AIN- SHAMS UNIVERSITY

FACULTY OF ENGINEERING

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REACTOR DYNAMICS PROCESS IDENTIFICATION

BY

USE OF NOISE ANALYSIS IN CONTROL LOOP .

A THESIS

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BY

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S U M M A R Y

This work is a study of the methods of parameters identification used in reactor dynamics study, with application to Egyptian research reactor E-RR-1 . The approach is based on statistical analysis of random variations in neutrons density due to the different variables. The neutrons fluctuate due to the random nature of fission, random movement of control rods, irregularities in coolant flow in primary loop, the bubbles in the active zone, and the temperature change in the reactor core.

A primary discussion of the theory of statistical fluctuations in chain reactors is included, however, this discussion is biased towards engineering application rather than rigorous mathematical derivations. Using the fundamental concepts of statistics (correlation, and spectrum density), experiments were conducted on E-RR-1 to identify its parameters.

The measurment of prompt neutron life-time, and the estimation of the absolute power of the reactor for very low levels, were the aims of the first experiment conducted. The experimental result of neutron life time was in agreement with the theoretical value calculated using two-neutrons groups approximation.

There was some disagreement detected in power level estimation which calls for new calibration of control room instruments.

The identifecation of the temperature feed-back loop transfer function was the aim of the second experiment. The experimental results of the process of heating the reactor core with almost a step change of power were analyzed, and the frequency response was calculated. These results had been compared with frequency responses of the derived theoretical model. The model proved useful in parameter calculations and found in agreement with experiments and previously and published work.

The perturbations in the coolant flow, and the irregularities in moderator due to bubbles, change the reactor dynamic charateristics. The third experiment was carried out to clearify these factors, and to determine their spectral charteristics. The interpretation of these results and their dependence on the rate of flow of coolant in primary loop is discussed. The damped resonances as a function of the frequency were detected experimentally and found in agreement with the theoretical study and previous work.

A main feature of this work is that all experiments not require special in-core equipment, and the measurements are made in relatively short time. Also they can be performed without disturbing normal operation (except the transient experiment), and safety standards of the reactor system, while giving a powerful method for identification reactor dynamic processes.

In this study a mixed analog and digital techniques was used in general, as far as contineous recording followed by digitization, and off-line digital computer calculations. This proved to be quite useful since it gives access to autocorrelation function, as well as power spectrum. For measuring neutron life-time, and for estimating the absolute power level a frequency analyzer was used to obtain the power spectrum of neutrons flux directly, which has the advantage of reaching results directly as recorded on the instrument.

The algorithms, and computer codes which were written for calculations are given in appendices. The codes are written in FORTRAN language. The calculations are carried out on IBM-1130 and ICL-1905E machines held at computing centers of Ain-Shams and Cairo Universities respectively.

INTRODUCTION

This work is a study of different methods of parameter identification used in reactor dynamics study with application to the egyptian Research Reactor E-RR-1, to identify its dynamic parameters in control loop. The dynamic process of a reactor is mainly influnced and controlled by the charateristics of the primary coolant loop which represents the main controlling loop. The complete study of the effects of this loop (temperature and flow effects) on reactor dynamics, as well as the identification of the neutronic behaviour (neutron lifetime) of E-RR-1 reactor is covered in this work.

E-RR-1 is a thermal hetrogeneous research reactor of Soviet type WW-R serving for radio-isotopes production and physics expriments. Its fuel is uranium enriched to 10% in uranium-235. The uranium fuel is in the form of rods grouped as baskets assebled in an alluminium cylinderical tank filled with distilled light water , which serves for moderation, reflection as well as cooling system in a closed loop forced circulation. A deaerator is connected with the cooling circuit in parallel and circulates 10% of the primary flow to rid the distillated water from evalved combustible gas. The power generated in the reactor is transfered to the ordinary water secondary loop. The biological shield is made of heavy concrete.

The reactor is provided with nine horizontal and nine vertical channels as experimental facilities. The maximum theraml flux is $2x10^{13}$ $n/cm^2/sec$, with a mean of 10^{13} $n/cm^2/sec$.

The measurment of prompt neutron life-time was the aim of the first experiment conducted. The prompt life-time is the average time elapsing between the release of a neutron in a fission reaction and its loss from the system by absorption or escape, assuming all the neutron are prompt. Also, in the same experiment the reactor power was estimated for very low levels.

The increase of the temperature in the reactor core will decrease the density of the reactor moderator and reflector. Also the fuel cross-sections will changed, as well as the leakage probability. This phenomena is known as "Temperature feed-back". The identification of the transfer function of this feed-back was the aim of the second experiment.

The perturbations in the coolant flow, and the irregularties in moderator due to bubbles change the reactor dynamic charateristics. The third experiment was carried out to clearify these actions.

The different method which may be used for reactor parameters identification can be classified into two categories. The first is known as the deterministic methods, the other has the name statistical or adeterministical methods. In the former the reactor is excited with a known reactivity function. These functions may be periodic (e.g. sinusoidal), or aperiodic (e.g. step or ramp).

