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HIGH POWER DENSITY ON REACTORS STABILITY AND SAFETY

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ABSTRACT: *This paper investigated the high power density problem cause by inefficient cooling system in nuclear power plants. Safety margin test was conducted on some typical water-cooled reactor design (WCRD) models at operational level and at an accident condition, secondly safety margin test was carried out on the thermal efficiency and thermal power output of the reactor when power supply failed and thirdly, safety margin test was perform on the reactor in relation to an increase in fuel temperature in the reactor core. The results of the statistical analysis on these types of nuclear reactor models reveals that the typical water-cooled reactor design (WCRD) models promises most stability under thermal efficiency of 45% and above. The research implication is that the WCRD models could be significantly most stable at thermal efficiency of 45% and above. Secondly, the safety margin prediction of up to 3.1% has been validated for reactor design models on water-cooled reactor. The research effort served 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*.

KEYWORDS: Water-Cooled Reactor Design Models Accident, High Power Density, High Fuel Temperature, Thermal Efficiency And Thermal Power Output, Reactor Stability And Safety.

INTRODUCTION

There have been recorded cases of high power density problem in typical operating light water power reactors, this problem is usually caused by cooling inefficiency in the reactor core and fuel. The cooling inefficiency situation could results to hydrogen built-up in the reactor core. The increase in the number of neutrons will split further uranium atoms., the reactivity increase dramatically and leads to an increase in power, fuel enthalpy and fuel temperature, the fuel can damage as identified in the Fukushima Daiichi nuclear plant, the fuel became critical as it could not cool down and then the reactor failed[1].

These failures have caused some major fatal accidents, these accidents has received international attention and, although there are still gaps in knowledge relating to details of some phenomena involved in the accident, the causes and the failure have been clearly identified and measures implemented to avoid a repetition of these events. As is often the case in major disasters, the causes relate to two areas – poor design of the reactor and coupled with the lack of a safety culture which led to violation of standard operating procedures [2]. System innovative technologies under consideration need safety hazards analyses process before testing or experimentation in other to avoid sudden failure that can lead to severe disaster in the economy. Therefore, this paper seeks to provide an approach for increasing cooling efficiency, cooling capacity, and low power density in the operation of nuclear power plant.

Nevertheless, researches have shown that small quantity of nuclear fuel will provide low power density reactor that will mitigate heat generation in the reactor core that limit reactor melt-down[3]. Whereas, large quantity of uranium (fuel) could contribute to the causes of pressure built-up within reactor core of nuclear power plant, the decay heat in the core assemblies may cause reactor to meltdown. A core melt accident occurs when the heat generated by a nuclear reactor exceeds the heat removed by the cooling systems to the point where at least one nuclear fuel element exceeds its melting point. This is different from a fuel element failure, which is not caused by high temperatures.

In most cases a reactor meltdown may be caused by; excessive heat within the reactor, a loss-of-coolant, loss-of-coolant-pressure, or low coolant flow rate nor be the result of a criticality excursion in which the reactor is operated at a power level that exceeds its design limits. A meltdown is considered a serious event because of the potential for release of radioactive material into the environment. Unlike all other forms of electrical power generation, upon shutdown of a nuclear power plant, the power from the core does not instantaneously stop.
This remaining power is generated when fission products release energy by undergoing radioactive decay, typically β - decay and γ - decay for uranium-235 fission daughter products. This power is commonly referred to as the decay heat production rate or decay heat.

Definition: 1 Power density can be defined as the energy deposited in the fissile material per unit volume per unit time. Can be written as;
$P(r) = \int_0^\infty wf\ (E,r)\Sigma f\ (E,r)\phi(E,r)dE$ ……………………………………………………(a)
where, Σ_f and φ denote macroscopic fission cross section and neutron spectrum, respectively.

Let take into consideration the prompt energy, radiation and particle release, the average total amount of energy released per fission of one uranium-235 atom as in Table 1.

Table 1 - Average Energy from Uranium-235 Fission[13] Source: [4]

S/n		Emitted Energy (MeV)	Recoverable Energy (MeV)
	Instantaneous Energy		
1	Kinetic energy of fission fragments	165.6	165.6
2	Kinetic energy of fission neutrons	4.8	4.8
3	Fission gamma rays	7.7	7.7
4	Neutron capture gamma rays	7.5	7.5
	Total instantaneous energy	**185.6**	**185.6**
	Delayed Energy		
5	Kinetic energy of beta particles	7.2	7.2
6	Delayed neutrons	˜0	˜0
7	Fission product decay gamma rays	7.2	7.2
8	Antineutrinos	10.2	0
	Total delayed energy	**24.6**	**14.4**
	Total energy released per fission	**210.2**	**200.0**

In electrodynamics, the force on a charged particle of charge *q* is the Lorentz force**:** Using figure 1

F = q(E+ v x B) …………………………………………………………………………………(1)

where velocity = **v**, **E** field and **B** field vary in space and time.

