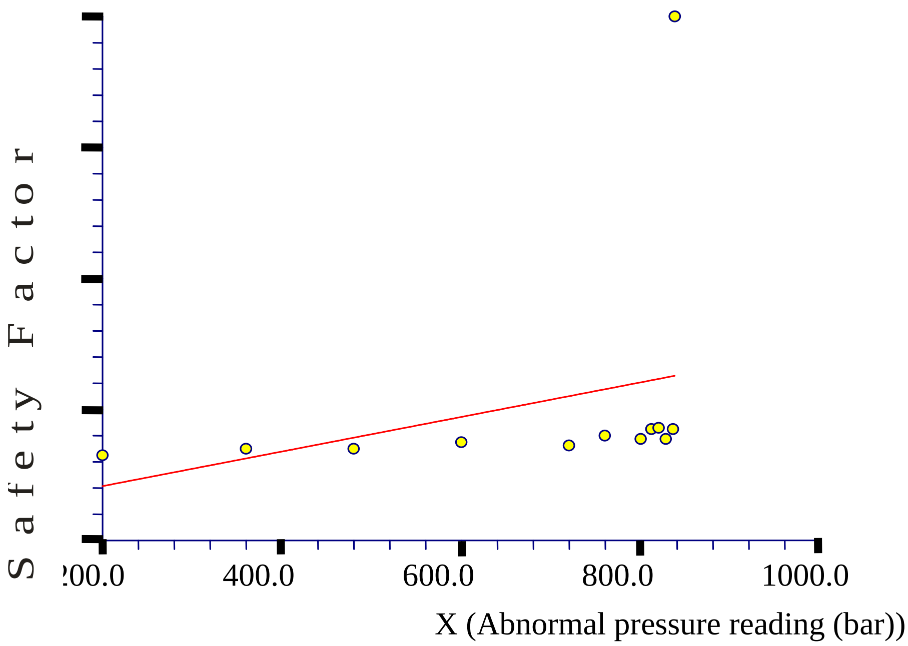

Figure 4: Safety Factor as a function of Abnormal Pressure Reading

(vi) F-test Result

<u>Table 5. Summary of F-test Statistical Data on Safety Factor Application of Abnormal Pressure in the Water-Cooled Reactor</u>

Parameter	Value of Pressure Reading (Normal)
Dependent Variable	Safety Factor
Independent Variable	Abnormal Pressure
Intercept(B_0)	0.3006
Slope(B_1)	0.0011
R-Squared	0.0921
Correlation	0.3034
Mean Square Error (MSE)	3.4976×10^{-3}
Coefficient of Variation	0.9024
Square Root of MSE	1.18855

3. Application to Accident Cases of Loss-of-Coolant in Water-Cooled Reactor

Loss-of Pressure-Control can be related to Loss-of-Coolant-Accident like where the operating reactor gradually loss coolant and stop functioning, we could take example from the Three Mile Island nuclear power plant in USA as reported that "a cooling malfunction caused part of the reactor core to melt in the # 2 reactor, the reactor was destroyed". As identified in the case of Fukushima Daiichi Nuclear Accident March 2011 that "the fuel became critical as it could not cool down". Furthermore, Reuters reported that Fukushima Daiichi Unit 3 has lost cooling capability and may be experiencing melting of the core, eventually reactor 1- 4 was written-off. These situations can be applied to Figure 5. During operation the reactor is stable as water coolant flow rise from 200kg/sec to 600kg/sec, and operate steadily between the safety factor of 6 to 11 and maintained cooling at 600kg/sec, but latter fall to 500kg/sec and suddenly drop from 100kg/sec to near zero level, at this point the reactor becomes unstable, safety is no longer guarantee, as the reactor can start melting since cooling of the reactor is no longer taken place, though other parameters may be held constant e.g. power, control rod, etc.