The statistical methods depend on the analysis of the neutrons dancing, in the reactor, over the average level. This neutrons fluctuation is known as the neutrons noise. Neutrons fluctuate due to the randomnature of fission, random movment of control rods, irregularities in coolant flow, the bubbles in the active zone, or the temperature change in the core. These methods have the advantages that they can be performed without disturbing normal operational mode, no special in-core equipment is required, the measurment is made in relatively short time, and no sefety hazards are expected.

These statistical methods for studying the dynamics (or, kinetics) of a system had been applied in many area before it was used in reactor analysis. The first published investigations (12) treated the theory of the neutrons fluctuations at low power level. In 1957 (3,3,5) noise analysis technique was applied to obtain informations about reactors kinetics.

The statistical analysis may be carried out in time, or frequency domains. In time domain it is necessary to measure the statistical correlation between two random variables (e.g. reactivity ρ and neutron density n) or the correlation of a random variable with itself. The function that characterizes the degree to which value of a varying quantity n, are related to another quantity ρ at some time later is known as cross-correlation. The autocorrelation is a special case of cross correlation, where the two random variable are the same (e.g. n n).

In frequency domain, the interesting functions to be measured, or calculated are the power spectra . These spectra may be measured directly using frequency (or harmonic) analyzer, or may be calculated from correlation functions.

The Fourier transform of the auto and cross correlation functions will yield the power and cross power spectra respectively. The transfer function of any linear dynamic system is related to these spectra.

For a reactor of input reactivity ρ, and output neutron density n, let the power spectra be Φ_{nn} (w) & $\Phi_{\rho\rho}$ (w) respectively. The reactivity power spectrum equal to the neutron power spectrum multiplied by the square of amplitude of transfer function. Also the cross-power spectrum between n and ρ $\Phi_{n\rho}$ (w) is equal to the transfer function of the reactor multiplied by the power spectrum of input reactivity. From the transfer function, the parameters which are to be determined can be identified. The detail discussions and mathematical relations between the spectra and transfer function are included in appendices land 2.

The statistical nalysis may be based on analog or digital techniques using on or off-line principles. In the following study, a mixed anolog and digital techniques are used. For measuring prompt neutron life-time, and for estimating the absolute power level a frequency analyzer is used to obtain the power spectra of neutrons. In the other experiments an analog recording followed by digitization, and off-line digital correlation and power-spectra analysis is used.

This thesis is devided into three chapters in addition to the appendices. The following gives a short description of the contents of each of these chapters.

The definitions of quantities involved, review of mathematical relation discribing statistical nature of fission process, review of the different techniques, theoretical derivations, experimental set up discription, as well as the experimental data and results for measuring the prompt neutron lifetime and absolute power estimation are given in the first chapter.

The second chapter, contains work on modeling. The experimental results of the process of heating the reactor core with a step change of power up to 2MWs are analyzed, and the frequency response is calculated. These results has been compared with the frequency response of the derived theoretical model.

The determination of spectral charateristics of bubbles and pumps noise (due to the irregularity in the flow) is included in the third chapter. The scheme of experimental work is given, as well as the results. The interpretation and discussion of these results, and the comparison with the theoretical calculations is also included.

The algorthims, and computer codes which were written for calculation are given in the appendices. The calculations are carried on IBM1130 and ICL 1905 E computers held at computing centers of Ain-Shams and Cairo Universities, respectively.

CHAPTER I

MEASUREMENT OF PROMPT NEUTRON LIFE-TIME AND ABSOLUTE POWER OF E-RR-1 BY NOISE TECHNIQUE

1-1 Review of Mathematical Relation Describing The Statistical Nature of Fission Process.

1-2 Review For The Methods For Determination of Prompt Neutron Life Time.

1-3 Theoretical Calculations of l.

1-4 Experimental Work For Measuring l.

1-5 Power Estimation.

1-1 Review of Mathematical Relation Describing The Statistical of The Fission Process:-

For any nuclear reactor the power is equivalent to the corresponding rate of fissions of nuclie of the Fuel (Fissions /sec.) For analysing any phenomena concerning the chain reaction in the reactor, we must consider its effect on the neutrons population No.

Let, ℓ (sec) represents the mean time elapces between the neutrons are produced in fission until their loss from the system by absorption or escape. This factor is known as the neutron life time. And k_p, the prompt neutron multiplication factor is the excess of prompt neutrons in a finite reactor from one generation over the proceeding generation per unit of those in the preceeding generation. Hence, the rate of change of neutrons population n at any instant t, which is the change of neutrons per unit time, can be expressed by,

$$\frac{dn}{dt} = \frac{n k_p - n}{\ell} = \frac{-n}{\ell}(1 - k_p) = -\alpha n \quad \ldots.(1)$$

The constant $\alpha = \frac{1-k_p}{\ell}$ is the neutron decay constant, and is known as "Rossi-Alpha" constant,