Combining with Newton's second law gives a first order differential equation of motion, in terms of position of the particle:

$$m\frac{\mathrm{d}^2\mathbf{r}}{\mathrm{d}t^2} = q\left(\mathbf{E} + \frac{\mathrm{d}\mathbf{r}}{\mathrm{d}t}\times\mathbf{B}\right) \qquad \text{……..(2)}$$

or its momentum:

$$\frac{\mathrm{d}\mathbf{p}}{\mathrm{d}t} = q\left(\mathbf{E} + \frac{\mathbf{p}\times\mathbf{B}}{m}\right) \qquad \text{……..(3)}$$

The same equation can be obtained using the Lagrangian (and applying Lagrange's equations above) for a charged particle of mass *m* and charge *q*:

$$L = \frac{m}{2}\dot{\mathbf{r}}\cdot\dot{\mathbf{r}} + q\mathbf{A}\cdot\dot{\mathbf{r}} - q\phi \qquad \text{……..(4)}$$

where **A** and ϕ are the electromagnetic scalar and vector potential fields. The Lagrangian indicates an additional detail: the canonical momentum in Lagrangian mechanics is given by**:**

$$\mathbf{P} = \frac{\partial L}{\partial \dot{\mathbf{r}}} = m\dot{\mathbf{r}} + q\mathbf{A} \quad \ldots\ldots(5)$$

instead of just $m\mathbf{v}$, implying the motion of a charged particle is fundamentally determined by the mass and charge of the particle. The Lagrangian expression was first used to derive the force equation.
Alternatively the Hamiltonian (and substituting into the equations):

$$H = \frac{(\mathbf{P} - q\mathbf{A})^2}{2m} - q\phi \quad \ldots\ldots(6)$$

Therefore, we can derive the Lorentz force equation using the figure 1, and this shown Lorentz force **f** on a charged particle (of charge *q*) in motion (instantaneous velocity **v**). The **E** field and **B** field vary in space and time.

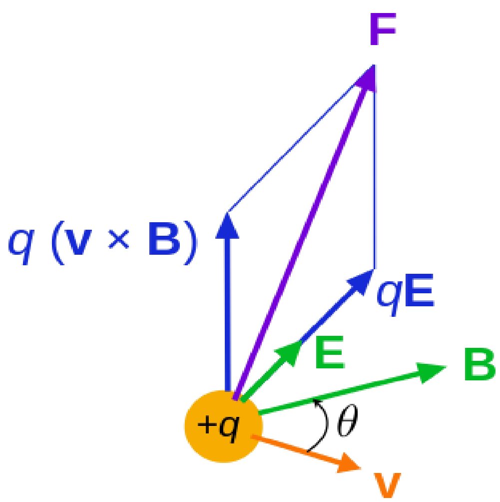

Figure 1: shown Lorentz force **f** on a charged particle (of charge *q*) in motion (instantaneous velocity **v**).
The **E** field and **B** field vary in space and time.

NUCLEAR FUEL TEMPERATURE

The nuclear fuel temperature coefficient of reactivity is the change in reactivity of the nuclear fuel per degree change in the fuel temperature. The coefficient quantifies the amount of neutrons that the nuclear fuel (uranium-238) absorbs from the fission process as the fuel temperature increases. It is a measure of the stability of the reactor operations. This coefficient is also known as the Doppler coefficient [5].
Table 1 presents typical operational temperatures that should not be exceeded to avoid fission production release.

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Table 2 operational temperature of different reactor fuels. Source: [6]

S/N	Reactor Concept	Temperature Degree Celsius
1	Pressurized Water Reactor (PWR)	320
2	Boiling Water Reactor (BWR)	300
3	Liquid Metal Reactor(LMR)m Na Cooled	750
4	Magnesium Alloy cladding (magnex)	450
5	AGR stainless steel cladding	750

When the nuclear fuel increases in temperature, the rapid motion of the atoms in the fuel causes an effect known as Doppler broadening. When thermal motion causes a particle to move towards the observer, the emitted radiation will be shifted to a higher frequency. Likewise, when the emitter moves away, the frequency will be lowered. For non-relativistic thermal velocities, the Doppler shift in frequency will be:

$$f = f_0 \left(1 + \frac{v}{c}\right) \qquad (7)$$

where f is the observed frequency, f_0 is the rest frequency, v is the velocity of the emitter towards the observer, and c is the speed of light.
Since there is a distribution of speeds both toward and away from the observer in any volume element of the radiating body, the net effect will be to broaden the observed line.