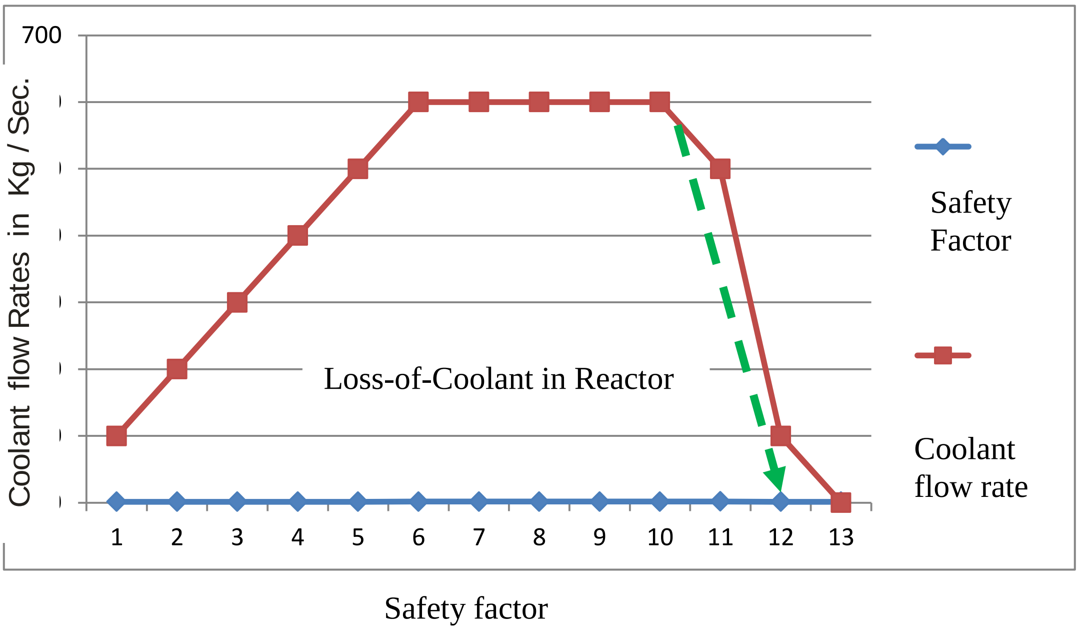

Figure 5: Demonstration of Loss-of-Coolant in Water-Cooled Reactor

4. Summary/Conclusion

This work focus on a typical water-cooled reactor designs for normal pressure within reactor core viz WCRD NP and a typical water-cooled reactors designs for high pressure within reactor core viz WCRD AP. The empirical expressions for the optimization of nuclear reactor Safety Factor ($\grave{Y}$) as functions of pressure for water-cooled reactor design models (WCRDMs) are obtained as:

(i) $\grave{Y} = (10.3887) + (1.2608)*(X_j) + e_j$, for WCRD NP

(ii) $\grave{Y} = (273.9054) + (0.7650)*(X_j) + e_j$, for WCRD AP

(iii) $\grave{Y} = (0.0000) + (1.9829)*(X_j) + e_j$, for WCRD AP

Types of Nuclear Power Reactor Design Model	**Correlation**	**R^2**	**Mean Square Error**
Water-Cooled Reactors Design			
WCRD NP	0.9012	0.9683	14927.28×10^{-3}
Water-Cooled Reactors Design			
WCRD AP High Pressure	0.5651	0.5852	116351.9×10^{-3}
WCRD AP Zero Pressure	0.6024	0.6087	126351.9×10^{-3}

- 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 pressure to the stability of the reactor.

The Table 6 highlights the summary results on pressure effects on water-cooled reactors.

<u>Table 6. Summary Results on Effects of Pressure on Water Cooled Reactors</u>

❖ In Figure 6 it is obvious that the WCRDs NP has the highest values of correlation. It is also understandable that WCRD NP model with correlation value of 0.9012 is better optimized than any other WCRD AP models.

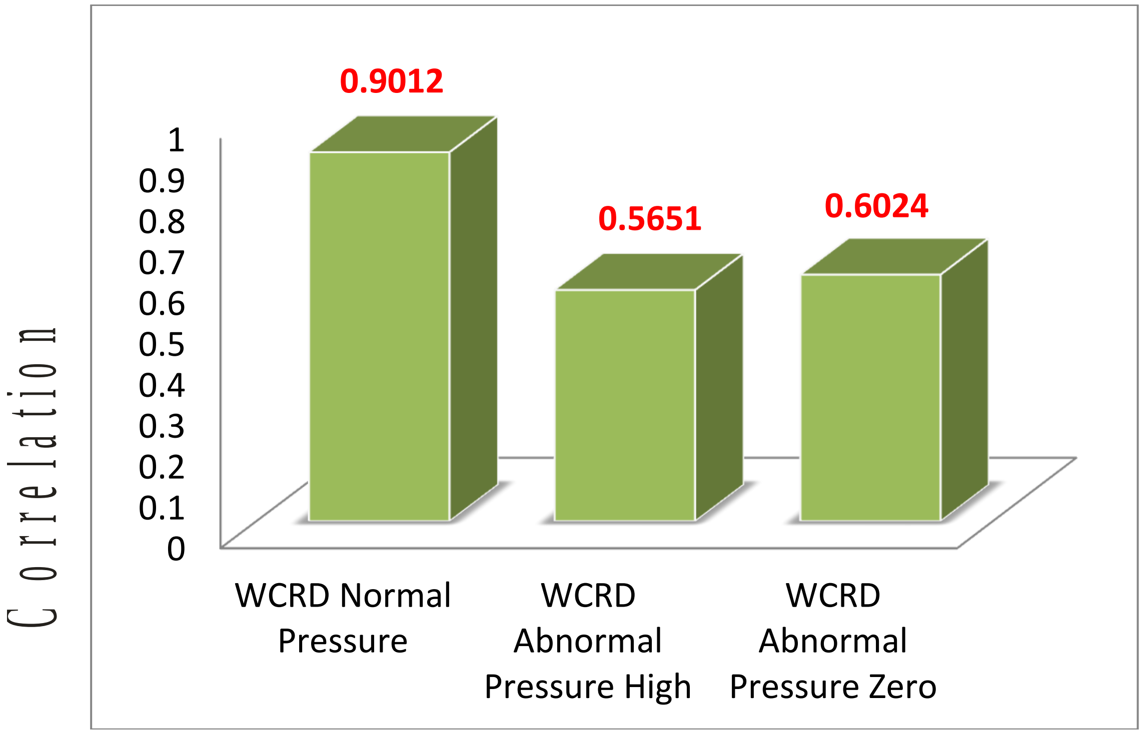

Figure 6: Effect of Pressure WRT Correlation on the Reactors

❖ In Figure 7 the charts reveal that, WCRDs AP have low values of coefficient of determination (R^2), than WCRDs NP. The WCRD NP model with R^2 of 0.9683 could be said to promise the best stability and possibly the safest when compared to WCRD AP models. Therefore, it could be said that WCRD NP are better optimized than WCRDs AP.