If $P_v(v)dv$ is the fraction of particles with velocity component v to $v + dv$ along a line of sight, then the corresponding distribution of the frequencies is:

$$P_f(f)df = P_v(v_f)\frac{dv}{df}df, \qquad (8)$$

where

$$v_f = c\left(\frac{f}{f_0} - 1\right) \qquad (9)$$

is the velocity towards the observer corresponding to the shift of the rest frequency f_0 to f. therefore,

$$\boxed{P_f(f)df = \frac{c}{f_0}P_v\left(c\left(\frac{f}{f_0} - 1\right)\right)df} . \qquad (10)$$

We can also express the broadening in terms of the wavelength λ. Recalling that in the non-relativistic limit $\frac{\lambda - \lambda_0}{\lambda_0} \approx -\frac{f - f_0}{f_0}$, we obtain

$$\boxed{P_\lambda(\lambda)d\lambda = \frac{c}{\lambda_0}P_v\left(c\left(1-\frac{\lambda}{\lambda_0}\right)\right)d\lambda}.$$
……… ……………………….(11)

In the case of the thermal Doppler broadening, the velocity distribution is given by the Maxwell distribution

$$P_v(v)dv = \sqrt{\frac{m}{2\pi kT}}\exp\left(-\frac{mv^2}{2kT}\right)dv$$
,……… .. …… …………………………(12)

where,
m is the mass of the emitting particle, T is the temperature and k is the Boltzmann constant.
Then,

$$P_f(f)df = \left(\frac{c}{f_0}\right)\sqrt{\frac{m}{2\pi kT}}\exp\left(-\frac{m\left[c\left(\frac{f}{f_0}-1\right)\right]^2}{2kT}\right)df$$
,… …… …………...(13)

We can simplify this expression as:

$$P_f(f)df = \sqrt{\frac{mc^2}{2\pi kT{f_0}^2}}\exp\left(-\frac{mc^2\left(f-f_0\right)^2}{2kT{f_0}^2}\right)df$$
,…… ………………(14)

which we immediately recognize as a Gaussian profile with the standard deviation

$$\sigma_f = \sqrt{\frac{kT}{mc^2}}f_0$$
,……………………………………… ……
……………………………….(15)

and full width at half maximum (FWHM)

$$\boxed{\Delta f_{\text{FWHM}} = \sqrt{\frac{8kT\ln 2}{mc^2}}f_0}.$$
,…………… … ……
…………………………………(16)

The fuel then sees a wider range of relative neutron speeds. Uranium-238, which forms the bulk of the uranium in the reactor, is much more likely to absorb fast or epithermal neutrons at higher temperatures. This reduces the number of neutrons available to cause fission, and reduces the power of the reactor. Doppler broadening therefore creates a negative feedback because as fuel temperature increases, reactor power decreases. All reactors have reactivity feedback mechanisms, except some gas reactor such as pebble-bed reactor which is designed so that this effect is very strong and does not depend on any kind of machinery or moving parts.

SYSTEM RELIABILITY

The life of a system or a device under reliability study follows a sequence that results in an observable time to failure. A new device is put into service, it functions acceptably for a period of time and then it fails to function satisfactorily. The observed time to failure is a value of the random variable T, which represents the lifetime of the device. T takes its values in an interval of the real numbers, R, most often in the closed interval [0,∞). Since the lifetime of a device is represented by a random variable T, there is a probability distribution function (cdf) of T,

FT(t) = P(T ≤ t), 0 < t. … ……
……………………………………………………………………(17)
FT(t) is usually called the unreliability at time t. It represents the probability of failure in the interval [0,t]. The probability of failure in the interval (t1,t2] equals F(t2) − F(t1).

Definition 5: The reliability function is:
RT(t) = P(T>t) =1 − FT(t) . … ……
……………………………………………………………(18)

Thus, reliability is the probability of no failures in the interval [0,t] or equivalently, the probability of failure after time t. Sometimes T will take on only a countable number of values in R. This case, called the discrete case, occurs when T is a number of cycles, for example, or when the failure time can occur at only discrete points.
Most of the time, however, T will be a continuous random variable and its distribution FT(t) will be a continuous distribution having a density fT(t).

RELIABILITY WITH CONTINUOUS RANDOM VARIABLES

Assume T is a continuous random variable, taking values in open interval (0,∞) and with density function fT(t). The reliability function RT(t) is**:**
RT(t) = $\int_t^{\infty} fT(x)dx = 1 - \int_0^t fT(x)dx = 1 - FT(t)$. … …… ………(19)
where, FT(t) ≥ 0 and $\int_0^{\infty} fT(x)dx = 1$

HIGH DENSITY, FAILURE AND ACCIDENT ANALYSIS

Calculation of power density is possible using Monte Carlo N-Particle (MCNP) Transport Code software package for simulating nuclear processes[7]. MCNP computer code is a tool for particle transport calculations[8]. It can be used for transport of neutrons, photons and electrons. Transport of neutrons is of special interest for a reactor physicist. Precautions and measure to reduce cooling problem were stated in "Ten Cooling Solutions to Support High-Density Server Deployment"[9].

There have been several reports analysis on the safety of reactors with respect to nuclear fuel damage and an increase in fuel temperature these includes; "Nuclear Fuel Safety Criteria

Technical Review"[10], "Nuclear fuel behaviour under reactivity-initiated accident (RIA) conditions - State-of-the-art report,"[11], "Current Trends in Nuclear Fuel for Power Reactors,"[12] and "Review of Fuel Failures in Water Cooled Reactors,"[13]. Others are "Analysis of Severe Accidents in Pressurized Heavy Water Reactors"[14], "Nuclear Reactor Theory and Reactor Analysis"[15], "Backgrounder on the Three Mile Island Accident"[16], "Status of thermohyraulic research in nuclear safety and new challenges"[17], "Regulatory Guide"[18] and "The unsteady state operation of chemical reactors"[19].

These accidents may perhaps be as a result of design concept process of some of these reactors (which could involve novel technologies) that have inherent risk of failure in operation and were not well studied/understood. In avoiding such accidents the industry has been very successful. As in over 14,500 cumulative reactor-years of commercial operation in 32 countries, there have been only three major accidents to nuclear power plants – Fukushima, Chernobyl and Three Mile Island. As in other industries, the design and operation of nuclear power plants aims to reduce the likelihood of accidents, and avoid major human consequences when they occur.

However, recent study of the reactor fuel under accident conditions, reveal that after subjecting the fuel to extreme temperatures — far greater temperatures than it would experience during normal operation or postulated accident conditions — TRISO fuel is even more robust than expected. Specifically, the research revealed that **at *1,800 degrees Celsius*** (more than 200 degrees Celsius greater than postulated accident conditions) most fission products remained inside the fuel particles, which each boast their own primary containment system.

METHODOLOGY

A brief discussion of some past and recent accident of nuclear power plant due to control rod trip – failures. An assessment risk of the control rod trip failures and comparative analyses of such incident. The design parameter of control rod was used to test the correlation between reactor safety margin and fuel temperature especially in an accident scenario.
Therefore, the safety factor (Ỳ), of the reactor can be calculated or determined using the linear regression empirical formula.

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**:** "Statistical Analysis of Reactor Pressure Vessel Fluence Calculation Benchmark Data Using Multiple Regression Techniques"[20], and "Simplified modeling of a PWR reactor pressure vessel lower head failure in the case of a severe accident"[21].

Others are, "Analyses of loads on reactor pressure vessel internals in a pressurized water reactor due to a loss-of-coolant accident considering fluid-structure interaction"[22], "Regression analysis of gross domestic product and its factors in Lithuania"[23], "Optimization of the Stability Margin for Nuclear Power Reactor Design Models Using Regression Analyses Techniques,"[24] and "Comparative Analyses of Water-Cooled Reactors Design Models & Gas-Cooled Reactors Design Models"[25].

Objective of the Research

In this work comparism of different test on water-cooled reactor design (WCRD) models with respect to high power density during operation or accident was carried out by testing for fuel temperature, thermal power and thermal efficiency using regression analysis technique. The research aimed at demonstrating sufficient safety margins, for nuclear power plants. One objective of this research is to evaluate power system reliability analysis improvements with distributed generators while satisfying equipment handling constraints.

In this research, a computer algorithm involving pointers and linked list is developed to analyze the power system reliability. This algorithm needs to converge rapidly as it is to be used for systems containing thousands of components. So an efficient "object-oriented" computer software design and implementation is investigated. This algorithm is also used to explore the placement of control rod and how the different placements affect system reliability, which has not been done in previous research. This exploration makes possible the comparison of alternative system designs to discover systems yielding desired reliability material properties. In this paper, variation of system reliability with the varying loads is also investigated. Other publications of distribution system reliability analysis associated with time varying loads have not been found.

Motivation of the Research

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. Also, 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. Moreover, the study is to 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. Furthermore, this is to help Nigeria meet its international obligations to use nuclear technology for peaceful means. Finally, the achievement is to make worldwide contribution to knowledge.