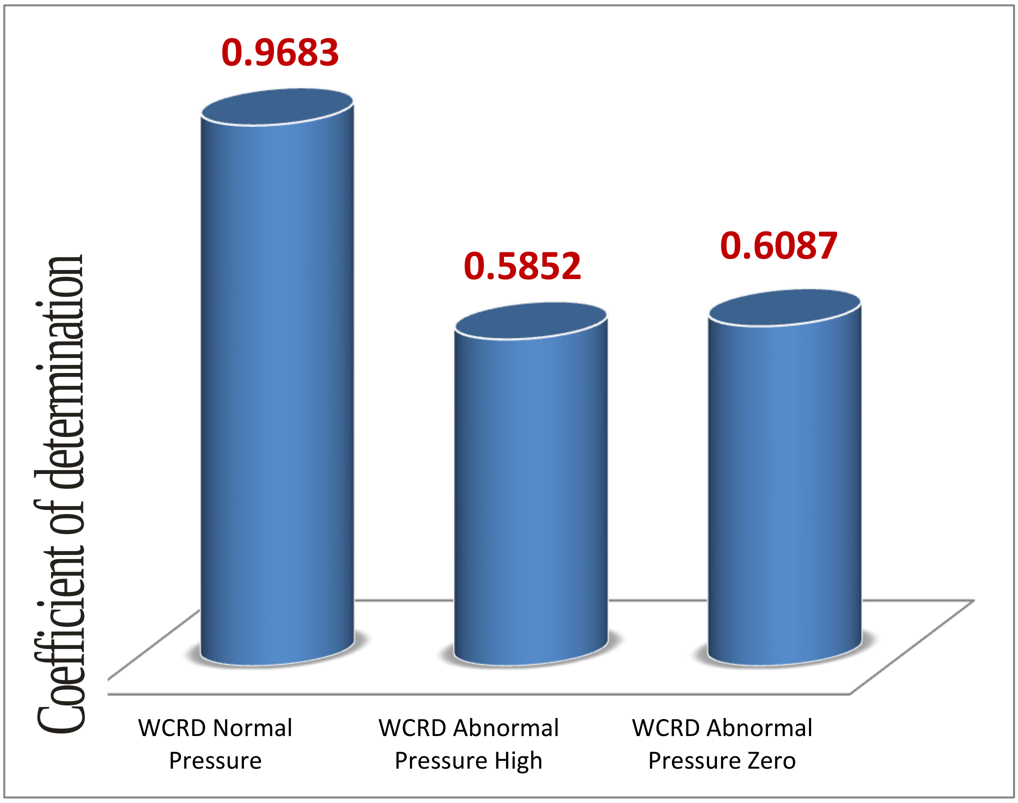

Figure 7: Effect of Pressure on the Reactor

❖ In Figure 8, charts reveal that WCRDs AP have higher values of the mean square of errors (126351.9) and (116351.9) respectively than WCRD NP (14927.28). Since WCRD NP has minimum mean square of error it means that WCRD NP models could promises most safety features than WCRD AP models.

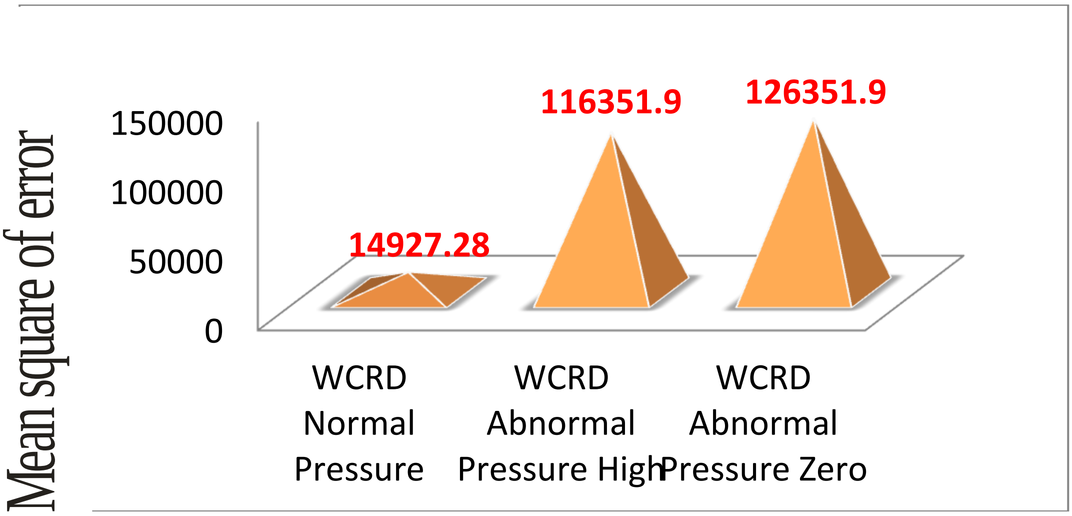

Figure 8: Effect of Pressure on the Reactor

In conclusion linear regression analysis is applied on two typical Water-Cooled Nuclear Power Reactor Design Models. Empirical expressions are obtained for WCRD NP model and WCRD AP models. The results of the statistical analyses on these types of nuclear reactor models reveal that the WCRD NP models promises to be more stable than WCRD AP models.

In Table 3 the value of R^2 = 0.9683 or 97% is obtained for the model equation (1) in this work. This is higher than the threshold value of R^2 = 0.673 or 67.3% for n=2 and j = 10, and this promises an acceptable level of validity. Thus the estimated model equation is significant at the given significant level of 5%. Unlike WCRD AP models the estimated model equations with R^2 (0.5852) and R^2 (0.6087) respectively less than R^2 (0.6723), the models are not acceptable as they may has no significant practical application.

In this method of regression analysis the safety margin prediction of up to 3.2% has been validated for reactor design models on normal pressure reactor core as an advantage over the current 5.1% challenging problem for plant engineers to predict the safety margin limit. In this method of regression analysis the safety margin prediction of up to 41.48% and 39.13% has been validated for reactor design models on abnormal pressure of water-cooled reactor as a disadvantage 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 discoveries on water-cooled reactor safety factor should provide a new method for reactor design concept taken cognizant of pressure built-up trouble in the reactor core. This shall also provide 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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References

Wikipedia Encyclopedia, 2006. Loss-of-Pressure-Control Accident. [online] Available at: <http://en.wikipedia.org/wiki/Loss-of-pressure-control_accident>[AccessedOctober 2006].