Research Design/Approach

In this work, a statistical analysis of a design input parameter of a typical reactor water-cooled reactor was investigated for safety under a high power density reactor. Specifically, the studies concentrated on technical factors that limit the functionality of the reactor with respect to power density, such as the mechanical interaction, malfunctioning, failure, reactor fuel temperature, thermal power and thermal efficiency. The Table 3 presents data **i**nput for safety margin against thermal power and thermal efficiency of some typical water-cooled reactor design model.

Table 3: Data input for thermal power and thermal efficiency of some typical water-cooled reactor design model.

Nos. of trial (j)	**Thermal Power (MW)**	**Thermal Power (MWe)**	**Thermal Efficiency (%)**
1	200	100	30.00
2	210	105	31.00
3	215	107	32.50
4	218	110	33.30
5	225	112	34.80
6	233	115	35.00
7	240	117	36.70
8	247	119	41.00
9	250	120	45.00
10	253	123	47.60
11	260	129	49.80
12	263	130	50.00

Table 4: Input data for fuel size and heat generation in a typical water-cooled reactor.

Nos. of trial (j)	**Fuel size in Mass (g)**	**Heat Generated ^oC**
1	2.8	200
2	3.5	270
3	4.2	300
4	5.0	440
5	5.7	480
6	6.0	520
7	7.4	600
8	8.3	760
9	9.0	900
10	10.6	1050
11	11.0	1100
12	**12.0**	**1200**

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RESULTS AND ANALYSES

1. Water-Cooled Reactor Design Model **(WCRDM)**

The result of the application of the linear regression analysis of the data in Tables 2 and 3 of a typical water-cooled reactor design model is presented as follows**:**

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

In the investigation of high density reactor cooling problem on reactor stability and safety during operation, the data obtained in Tables 2 and 3 which represents a typical parameter for a typical water-cooled reactor design model was used in order to obtain the best fit for the model. The new conceptual fuel design for reactor operation could optimize the performance of this type of water-cooled reactor design model.

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

$$\grave{Y} = B_0 + B_1 X_j + e_j \quad \ldots\ldots (20)$$

where,

B_0 is an intercept, B_1 is the slope, X_j is the rate of increase in fuel volume
e_j = error or residual, j = 1,2,3,…,k and k is the last term.

Empirical Expression for Safety Factor, $\grave{Y}$ for Normal Pressure Reading

The model empirical expression is the equation of the straight line relating heat in the reactor and the volume of fuel in the reactor as a measure of safety factor estimated as**:**

$$\boxed{\grave{Y} = (-14.0347) + (5.9992)*(X_j)} + e_j \quad \ldots\ldots (21)$$

- the equation (21) is the estimated model or predicted
where,

$\grave{Y}$ = Dependent Variable, Intercept = -14.0347, Slope = 5.9992, X = Independent Variable,
e = error or residual, j = 1,2,3,…,12 and 12 is the last term of trial.

The Figure 2 shows the linear regression plot section on thermal efficiency and thermal power

(ii) Linear Regression Plot on the relationship between thermal efficiency and thermal power