U.S.NRC Library, 2013**.** Backgrounder on the Three Mile Island Accident. [online] Available at: <http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html> [Accessed February 2013].

IAEA, 2008. Analysis of Severe Accidents in Pressurized Heavy Water Reactors[pdf] Available at: <http://*www-pub.*iaea*.org/MTCD/publications/PDF/te_1594_web.pdf*>[Accessed June 2008].

Ordan, M., Tauscher R. and Mayinger F., *(*1997*)*. Influence of Turbulence on the Deflagrative Flame Propagation in Lean Premixed Hydrogen Air Mixtures[pdf] Available at: <http://*www.td.mw.tum.de/tum-td/de/forschung/pub/CD_**Mayinger**/293.pdf*>[Accessed *1997).*

Neil Todreas (2005). Loss of Coolant Accidents (LOCA) in PWRs Deterministic View [pdf] Available at: <http://*www.space.mit.edu/bitstream/handle/1721...**2005**/...**2005**/.../lec9_nt.pdf*>[Accessed December, 2005*).*

IAEA, 2010. Technical reports on nuclear energy series publications No. NF-T-2. 2010 - Review of Fuel Failures in Water Cooled Reactors [pdf] Available at: <http://*www-pub.**iaea**.org/MTCD/**publications**/PDF/Pub1445_web.pdf*>[Accessed, 2010*).*

Hui-Wen Huang, et al. (2006). Digital Instrumentation and Control Failure Events Derivation and Analysis for Advanced Boiling Water Reactor[pdf] Poland: International Conference on Dependability of Computer Systems. Available at: <http://www.doi.ieeecomputersociety.org/10.1109/DEPCOS-RELCOMEX.2006.18>[Accessed, May 25-May 27 2006*).*

R. Feres, et al. (2009). A general formula for reactant conversion over a single catalyst particle in TAP pulse experiments [pdf] Washington: Washington University, Department of Mathematics. Available at: <http://www.math.wustl.edu/~feres/conversion_finite_k.pdf>[Accessed, June 27 2009)

M. Raghed, 2013. Nuclear, Plasma and Radiation Science.[pdf]Available at: <http://mragheb.com/NPRE%20402%20ME%20405%20Nuclear%20Power%20Engineering/Title-Preface.pdf> [Accessed August 26, 2013].

Recent research (A.I. Oludare, M.N. Agu and P.O. Akusu 2013). Optimization of the Stability Margin for Nuclear Power Reactor Design Models Using Regression Analyses Techniques[pdf]Available at: *<http://www.article.sapub.org/10.5923.j.jnpp.20130303.01.html*>[Accessed Mar 3, 2013].

Jonas R., et al. (2009). Modeling and Simulation of an Industrial Trickle-Bed Reactor for Benzene Hydrogeneration**:** Model Validation against Plant Data [pdf] Available at: *<http://www*. http://pubs.acs.org/doi/abs/10.1021/ie801411n>[Accessed January 5, 2009].

X. Yuan, 2007. Stochastic Modeling of Deterioration in Nuclear Power Plants Components. Ph. D. University of Waterloo, Ontario, Canada. Available through**:** *<http://www.ryerson.ca/~xyuan/docs/thesis.pdf*>[Accessed 2007].

Recent research (E. Zio, G. E. Apostolakis, and N. Pedroni 2010). Quantitative functional failure analysis of a thermal-hydraulic passive system by means of bootstrapped Artificial Neural Networks[pdf]Available at:
<http://yadda.icm.edu.pl/yadda/element/bwmeta1.element.elsevier-84d44dcb-d9b1-3649-a59f-d51d1d8cd924>[Accessed 2010].
M. Raghed, 2011. Chernobyl Accident[pdf] Available at:
http://www.mragheb.com/.../Chernobyl%20Accident.pdf>[Accessed September 4 2011].
M. Raghed, 2011. Chernobyl Accident [pdf] Available at:
<http://mragheb.com/NPRE%20402%20ME%20405%20Nuclear%20Power%20Engineering/Chernobyl%20Accident.pdf [Accessed September 4 2011].