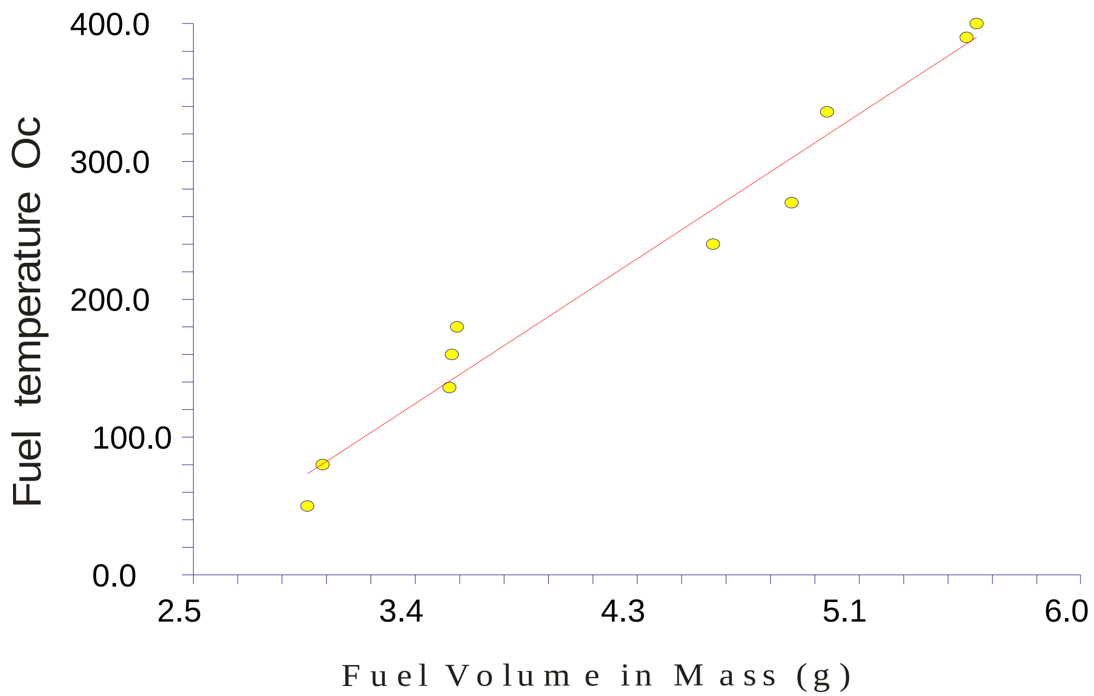

Figure 1**:** Fuel effect on the stability of operating reactor

(iii) F-test Result

Table 5: Summary of F-test Statistical Data

Parameter	**Value**
Dependent Variable	Ỳ (Decay heat or temperature)
Independent Variable	X (fuel volume)
Intercept(B_0)	-14.0347
Slope(B_1)	5.9992
R-Squared	0.9690
Correlation	0.9844
Mean Square Error (MSE)	1.338244×10^{-2}
Coefficient of Variation	0.1032
Square Root of MSE	1.156825

Table 6: Descriptive Statistics Section

Parameter	Dependent	Independent
Variable	Thermal efficiency	Thermal power
Count	10	10
Mean	11.2100	4.2080
Standard Deviation	6.1967	1.0168
Minimum	2.5000	2.9500
Maximum	20.0000	5.5900

The Table 7 is the regression estimation section results that show the least-squares estimates of the intercept and slope followed by the corresponding standard errors, confidence intervals, and hypothesis tests. These results are based on several assumptions that are validated before they are used.

Table 7: Regression Estimation Section

Parameter	Intercept B(0)	Slope B(1)
Regression Coefficients	-14.0347	5.9992
Lower 95% Confidence Limit	-17.8101	5.1247
Upper 95% Confidence Limit	-10.2592	6.8737
Standard Error	1.6372	0.3792
Standardized Coefficient	0.0000	0.9844
T-Value	-8.5722	15.8190
Prob Level (T-Test)	0.0000	0.0000
Reject H0 (Alpha = 0.0500)	Yes	Yes
Power (Alpha = 0.0500)	1.0000	1.0000
Regression of Y on X	-14.0347	5.9992
Inverse Regression from X on Y	-14.8417	6.1910
Orthogonal Regression of Y and X	-14.8205	6.1860

In Table 8 the analysis of variance shows that the F-Ratio testing whether the slope is zero, the degrees of freedom, and the mean square error. The mean square error, which estimates the variance of the residuals, was used extensively in the calculation of hypothesis tests and confidence intervals.

Table 8: Analysis of Variance Section

Source	DF	Sum of Squares	Mean Squares	F-Ratio	Prob. Level	Power(5 %)
Intercept	1	1256.641	1256.641			
Slope	1	334.8831	334.8831	250.2406	0.0000	1.0000
Error	8	10.70595	1.338244×10^{-2}			
Adj. Total	9	345.589	38.39878			
Total	10	1602.23				
S = Square Root(1.338244×10^{-2}) = 1.156825						

In Table 9 Anderson Darling method confirms the rejection of H_0 at 20% level of significance but all of the above methods agreed that H_0 Should not be rejected at 5% level of significance. Hence the normality assumption is satisfied as one of the assumptions of the Linear Regression Analysis is that the variance of the error variable δ^2 has to be constant.

Table 9: Tests of Assumptions Section

Assumption/Test Residuals follow Normal Distribution?	**Test Value**	**Probability Level**	**Is the Assumption Reasonable at the 20% or 0.2000 Level of Significance?**
Shapiro Wilk	0.9604	0.790403	Yes
Anderson Darling	0.2298	0.807631	Yes
D'Agostino Skewness	-0.0886	0.929399	Yes
D'Agostino Kurtosis	-0.5255	0.599243	Yes
D'Agostino Omnibus	0.2840	0.867626	Yes
Constant Residual Variance?			
Modified Levene Test	0.0024	0.961796	Yes
Relationship is a Straight Line?			
Lack of Linear Fit F(0, 0) Test	0.0000	0.000000	No

Notes:
A 'Yes' means there is not enough evidence to make this assumption seem unreasonable.
A 'No' means that the assumption is not reasonable

(iv) Residual Plots Section

The plot section is used as further check on the validity of the model to satisfy all the assumptions of the linear regression analysis. Amir D. Aczel (2002, P528) have stated that the normality assumption can be checked by the use of plot of errors against the predicted values of the dependent variable against each of the independent variable and against time (the order of selection of the data points) and on a probability scale. The diagnostic plot for linear regression analysis is a scatter plot of the prediction errors or residuals against predicted values and is used to decide whether there is any problem in the data at hand Siegel F (2002, p.578).
The Figure 2 is for the plot of errors against the order to selection of the data points (e = 1,2,…,12). Although the order of selection was not used as a variable in the mode, the plot reveal whether order of selection of the data points should have been included as one of the variables in our regression model. This plot shows no particular pattern in the error as the period increases or decreases and the residuals appear to be randomly distributed about their mean zero, indicating independence. The residuals are randomly distributed with no pattern and with equal variance as volume of fuel increases.

Note:

1. Residual = original value for heat (Y) minors predicted value for heat, Ỳ
2. Count = the design number (design 1, 2, 3, …, 12)

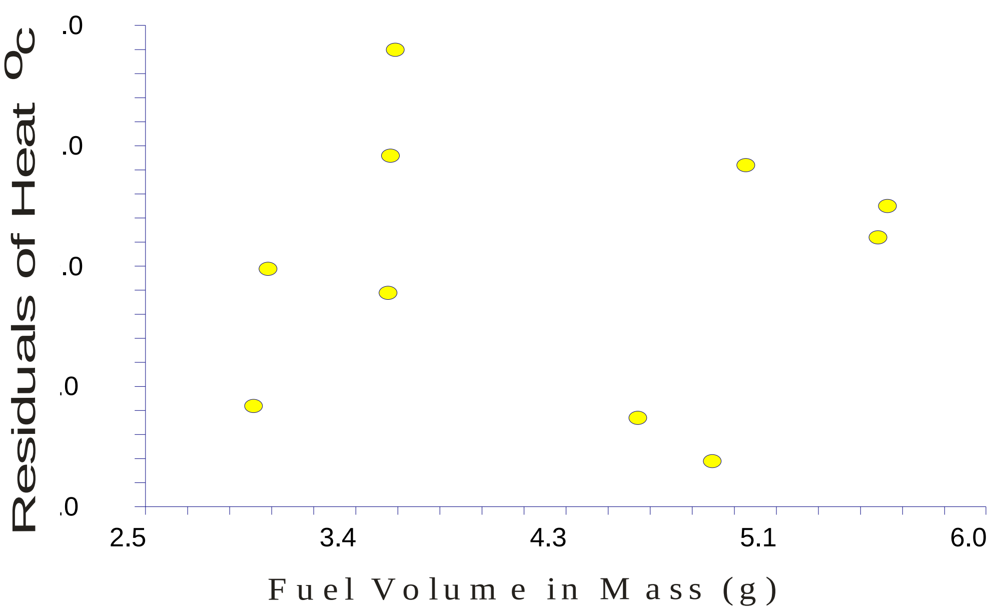

Figure 3: Residuals of Heat (^{0}C) versus Fuel (g)

Figure 3 shows the histogram of residuals of error (e_t) and this is nearly skewed to the right but the software used indicated that the plot is normal.

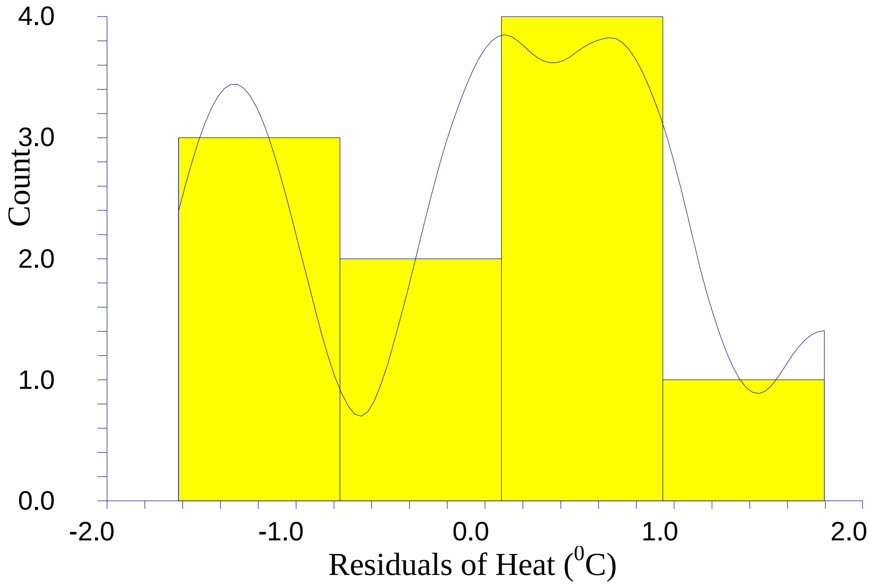

Figure 4: Histogram of Residuals of Heat (^{0}C)

While Figure 4 is the result on plot graph of experimental errors. The residuals are perfectly normally distributed as most of the error terms align themselves along the diagonal straight line with some error terms outside the arc above and below the diagonal line. This further indicates that the estimated model is valid.

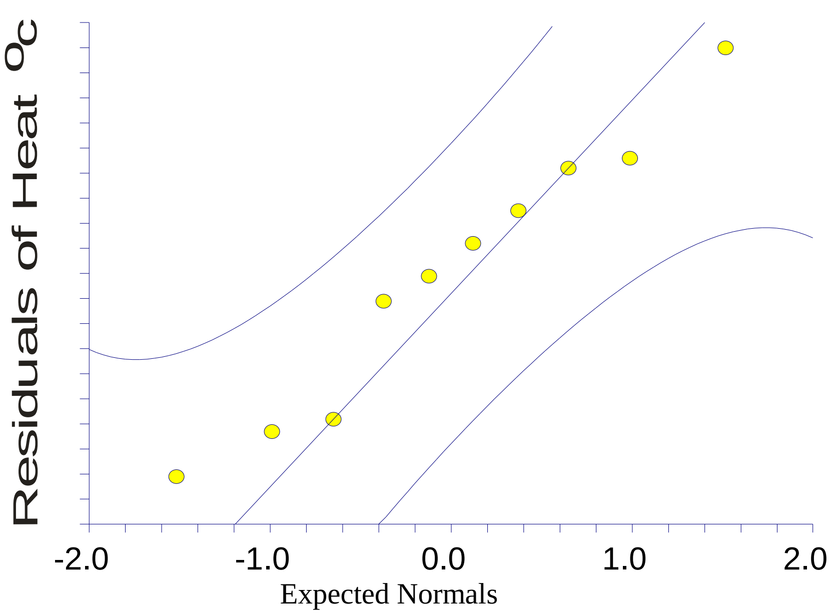

Figure 5: Normal Probability Plot of Residuals of Heat (^{0}C)

SUMMARY/CONCLUSION

In summary this paper examined the possibilities to derive and implement a method for reactor safety based on regression analysis techniques. Safety margin test was conducted on some typical water-cooled reactor design (WCRD) models at operational level and at an accident condition, secondly safety margin test was carried out on the thermal efficiency and thermal power output of the reactor when power supply failed and thirdly, safety margin test was perform on the reactor in relation to an increase in fuel temperature in the reactor core. The results of the statistical analysis on these types of nuclear reactor models reveals that the typical water-cooled reactor design (WCRD) models promises most stability under thermal efficiency of 45% and above.
The research implication is that in practice the WCRD models could be significantly stable at thermal efficiency of 45% and above than below 45%. Secondly, the safety margin prediction of up to 3.1% or 0.031 has been validated for reactor design models on water-cooled reactor which is of practical significant at 95% upper and lower confidence limit. The research effort served 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.

The current design limits for various reactors safety 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. However, the safety design of a nuclear power plant should include provisions in terms of pumping facilities and decay heat exchange equipment that could accommodate the decay heat generation immediately after shutdown, which would amount to safety margin say about 6 percent of the operational power level at one second after shutdown.

Thermodynamically speaking, it is suggested that the WCRD models "*should allow for thermal efficiency of 45% and above in their construction and possibly provision for extra in-built control rods in the design features to ensure safe operation of nuclear reactor"*.

If technology solution must be addressed properly then the following areas of applicable EPS technology needs to be well study these include power system reliability analysis improvements with distributed generators while satisfying equipment power handling constraints. An efficient "object-oriented" computer software design and implementation needs employ for investigation. Dynamic and seismic analysis; safety and reliability; and verification and qualification of analysis with relevant software.

The discoveries shall provide a good, novel approach and method for multi-objective decision-making based on seven dissimilar objectives attributes**:** materials selection, evolving technology, effectiveness, efficiency, cost, safety and failure. The implication of this research effort to Nigeria's nuclear power project drive.
It is therefore recommended that for countries wishing to include nuclear energy for the generation of electricity, like Nigeria, the design input parameters of the selected nuclear reactor should undergo test and analysis using this method for optimization and choice.

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